KEWAUNEE NUCLEAR POWER PLANT

INDIVIDUAL PLANT EXAMINATION

SUMMARY REPORT

DECEMBER 1, 1992

WISCONSIN PUBLIC SERVICE CORPORATION

LIC\NRC\PRA\IPEREP.WP

TABLE OF CONTENTS

SECTION				PAGE
1.0	Executive Summary			1
	1.1	Background and Objectives		1
	1.2	Plant Familiarization		2
	1.3	Overall Methodology		5
	1.4	Summary of Major Findings		6
2.0	Examination Description			13
	2.1	Introduction		13
	2.2	Conformance with Generic Letter and Supporting Material		14
	2.3	General Methodology		15
	2.4	Information Assembly		21
		2.4.1 Containment Building Information		21
		2.4.2 Other PRA Reports Reviewed		21
		2.4.3 PRA Basis Documentation		22
		2.4.4 PRA Walk-Throughs/Verifications		23
3.0	Front-End Analysis			30
	3.1	Accident Sequence Delineation		30
		3.1.1 Initiation Events		30
			3.1.1.1 Large Break LOCA	31
			3.1.1.2 Medium Break LOCA	31
			3.1.1.3 Small Break LOCA	31
			3.1.1.4 Steam Generator Tube Rupture	32
			3.1.1.5 Reactor Vessel Failure	32
			3.1.I.6 Interfacing Systems LOCA	32
			3.1.1.7 Transients With Main Feedwater Available	32
			3.1.1.8 Transients Without Main Feedwater Available	32
			3.1.1.9 Large Steam/Feed Line Break	33
			3.1.1.10 Loss of Offsite Power	33
			3.1.1.11 Station Blackout	33
			3.1.1.12 Anticipated Transient Without Scram Or Main Feedwater	33
			3.1.1.13 Loss of Service Water System	34
			3.1.1.14 Loss of Component Cooling Water System	34
			3.1.1.15 Loss of 125V Emergency DC Bus	34

TABLE OF CONTENTS (Continued)

SECTION			PAGE
		3.1.1.16 Loss of Instrument Air	
	3.1.2/3	Event Trees	37
		3.1.2/3.1 Event Tree Guidelines	37
		3.1.2/3.2 Large Break LOCA	41
		3.1.2/3.3 Medium Break LOCA	45
		3.1.2/3.4 Small Break LOCA	51
		3.1.2/3.5 Steam Generator Tube Rupture	60
		3.1.2/3.6 Vessel Failure	69
		3.1.2/3-7 Interfacing Systems LOCA	71
		3.1.2/3.8 Transients With Main Feedwater	77
		3.1.2/3-9 Transients Without Main Feedwater	84
		3.1.2/3.10 Large Steam Line/Feed Line Break	90
		3.1.2/3.11 Loss of Offsite Power	96
		3.I.2/3.12 Station Blackout	102
		3.1.2/3.13 ATWS Without Main Feedwater	110
		3.1.2/3.14 Loss of Service Water System	117
		3.1.2/3.15 Loss of Component Cooling Water System	121
		3.1.2/3.16 Loss of 125V DC Bus	125
		3.1.2/3.17 Loss of Station and Instrument Air System	130
		3.1.2/3.18 Additional Information Regarding Bases for Top Event Success Criteria	133
		3.1.2/3.19 Scalar Development	146
	3.1.4	Support System Event Tree	149
	3.1.5	Sequence Grouping and Back-End Interfaces	150
		3.1.5.1 Plant Damage States Definitions	151
		3.1.5.2 Containment Safeguards System	152
		3.1.5.3 List of Plant Damage States	155
3.2	System Analysis		173
	3.2.1	System Descriptions	168
		3.2.1.1 Auxiliary Feedwater System	168

TABLE OF CONTENTS (Continued)

SECTION				PAGE
		3.2.1.2	Component Cooling Water	173
		3.2.1.3	Containment Air Cooling	177
		3.2.1.4	Containment Isolation	182
		3.2.1.5	Electrical Power	187
		3.2.1.6	High Pressure Safety Injection	207
		3.2.1.7	Internal Containment Spray	215
		3.2.1.8	Low Pressure Safety Injection	218
		3.2.1.9	Main Feedwater	223
		3.2.1.10	Reactor Protection	228
		3.2.1.11	Service Water	234
		3.2.1.12	Miscellaneous Systems	240
	3.2.2	System Analysis		278
	3.2.3	System Dependencies		280
3.3	Sequence Quantification			305
	3.3.1/2	Data Analysis		305
	3.3.3	Human Failure Data		323
		3.3.3.1	Approach	323
		3.3.3.2	Operator Action Descriptions	340
	3.3.4	Common Cause Failure Data		356
	3.3.5/6/7	Core Damage Quantification		362
	3.3.8	Internal Flooding Analysis		365
		3.3.8.1	Methodology	365
		3.3.8.2	Analysis	366
		3.3.8.3	Quantification of Internal Flooding Initiating Event Frequencies	376
		3.3.8.4	Quantification of Internal Flooding Contribution to Core Damage	379
		3.3.8.5	Quantification of Internal Flooding Contribution to Level 2 Results	380

TABLE OF CONTENTS (Continued)

SECTION				PAGE
	3.4	Results and Screening Process		381
		3.4.1	Application of Generic Letter Screening Criteria	381
		3.4.2	Sequences	381
		3.4.3	Vulnerability Screening	395
		3.4.4	Sensitivity and Importance Analysis	398
		3.4.5	Decay Heat Removal Evaluation	406
		3.4.6	USI and GSI Screening	406
4.0	Back-End Analysis			439
	4.1	Plant Data and Plant Description		439
		4.1.1	Containment Description	439
			4.1.1.1 Containment Structure	439
			4.1.1.2 Containment Systems	441
		4.1.2	Containment Data	442
	4.2	Plant Models and Methods for Physical Processes		446
		4.2.1	Containment Event Trees (CETs)	448
		4.2.2	CET Top Events and Success Criteria	449
		4.2.3	CET Structures and End States	454
		4.2.4	CET Quantification	455
		4.2.5	Containment Failure Characterization	455
			4.2.5.1 Containment Ultimate Strength	456
			4.2.5.2 Likely Containment Failure Modes	456
			4.2.5.3 Unlikely Containment Failure Modes	458
		4.2.6	Post Core Damage Accident Progression	461
	4.3	Kewaunee Level 2 Source Term Analysis		470
		4.3.1	Introduction	470

TABLE OF CONTENTS (Continned)

SECTION				PAGE
	4.3.2	Overview		470
	4.3.3	Source Term Sequence Selection		472
		4.3.3.1	Screening Process	473
		4.3.3.2	Sequence Binning and Selection of Representative Sequences	474
		4.3.3.3	Release Categories	475
	4.3.4	Source Term Analysis		475
	4.3.5	Source Term Results		493
4.4	Kewaunee Level 2 Sensitivity Analysis			509
	4.4.1	Methodology		509
	4.4.2	MAAP Sensitivity Sequences		510
5.0	Utility Participation and Internal Review Team			528
5.1	IPE Program Organization			528
5.2	Composition of Independent Review Team			529
5.3	Areas of Review and Major Comments			533
5.4	Resolution of Comments			534
6.0	Plant Improvements and Unique Safety Features			535
6.1	Level 1 Unique Safety Features			535
6.2	Level 2 Unique Safety Features			535
6.3	Level 1 Plant Improvements			536
6.4	Level 2 Plant Improvements			537
7.0	Summary and Conclusions			539
8.0	References			540

1.0 EXECUTIVE SUMMARY

1.1 Background and Objectives

In response to Generic Letter (GL) 88-20[1] (NOTE: All references are listed in section 8), Wisconsin Public Service Corporation (WPSC) established a full-time permanent group in the Nuclear Licensing and Systems Department responsible for developing and maintaining a Probabilistic Risk Assessment (PRA) for the Kewaunee Nuclear Power Plant. In June of 1989, the Kewaunee PRA was initiated. It was decided to perform a Level 1 PRA for the internal initiating events including internal flooding, and a limited scope Level 2 containment performance analysis. This decision was consistent with GL 88-20 and GL 88-20 Supplement 1[2], which describe the actions necessary for an Individual Plant Examination (IPE).

When this project began, WPSC staff had the limited PRA experience associated with performing a small scale PRA on the auxiliary feedwater system at Kewaunee. For this reason, outside contractor support was obtained to train the WPSC personnel involved in the Kewaunee PRA and to work with them in the initial stages of each portion of the project. Westinghouse Electric Corporation was contracted for Level 1 PRA support, and their IPE partner Fauske and Associates, Incorporated was contracted for the Level 2 containment performance analysis.

WPSC initially staffed the PRA group with one Shift Technical Advisor (STA) certified engineer and one former Senior Reactor Operator (SRO). During this time WPSC performed over 50 percent of the work including the development of all the system fault trees except those associated with reactor protection circuitry, engineered safeguards feature (ESF) actuation circuitry, and diesel generator sequencer circuitry. In September of 1991, WPSC added a third member to the PRA group to provide day to day management of the project and provide additional engineering support. Since that time, WPSC has performed approximately 95 percent of all Level 1 PRA activities and has been working to reach the same level for the containment performance analysis. Each member of the group has operations experience, one member was once a shift supervisor and the other two are former STAs. In addition, the group members have experience in training, licensing, core thermal hydraulics, design modifications, and technical support of the Kewaunee plant.

The objectives for the Kewaunee PRA encompass those presented in GL 88-20 with the addition of several others. These additional objectives are consistent with the intent of the GL, and include:

1. Satisfying the requirements of GL 88-20.

2. Developing a living PRA of the Kewaunee plant which can be used as a tool in decision making for the life of the plant.

3. Gain additional insight in the area of the effects, mitigation, and prevention of severe accidents at the Kewaunee plant.

4. Identify potential improvements in the plant design and/or operation that will reduce the overall core damage frequency and/or the containment failure frequency.

1.2 Plant Familiarization

The Kewaunee Nuclear Power Plant is a 2-loop pressurized water reactor licensed at 1650 MW (thermal). It is located in Kewaunee County, Wisconsin, along Lake Michigan's western shoreline and is jointly owned by WPSC, Wisconsin Power and Light Company, and Madison Gas and Electric Company. Kewaunee is the only nuclear power plant operated by WPSC. The nuclear steam supply system was supplied by Westinghouse Electric Corporation as was the turbine-generator, which is rated at 535 MW (net electrical). The architect/engineer was Pioneer Service and Engineering. Operating license was granted on December 21, 1973. Initial criticality was achieved on March 7, 1974. Initial power generation was reached April 8, 1974, and the plant was declared commercial on June 16, 1974. As of December 31, 1991, Kewaunee has operated with an availability factor of 84.4%.

The following is a summary of some of the important design features at the Kewaunee plant.

1. High Pressure Injection

    - Two 2200 psig centrifugal safety injection (SI) pumps deliver approximately 700 gpm each.
    - Two SI accumulators each containing 1250 ft^3 of borated water are ready to inject if reactor coolant system (RCS) pressure is less than 700 psig.
    - SI pumps require support from the component cooling water and service water systems.

2. Low Pressure Injection

    - Two residual heat removal (RHR) pumps deliver approximately 2000 gpm each when the RCS is depressurized.
    - RHR heat exchangers downstream of each pump provide recirculation heat removal.
    - Recirculation mode takes suction from containment sump B and discharges to the RCS, SI pump suction, and/or containment spray pump suction.

- RHR pumps and heat exchangers require support from the component cooling water system.

- RHR pump fan coil units are supplied by service water.

3. Auxiliary Feedwater

   - Two motor-driven and one turbine-driven auxiliary feedwater (AFW) pumps. Each pump and associated lube oil cooler is cooled by the fluid being pumped.

   - Pumps take suction through a single supply header from the condensate storage tanks.

   - An alternate supply of water to the AFW pumps is provided by the service water system. AFW pump A is supplied by service water train A. AFW pump B is supplied by service water train B. The turbine driven AFW pump can be supplied by either service water train.

4. Emergency Power System

   - Two 4160VAC buses feed two 480VAC buses each.

   - Two diesel generators provide power to the 4160VAC buses should off-site power become unavailable.

   - DC power is provided by four 8 hour station batteries and four battery chargers (2 vital and 2 non-vital).

   - Vital instrument power is provided to four instrument buses from the 480VAC buses by way of 480VAC/120VAC instrument bus transformers, or from the vital DC system by way of four instrument bus inverters.

5. Component Cooling

   - Consists of two pumps, two heat exchangers and one surge tank.

   - Cools reactor coolant pumps (RXCPs), RHR pumps, and SI pumps.

   - Component cooling heat exchangers are cooled by service water.

6. Service Water

- Consists of two normally cross-connected headers with two pumps in each header. These two headers are isolated from each other by a SI signal and thereby made separate and independent.

- Cools component cooling heat exchangers, containment fan coil units, SI pump lube oil heat exchanger and stuffing box, diesel generator coolers, safeguard fan coil units, and provides the emergency water supply to the AFW pumps.

7. Containment

- Large, dry type

- Primary containment consists of a low leakage steel vessel.

- Secondary containment consists of a medium leakage concrete shield building surrounding the primary containment vessel.

- Containment vessel free volume is 1.32×10^6 cubic feet.

- The containment vessel design pressure is 46 psig and design temperature is 268°F.

8. Containment Spray

- Two independent spray headers with one pump in each header deliver 1300 gpm each.

- Water is supplied by the refueling water storage tank (RWST) and the sodium hydroxide tank initially. When the RWST is depleted recirculated fluid can be supplied to internal containment spray (ICS) pump suction from the RHR pumps.

9. Containment Fan Coil Units

- Four containment fan coil units, two supplied by each service water header.

- Service water is supplied at maximum flow during accident conditions.

10. Chemical and Volume Control

- Three positive displacement/air cooled charging pumps provide 60.5 gpm each for reactor coolant system (RCS) makeup and RXCP seal injection.

- One of the three charging pumps is provided with a variable DC drive for speed control and is therefore not dependent on instrument air for attaining maximum pump output.

- Two boric acid transfer pumps provide the capability of supplying concentrated boric acid to the suction of the charging pumps for reactivity control.

1.3 Overall Methodology

The Kewaunee PRA was performed using standard PRA techniques. The small event tree - large fault tree methodology was used in the Level 1 PRA. Systemic event trees were developed to define the possible accident scenarios for each specific initiating event. In general, the event trees were developed using an approach similar to that outlined in the PRA Procedures Guide [(3)]. Detailed fault trees were created for each front-line system identified in the logic of the event tree. Equally detailed fault trees were developed for the front-line systems' support systems including all of the actuation systems associated with reactor protection and engineered safety features. In addition, detailed models of the feedwater and instrument air systems including all of their non-safety related support systems were developed. Success criteria for the system fault trees were chosen to be consistent with the design and licensing basis of the Kewaunee plant.

A combination of generic and Kewaunee plant specific data was used in determining the initiating event frequencies and the equipment failure probabilities. Fifteen years of plant data from sources such as Incident Reports, Licensee Event Reports, and Work Requests was reviewed in making these determinations. Common cause failure modes were modeled using the Multiple Greek Letter (MGL) method. Human Reliability Analysis (HRA) was performed using the Technique for Human Error Rate Prediction (THERP) methodology. The operator actions that were modeled follow the Kewaunee Integrated Plant Emergency Operating Procedures (IPEOPs).

The Westinghouse software WLINK was used to perform the core melt quantification and the plant damage state quantification. The plant damage state quantification serves as a link between the Level 1 PRA and the containment performance analysis.

Sensitivity studies were performed on the model to determine the variability in the core damage frequency as influenced by such factors as changes in the cutoffs, operator actions, common cause, etc. Importance analyses were performed to identify the important accident sequences, system failures, component failures, and operator errors that contribute to the core damage frequency. Detailed notebooks were developed for each section of the Kewaunee PRA to provide documentation of the decisions and assumptions that served as input to the models and calculations.

Level 1 results were then grouped into containment event tree (CET) end states. Representative sequences from each CET end state were analyzed with the Modular Accident Analysis Program (MAAP). Results of MAAP runs were used to determine radioactive release characteristics for each CET end state. These end states were then assigned release categories. The frequencies of the release categories are the end result of Level 2 analysis. Sensitivity studies were run in which the parameters suggested in EPRI TR-100167[46] were varied. Results showed these parameters had very little effect on the release characteristics.

One of the important steps in the Kewaunee PRA project was the extensive reviews performed by the WPSC PRA staff, independent WPSC reviewers, and independent external reviewers. The PRA group members thoroughly reviewed the results of every iteration of the core melt quantification using their operations background to identify invalid cutsets. The models were reviewed to identify the problem or problems that caused the invalid cutset, and then the problems were corrected. An independent group of experienced Kewaunee plant staff members performed an extensive review of the different phases of the Kewaunee PRA, and identified numerous improvements that were then incorporated into the PRA. This group of individuals was provided with three days of training on PRA by the PRA group and the contracted vendors. One additional review was performed by a team completely independent of WPSC personnel and the contracted vendors. This review was performed by a total of 6 individuals from 3 different companies plus a member of the Wisconsin Electric Power Company PRA staff. This external review focused on the methodology and assumptions used in the Kewaunee PRA, and resulted in several improvements.

1.4 Summary of Major Findings

As stated earlier, the Kewaunee PRA does not deviate from the plant abnormal and emergency procedures. Therefore, there are no accident management actions modeled. By not taking credit for these actions, the results provided in this report are higher than they would be had credit been taken. The overall core damage frequency for the Kewaunee plant was determined to be 6.65E-5/yr considering internal events including flooding. This value is consistent with the results of other plants. It does, however, reflect the conservative approach taken by WPSC. Figure 1.4-1 provides a summary of the contribution to the overall core damage frequency by the different initiating events.

The initiating event providing the largest contribution to core damage frequency is the station blackout event. The core damage frequency associated with this event would have been higher had not WPSC committed to plant modifications in response to the station blackout rule. The station blackout rule modifications, which include providing backup power via the non-safeguard technical support center (TSC) diesel generator to a charging pump for reactor coolant pump seal injection, are currently scheduled to be completed prior to startup from 1993 refueling outage. The dominant cutsets involve failure of the turbine driven auxiliary feedwater pump or the TSC diesel generator.

The second most important initiating event in terms of contribution to core damage frequency is the small LOCA. The dominant cutsets involve loss of the emergency core cooling system primarily by common cause failures and operator errors. The medium LOCA event is the third most important event in contribution to the overall core damage frequency. The important cutsets for this event are similar to those for a small LOCA. The primary reason for the small LOCA core damage frequency exceeding that for a medium LOCA is that the initiating event frequency for the small LOCA is approximately twice as large.

The fourth most important initiating event is the steam generator tube rupture event. The dominant cutsets for this event involve a failure to cooldown and depressurize the RCS which is greatly influenced by dependencies between the different operator actions. This event is of particular interest because it may involve a containment bypass allowing fission products to be released directly to the environment.

The four initiating events described above contribute over 80 percent of the overall core damage frequency. The remaining initiating events and their contribution to overall core damage are discussed in detail later in this report. Although these other initiating events do not play as large a role in the overall core damage frequency for Kewaunee, there are some vulnerabilities identified in the PRA that are significant enough to warrant an improvement. Table 1.4-1 lists the improvements that have been made or are scheduled to be made resulting from the IPE and the initiating event associated with the improvement. Although some of these improvements have not been made as of this date, they are reflected in the Kewaunee PRA because they have been approved by WPSC management, and are scheduled to be completed in the near future.

Results of the Kewaunee back-end analysis (See Figure 1.3-2) show that, should a core damage event occur, there is a 92% probability of containment success. In other words, there is a 92% probability that the final barrier to fission product release is not breached, impaired, or bypassed within the 48 hour mission time. However, the majority of the core damage events at Kewaunee (51%) require some additional recovery actions not credited in the IPE in order to prevent containment failure at some time beyond 48 hours.

Based on performance of the Level 1 PRA analysis, several features of the Kewaunee design have been identified that reduce the likelihood of core damage. These include:

- High head safety injection pumps inject at 2200 psig which is significantly higher than typical Westinghouse plants designated as low pressure plants.

- Containment sump recirculation can be aligned to the high head safety injection, low head safety injection and containment spray pumps from the control room.

- Three auxiliary feedwater (AFW) pumps (two motor-driven and one turbine driven for diversity), which are independent of cooling water systems. The service water system serves as a backup suction supply to the three AFW pumps.

- Separate eight hour batteries for safeguards and non-safeguards equipment.

- Four safety related service water pumps for a single unit site.

- The chemical volume and control system has three positive displacement charging pumps which are independent of cooling water systems. One of the pumps is driven by a variable speed DC motor for speed control and is not dependent on instrument air for attaining maximum pump output.

- Two independent methods for maintaining reactor coolant pump seal integrity, seal injection from the charging pumps and thermal barrier cooling via the component cooling water system.

The Level 2 results show a robust containment design capable of responding to accidents well beyond design basis. The redundant containment heat removal capability combined with physical design features, including free volume, cavity geometry, and floor areas contribute to the containment's capabilities. Results show that Kewaunee does not exhibit any Level 2 vulnerabilities.

FIGURE 1.4-1

CORE DAMAGE FREQUENCY IMPORTANCE BY INITIATING EVENT

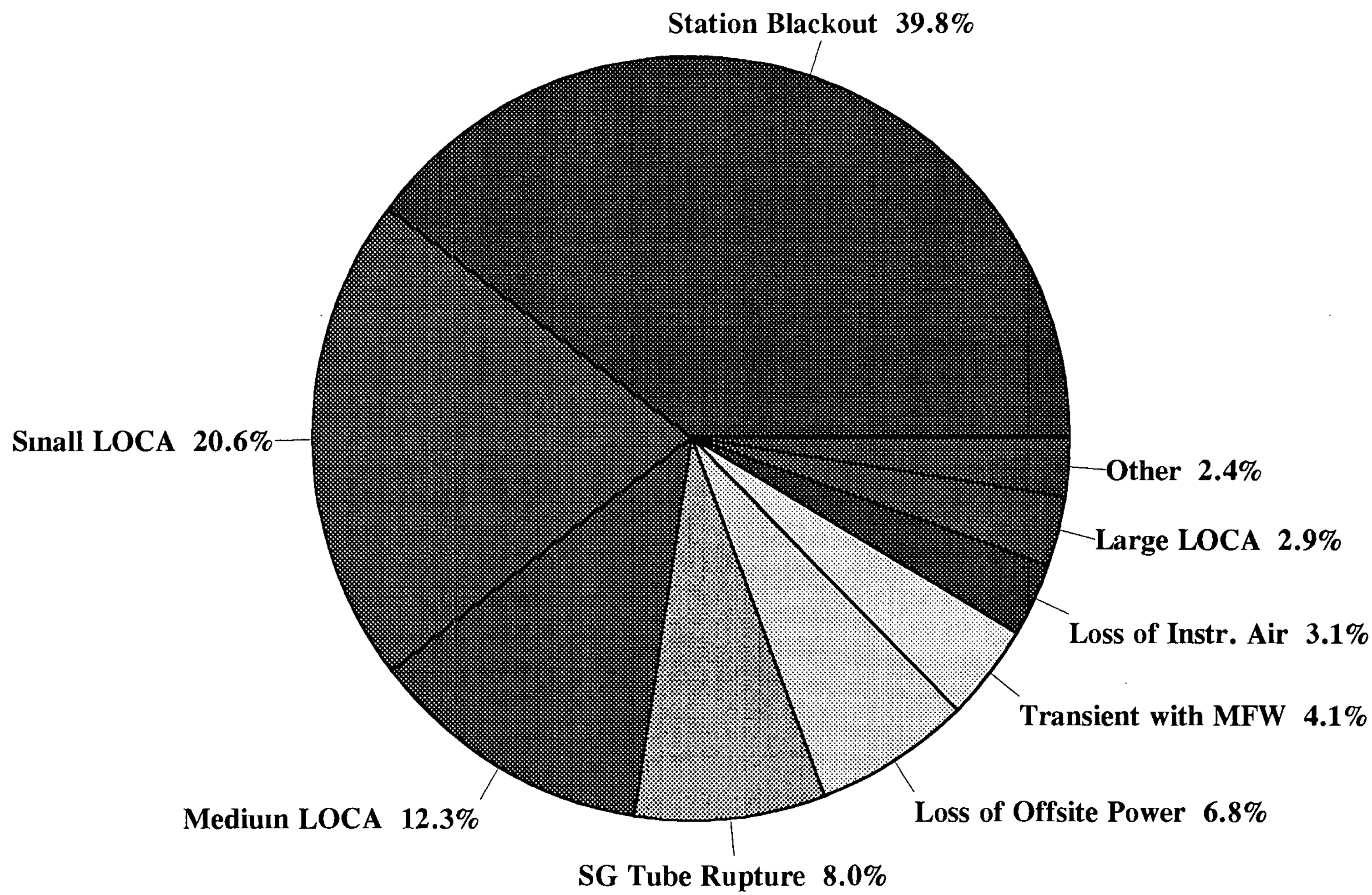

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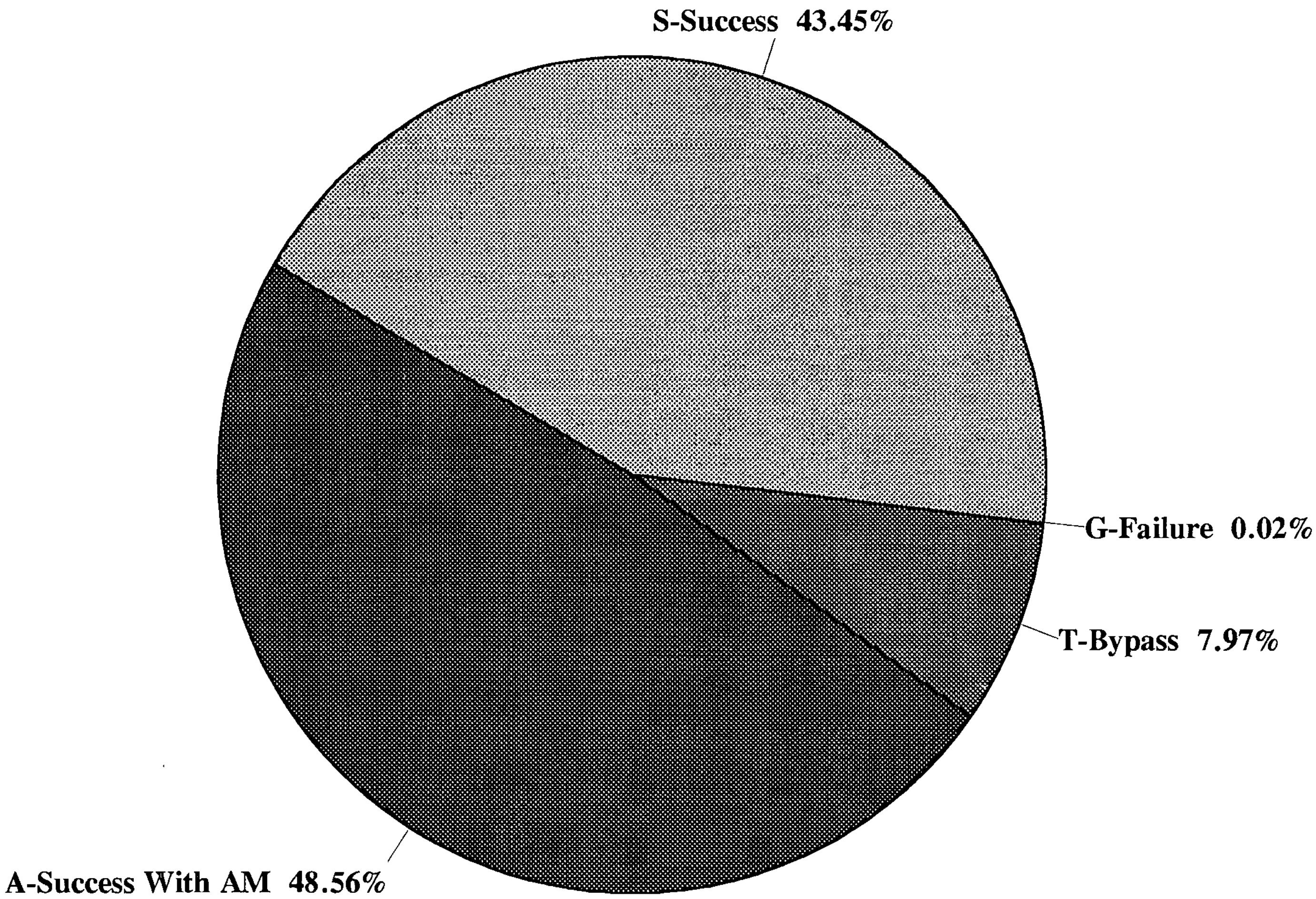
TABLE 1.4-1
PLANT IMPROVEMENTS INITIATED BY THE IPE

IMPROVEMENT	INITIATING EVENT	SCHEDULE
Perform leak testing of an additional four valves serving as a boundary between the reactor coolant system and a low pressure system	Interfacing Systems LOCA	End of Refueling Outage 1993
Modify the normal position of two motor operated valves located on the low pressure safety injection line from open to closed	Interfacing Systems LOCA	End of Refueling Outage 1994
Modify emergency operating procedure ECA 1.2 to improve guidance to the operators in identifying and mitigating an interfacing systems LOCA	Interfacing Systems LOCA	Summer 1993
Modify the swing direction of three doors separating the turbine building basement with areas containing safeguards equipment in order to reduce the likelihood of a turbine building basement flood propagation into those other areas.	Internal Flooding	2 Doors in 1992 Refueling Outage 1 Door in 1993 Refueling Outage
Improved the inspection method for rubber expansion joints to identify possible flooding problems before they occur	Internal Flooding	1992 Refueling Outage
Modify emergency operating procedures to provide instruction for switching the power supply to bus 5262 in the event of the loss of either safeguards bus 5 or 6 in order to have power available to 2 instrument air compressors	Loss of Offsite Power, Station Blackout	Summer 1993

FIGURE 1.4-2

RELEASE CATEGORIES



Release Categories are Defined in Table 1.4-2

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TABLE 1.4-2

RELEASE CATEGORY DEFINITIONS

Release Category	Definition
A	No containment failure occurs with 48 hour mission time but failure could eventually occur without accident management action; noble gases and less than 0.1% volitiles released.
G	Containment failure prior to vessel failure with noble gases and up to 10% of the volitiles released (containment isolation impaired).
S	No containment failure (leakage only, successful maintenance of containment integrity; contaimment not bypassed; isolation successful).
T	Containment bypassed with noble gases and more than 10% of the volitiles released.

2.0 EXAMINATION DESCRIPTION

2.1 Introduction

The objectives of the Kewaunee PRA are to:

1. Satisfy the requirements of Generic Letter 88-20.

2. Develop a living PRA of the Kewaunee plant that can be used as a tool in decision-making for the life of the plant.

3. Gain additional insight in the area of the effects, mitigation, and prevention of severe accidents at the Kewaunee plant.

4. Identify potential improvements in the plant design and/or operation that will reduce the overall core damage frequency and/or the containment failure frequency.

WPSC has met these objectives by performing a Level 1 PRA for the internal initiating events including internal flooding, and a limited scope Level 2 containment performance analysis. In order to satisfy the second and third objectives, WPSC established a three person PRA group which has performed over 50% of the IPE effort. As the project progressed and the WPSC PRA group gained experience, they began to perform a greater portion of the work such that the WPSC staff has performed approximately 95% of the Level 1 PRA activities over the past year. The support that WPSC received was from Westinghouse Electric Corporation for the Level 1 PRA and Fauske & Associates, Incorporated for the Level 2 PRA.

The Kewaunee PRA consists of the following major tasks:

<u>Level 1 PRA</u>

1. Plant Definition and Information Gathering
2. Initiating Event Analysis
3. Accident Sequence Analysis
4. Plant Systems Analysis
5. Database Development
6. Human Reliability Analysis
7. Dependency and Common Cause Failure Analysis
8. Core Melt Quantification
9. Plant Damage States Quantification
10. Internal Flooding Analysis
11. Sensitivity and Importance Analysis
12. Training and Technology Transfer

Level 2 PRA

1. Containment Systems Analysis
2. Containment Structural Capability Assessment
3. Containment Event Tree Quantification
4. Source Term Analysis
5. Sensitivity Analysis

2.2 Conformance with Generic Letter and Supporting Material

NRC Generic Letter 88-20, which was issued on November 23, 1988, requested that licensees perform an Individual Plant Examination for severe accident vulnerabilities. Supplement 1 to GL 88-20 initiated the examination process and announced the availability of NUREG-1335 (4) which provides the guidance for reporting the results of the IPE. Supplement 2 to GL 88-20 (5) provided sample accident management strategies for consideration during performance of the IPE. Supplement 3 to GL 88-20 (6) announced the completion of the NRC Containment Performance Improvement Program and transmitted insights from the program for use in the IPE process.

As stated in section 2.1, the number one objective for the Kewaunee PRA was to satisfy the requirements of GL 88-20. Specifically the stated purpose of GL 88-20 was to (1) develop an appreciation of severe accident behavior, (2) understand the most likely severe accident sequences that could occur at a plant, (3) gain a more quantitative understanding of the overall probabilities of core damage and fission product releases, and (4) reduce the overall probabilities of core damage and fission product releases by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents. On November 1, 1989, WPSC submitted a letter (7) outlining the proposed Kewaunee PRA program to satisfy the requirements of GL 88-20 and GL 88-20 Supplement 1. The NRC responded in a letter dated January 17, 1990 (8) that the WPSC approach, methodology, and schedule were acceptable.

Additional requirements contained in GL 88-20 are listed below along with a discussion of how the Kewaunee PRA met the requirement or a reference to another section of this report which discusses this requirement further.

1. The licensee staff should be used to the maximum extent possible in the performance of the IPE.

   Refer to section 5 for details.

2. Unresolved Safety Issue (USI) A-45, "Shutdown Decay Heat Removal Requirements," should be resolved as part of the IPE.

   Refer to section 3.4.3 for details.

3. Any other USI that can be addressed by the IPE should be identified and resolutions should be proposed.

   Refer to section 3.4.4 for details.

4. Vulnerabilities identified during the IPE process should be corrected where appropriate.

   Refer to section 6 for details.

5. The containment analysis should include consideration of the insights gained from the NRC Containment Performance Improvement Program.

   Refer to section 4 for details.

6. The results of the IPE should be reported in a format consistent with NUREG-1335.

   This submittal follows the outline provided in NUREG-1335 except for minor exceptions noted in section 3.

2.3 General Methodology

In order to maintain an organized and comprehensive approach in the IPE process, separate tasks were clearly defined and are summarized in this section.

1. Plant Definition and Information Gathering - This task involved the identification and collection of essential plant reference material needed to support the development of the plant-specific probabilistic models. This information was used to develop system notebooks, to model accident sequences (event trees and fault trees), and to identify critical plant systems, initiating events, and system dependencies.

   Some examples of the information that was collected include logic diagrams, process and instrumentation diagrams (P&IDs), electrical "one-line" diagrams, system descriptions, maintenance procedures, test procedures, equipment location, plan and elevation drawings, piping and HVAC layout drawings, normal and emergency operating procedures, plant operating history, and plant-specific reports such as the Updated Final Safety Analysis Report, technical specifications, plant-specific transient and LOCA analyses, and other relevant reports.

2. Initiating Events Analysis - The initiating event definition task identified, categorized, and quantified all accident initiating events for which event tree models were developed. These initiating events were added to a list of generic accident initiators and were included in the event tree modeling task. Initiating events were identified by conducting a comprehensive review of previous PRAs for similar plants and by examining design

data and the operating history of Kewaunee. These examinations covered plant trip events that can or have occurred at Kewaunee that are not included in the list of generic initiating events.

The second purpose of this task was to examine plant-specific and generic data to determine the proper frequencies to be applied to initiating events for Kewaunee. This was considered both for the generic initiating events and for the special events identified. The types of events that were considered included expected plant transients, LOCAs, and loss of key plant support systems.

3. Accident Sequence Analysis - During this task, plant-specific event tree models were developed for each of the events identified in the initiating events task. The development of event trees consisted of a series of distinct steps. These were as follows:

   - Definition of critical safety functions relevant to the initiating event

   - Development of system event trees based on plant systems capable of performing the safety functions in the functional event trees

   - Modeling of operator actions and consequential failures related to the various accident sequences in the event trees

   - Development of system success and failure criteria for the various accident sequences

4. Plant System Analysis - During this task the system failure probabilities for all systems needed to quantify the system level event trees defined in the event tree development task were calculated. For this purpose, fault tree modeling and other engineering methods were used to calculate or estimate the system failure probabilities. The system failure criteria defined in the event tree task were used to determine all necessary top event definitions for each system fault tree mode. All trees were constructed according to the guidelines described in the Westinghouse Fault Tree Guidelines so that a consistent level of modeling detail and modeling technique was ensured in all fault trees. Modeling detail was extended to the level of system components, including instrumentation and control system faults.

   The fault trees were developed to the component level by using the GRAFTER code system. In addition, applicable failure modes of a given component were included by using the SIMON data manager function of the GRAFTER code system. The fault tree linking model in the WLINK code system was used to link the fault trees to obtain core melt accident sequences. For this purpose, fault trees were developed for frontline and support systems individually. The frontline system fault trees contained the support systems as subtrees which were linked into the frontline fault trees. Each component was given a unique basic event identification and may appear in multiple fault trees.

Thus, fault tree linking identified and properly treated components (or support systems) that may be common to different safety systems. For each component, random failure, test and maintenance unavailability, human errors, and common cause failure were modeled if applicable.

5. Database Development - Some plant-specific equipment failure data was collected during the initial data collection phase, but this effort was limited to the assembly of readily available data from such sources as incident reports, LERs, maintenance work requests, operator logs, NPRDS, Kewaunee Diesel Generator Reliability Study, Kewaunee Auxiliary Feedwater PRA and other existing equipment data bases at the plant. The purpose of the data collection task was to gather sufficient equipment performance and availability data from plant-specific sources to accurately quantify the PRA logic models.

   Plant-specific information was collected from a variety of work requests, control room logs, and completed surveillance test procedures in order to identify and examine plant-specific component failure, testing, and maintenance data and data related to initiating events that have led to reactor trips. The Kewaunee PRA used plant-specific data to calculate failure rates through classical means or through the use of Bayesian techniques. In some instances, generic data from NUREG-4550[9], IEEE-500[10], NUREG/CR-2728[11], WASH-1400[12] or other sources were used to supplement plant data.

6. Human Reliability Analysis - The human reliability analysis task established suitable models to represent the interaction of operators and other plant staff with plant systems and equipment during normal operation and during transient and accident conditions. For this task, those human tasks important to the analysis were identified, and the full range of plant procedures was examined to determine the types of human actions that are routinely performed and what kinds of actions operators are trained to take. All accident sequences and system failure modes developed in the event and fault tree models were carefully evaluated to determine those areas where operator intervention can, should, and must occur. Finally, the kinds of errors in all identified human actions critical to the analysis of plant risk were assessed in the Kewaunee PRA.

   After potentially important human errors were identified, detailed models were developed and were quantified so that their effects could be incorporated into the event and fault tree models. The Technique for Human Error Rate Prediction (THERP) methodology was used for the human reliability analysis.

7. Dependency and Common Cause Failure Analysis - The dependency and common cause analysis task determined both qualitatively and quantitatively, those dependent failure events that significantly effect the failure modes and failure probabilities of both frontline and support systems.

The evaluation of dependent failures was included in the PRA in several ways. In cases in which dependencies were clearly identified (the reliance of frontline systems on support systems, for example), the dependencies were included explicitly in the event and fault tree models. For situations in which dependencies are likely to exist but are not known or easily recognized, the importance of potential dependencies was determined through parametric modeling and sensitivity evaluations. Finally, qualitative evaluations of dependencies were developed from plant walkdowns, careful examinations of various plant operations and maintenance procedures, and from the subtle interactions list from NUREG-4550.

The parametric modeling and sensitivity evaluations used quantitative techniques to determine the possible bounds of influence of unknown dependencies and common cause effects. Parametric modeling was done using the Multiple Greek Letter method, an extension of the β-factor method.

8. Core Melt Quantification - The quantification task determined the unavailabilities associated with the frontline and support systems' fault tree models, and used these results to quantify the system level event trees. The fundamental products of this task were accident sequence cutsets, core damage frequencies for all the accident sequences, and identification of dominant accident sequences from among all event tree results.

   The quantification task involved a coordinated and systematic combination of information from the preceding analysis tasks. Though the task was primarily a computational task involving large amounts of computer analysis, it was not a once-through effort. It required numerous iterations in which fault trees and accident sequence results were examined for logical consistency, suitable accuracy, and consistent level of detail across the spectrum of the analysis. In cases in which problems were uncovered, the supporting information was reviewed to identify and correct the source of problems, and then the quantification process was repeated until the critical problems and issues were resolved. The WLINK Code system was used to perform the core melt quantification.

9. Plant Damage State Quantification - The definition of the plant damage states is the major link between the Level 1 PRA (plant systems analysis) and the limited scope Level 2 PRA (containment and fission product analysis). The plant damage states become the binning criteria for the plant systems analysis; the definition of the plant damage states represents the minimum set of key parameters that define differences in accident sequences that can impact the containment and fission product behavior for a core damage accident. To identify the minimum set of key parameters, a thorough understanding of the physical and chemical phenomena that can occur during a severe accident, and an evaluation of the expected plant response (considering those phenomena), was required. The impact of the operator actions taken before and during the core degradation process on the accident, as well as the impact of systems availabilities, were also considered in the definition of the plant damage states.

Endpoints of the plant event trees are termed "plant damage states". These endpoints link the beginning of the containment event trees to the fission product release analyses. The plant damage states represent the minimal unique set of accident sequence characteristics, for those sequences which lead to core damage, that can be used for binning of the core damage accident sequences. The definition of the plant damage states takes the form of a core damage sequence and a set of containment safeguard systems. All core damage accident sequences within a given plant damage state are expected to result in a similar accident progression from the standpoint of the containment response and the fission product behavior.

10. Internal Flooding Analysis - This task involved the completion of a plant specific internal flooding assessment as required by Generic Letter 88-20 to determine if the plant is susceptible to flooding events that could potentially lead to core damage. The assessment was divided into two parts: a qualitative assessment, which identified potential internal flooding events, and a quantitative assessment, which calculated the frequency of core damage given an internal flooding event. A separate walkdown was performed for the internal flooding event. The appropriate event trees from the other internal initiating events were used to quantify the contribution of flooding to the core damage frequency.

11. Sensitivity and Importance Analyses - The response of the core damage frequency to changes in input parameters and modeling assumptions 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.

13. Containment Systems Analysis - Quantitative models were developed and evaluated for containment systems, containment isolation failure events, and containment bypass events as an integral part of the systems analysis effort. The plant damage states incorporated the results of the containment systems analyses with other results regarding the state of the plant systems, the physical state of the core, the reactor coolant system, and the containment boundary.

14. Containment Structural Capability Assessment - Existing and updated structural analyses were used to determine the containment ultimate pressure capability. This effort used a substantial database that already existed and evaluated the ultimate capability based on a quasi-static method. The search for plant unique features (piping penetrations, electrical penetrations, etc.) was performed during two plant walkdowns.

15. Containment Event Tree Analysis - A containment event tree (CET) was developed to provide a systematic method for integrating the Level 1 results with the Level 2 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.

16. Source Term Analysis - This task defined and quantified the radionuclide release categories associated with the Kewaunee plant. Release categories often serve as surrogates for the risk measures typically represented in consequence analyses. The release categories were represented in terms of the magnitude, warning time, duration and type of radionuclide involved in the release as well as by the frequency of the releases. The set of CET end states produced in CET task were "binned" into the several release categories, and the source term magnitude and frequency of each category was assessed based upon available analyses.

    To quantify the magnitude of the radionuclide releases, the CET was applied to each plant damage state and a set of CET end states were produced. Those CET end states that represent dominant risk contributors needed to be quantified by plant specific analyses; this was done using the Modular Accident Analysis Program (MAAP). The analyses were performed using uncertainty analyses consistent with the NRC recommendations.

17. WPSC defines a vulnerability as a feature in plant design, procedures, training, etc., which results in a contribution to core melt risk greater that what is expected. Placing strict criteria in defining a vulnerability is not practical; however, Generic Letter 88-20 provides the following guidance:

    1. Any functional sequence that contributes 1E-6 or more per reactor year to core damage,

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

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

    4. Functional sequence that contribute to a containment bypass frequency in excess of 1E-7 per reactor year, or

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

ECP 5.10 also discusses the treatment of vulnerabilities with the overall theme being to implement strategies to correct the plant feature identified as a vulnerability if determined to be necessary. Possible solutions are: plant modifications, maintenance improvements, procedural changes, training program changes, and others. Another possible solution is consideration of the particular vulnerability in the accident management guidelines that will be developed as part of the WPSC accident management program. In most cases, some form of a cost benefit assessment is made to determine if a change is warranted.

18. Review Program - In all stages of the Kewaunee PRA, numerous levels of review were performed to ensure accuracy and completeness. Extensive reviews were performed by the WPSC PRA staff, independent WPSC reviewers, and independent external reviewers. The PRA group members thoroughly reviewed the results of every core melt quantification iteration using their operations background to identify invalid cutsets. The models were reviewed to identify the problem that caused the invalid cutset, and then the problems were corrected. A independent group of experienced Kewaunee plant staff members performed an extensive review of the different phases of the Kewaunee PRA, and identified numerous improvements that were made to the PRA. This group of individuals was provided three days of training on PRA by the PRA group and the contracted vendors. One additional review was performed by a team completely independent of WPSC personnel and the contracted vendors. This review was performed by a total of 6 individuals from 3 different companies plus a member of the Wisconsin Electric Power Company PRA staff. This external review focused on the methodology and assumptions used in the Kewaunee PRA, and it resulted in several improvements.

2.4 Information Assembly

2.4.1 Containment Building Information

Most of the plant layout and containment building information used in the Kewaunee PRA is contained in the Kewaunee Updated Final Safety Analysis Report (USAR).[14] Figures 2.4-1 through 2.4-6, however, are included for the completeness of this report. Additional information on the Kewaunee containment is found in section 4.

2.4.2 Other PRA Reports Reviewed

Several PRA reports were reviewed by the WPSC PRA staff during the performance of the Kewaunee PRA. Portions of the following studies were reviewed:

1. IPE studies already submitted to the NRC:
    a. Surry
    b. Turkey Point
    c. Diablo Canyon

d. DC Cook
e. Seabrook
f. Millstone 3

2. IPE studies currently under development or recently submitted to the NRC:
    a. Zion
    b. Farley
    c. V.C. Summer
    d. Vogtle
    e. Pt. Beach
    f. Ginna
    g. Wolf Creek

3. Other PRA studies:
    a. WASH-1400
    b. NUREG-4550 for Zion, Sequoyah and Surry
    c. NSAC-60 for Oconee Unit 3[15]
    d. NUREG-4458, Decay Heat Removal study for Pt. Beach[16]

As stated earlier in this report, the WPSC PRA staff had limited PRA experience prior to performing the Kewaunee PRA; for this reason numerous other PRA studies were reviewed during all phases of the project. The greatest benefit was for our staff to gain a greater appreciation for the typical orders of magnitude for occurrence of events, failure probabilities, etc. This proved to be very beneficial when reviewing the Kewaunee PRA results. If the Kewaunee PRA results differed significantly from the typical values in the industry, a close review was performed to determined whether the difference reflected a unique Kewaunee feature or if there was an error in the Kewaunee PRA models. Another area that review of other PRA studies aided was in the determination of initiating event frequencies for events that have never happened at Kewaunee.

2.4.3 PRA Basis Documentation:

Numerous sources were used in developing the Kewaunee PRA, the plant documentation used included:

- USAR
- Technical Specifications
- Operating Procedures
- Licensee Event Reports
- Piping and Instrumentation, General Arrangement, and Electrical One-Line Drawings
- System Descriptions
- Operator Surveys
- Plant Walk-Throughs

- Maintenance Records and Procedures
- Design Change Packages

The component failure rates, initiating event frequencies, and component test and maintenance data were based on Kewaunee plant operating experience, supplemented with industry data where needed. The Kewaunee PRA has been kept up-to-date with plant modifications and procedural changes in order to maintain it as a "Living PRA".

2.4.4 PRA Walk-Throughs/Verifications:

Having an experienced, Senior Reactor Operator (SRO) or Shift Technical Advisor (STA) trained PRA staff afforded WPSC several benefits. One in particular was in the need, or lack thereof, for official system walkdowns. The WPSC PRA staff did not need to familiarize themselves with the systems, equipment, or equipment locations since they had already had years of operations training and experience. Two of the three members of the PRA staff were active STAs for most of the PRA project, and were therefore able to regularly tour the plant and observe modifications first-hand. Another advantage was that the Kewaunee plant and the corporate offices in Green Bay where the PRA group is headquartered are only 30 miles apart. This allowed a PRA member to easily visit the plant to visually confirm information whenever necessary.

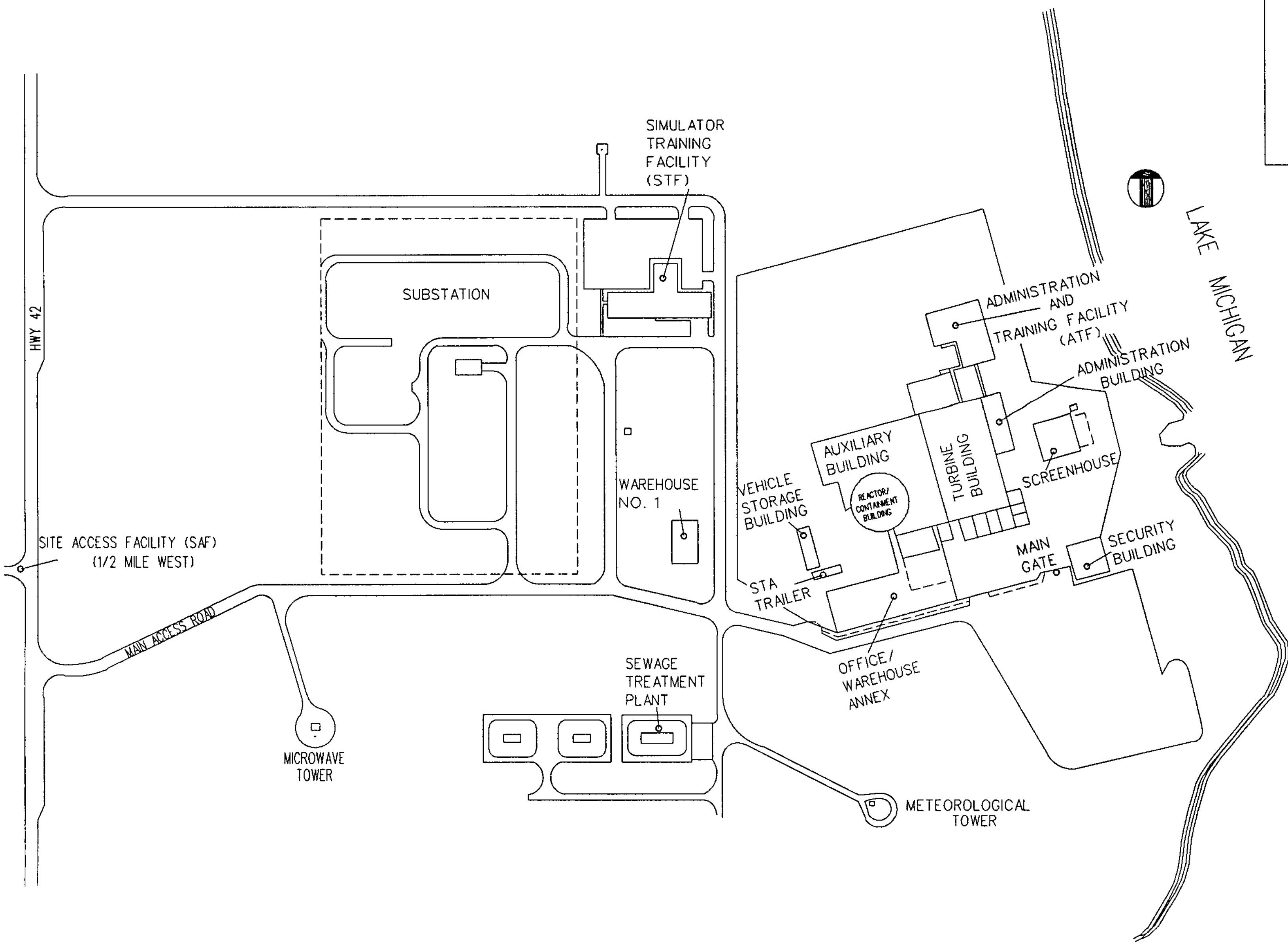
Official plant walkdowns with the PRA contractor staff involved were conducted in two areas: internal flooding and the Level 2 PRA.

Internal Flooding - The internal flooding walkdown was performed by two members of the WPSC PRA staff and the internal flooding analyst from Westinghouse. The purpose of this walkdown was to confirm flood sources and their propagation paths, detection, and potential impact. In addition, the walkdown confirmed spatial relationships between flood sources and important equipment.

Level 2 - Two walkdowns were performed by one member of the WPSC PRA staff and a Level 2 analyst from Fauske and Associates, Incorporated. The walkdowns were used to support decisions made in the Level 2 analysis.

FIGURE 2.4-1

SITE LAYOUT



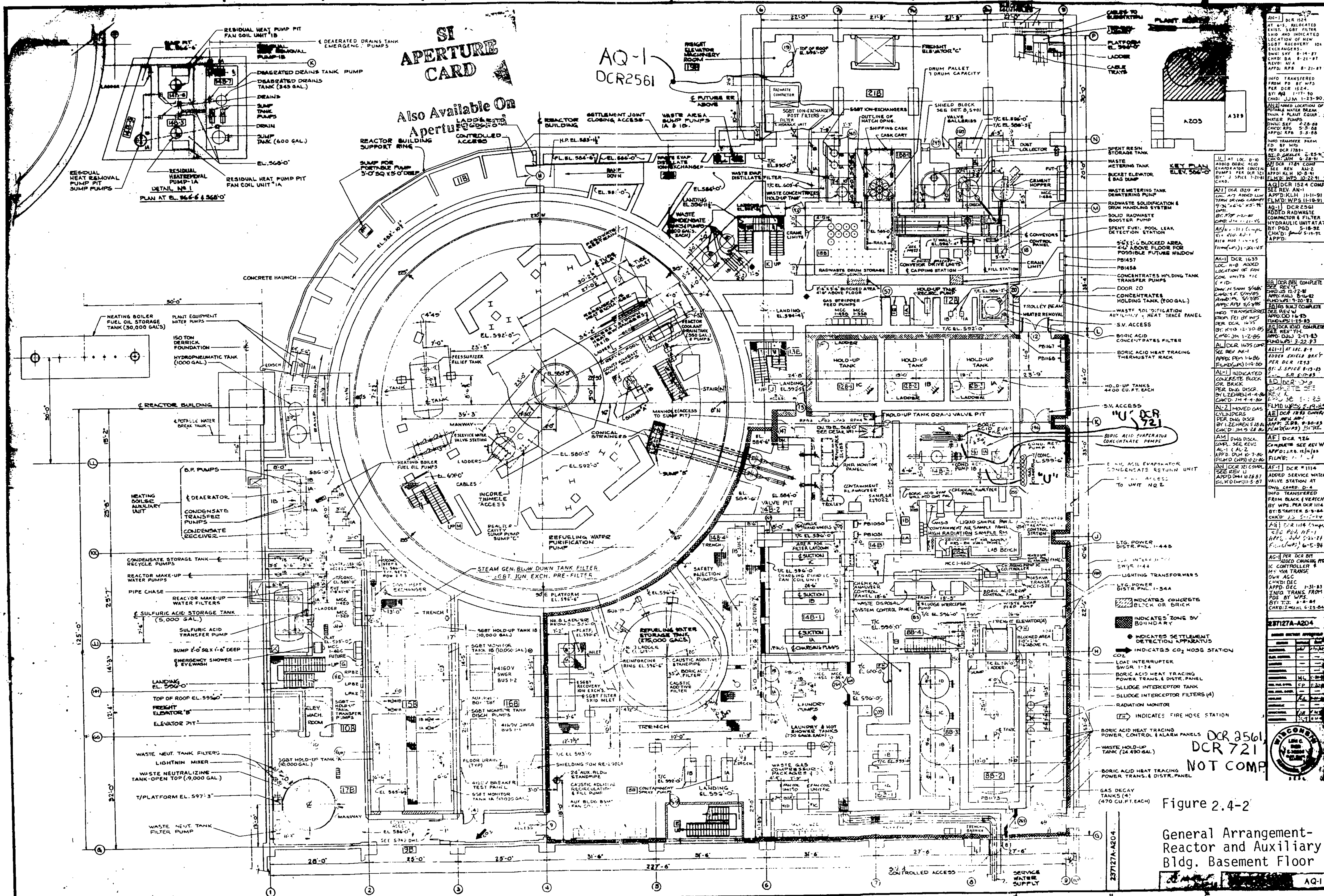


Figure 2.4-2

General Arrangement-Reactor and Auxiliary Bldg. Basement Floor

9212090118-01

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Aperture Card

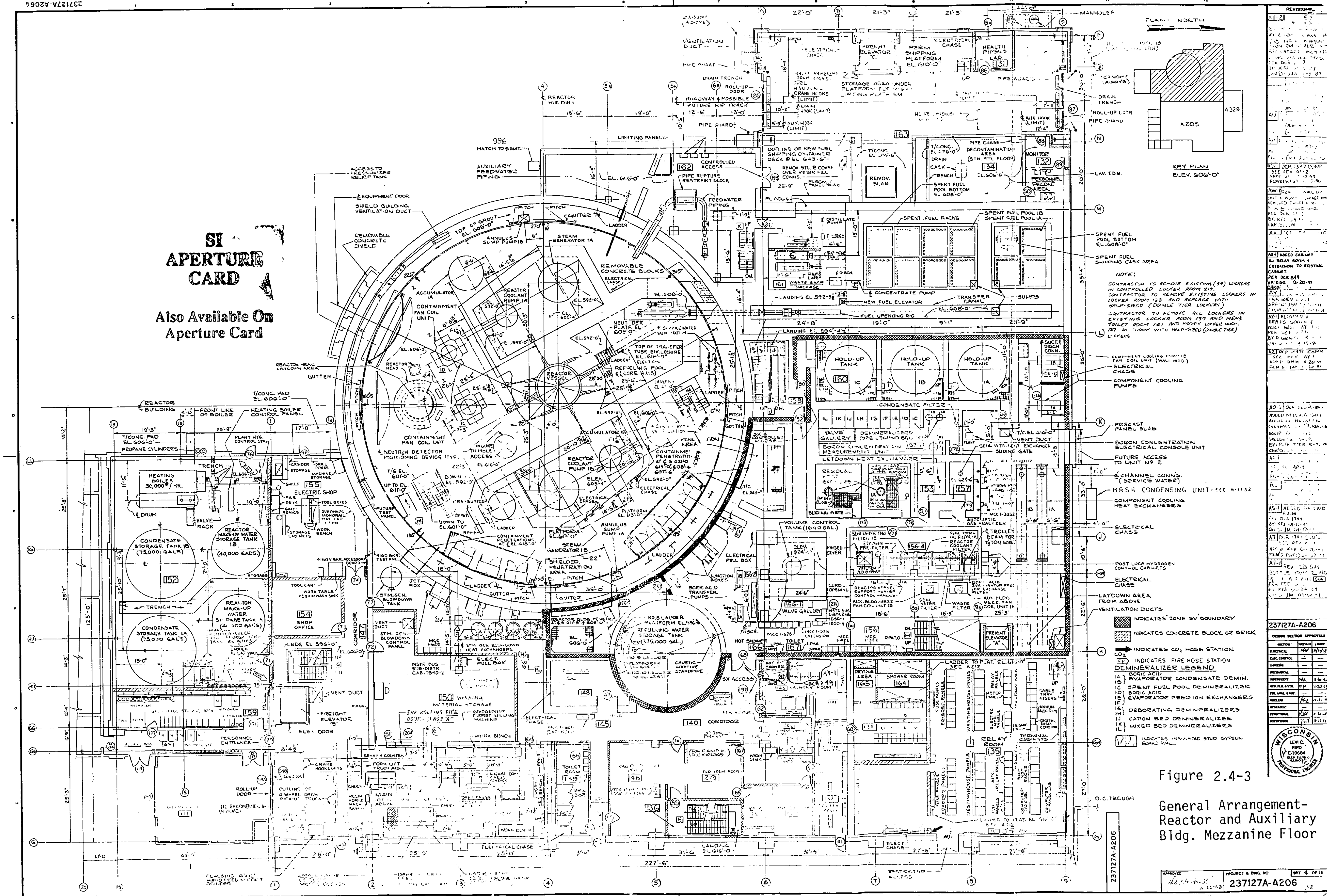


Figure 2.4-3

General Arrangement-
Reactor and Auxiliary
Bldg. Mezzanine Floor

9212090118-02

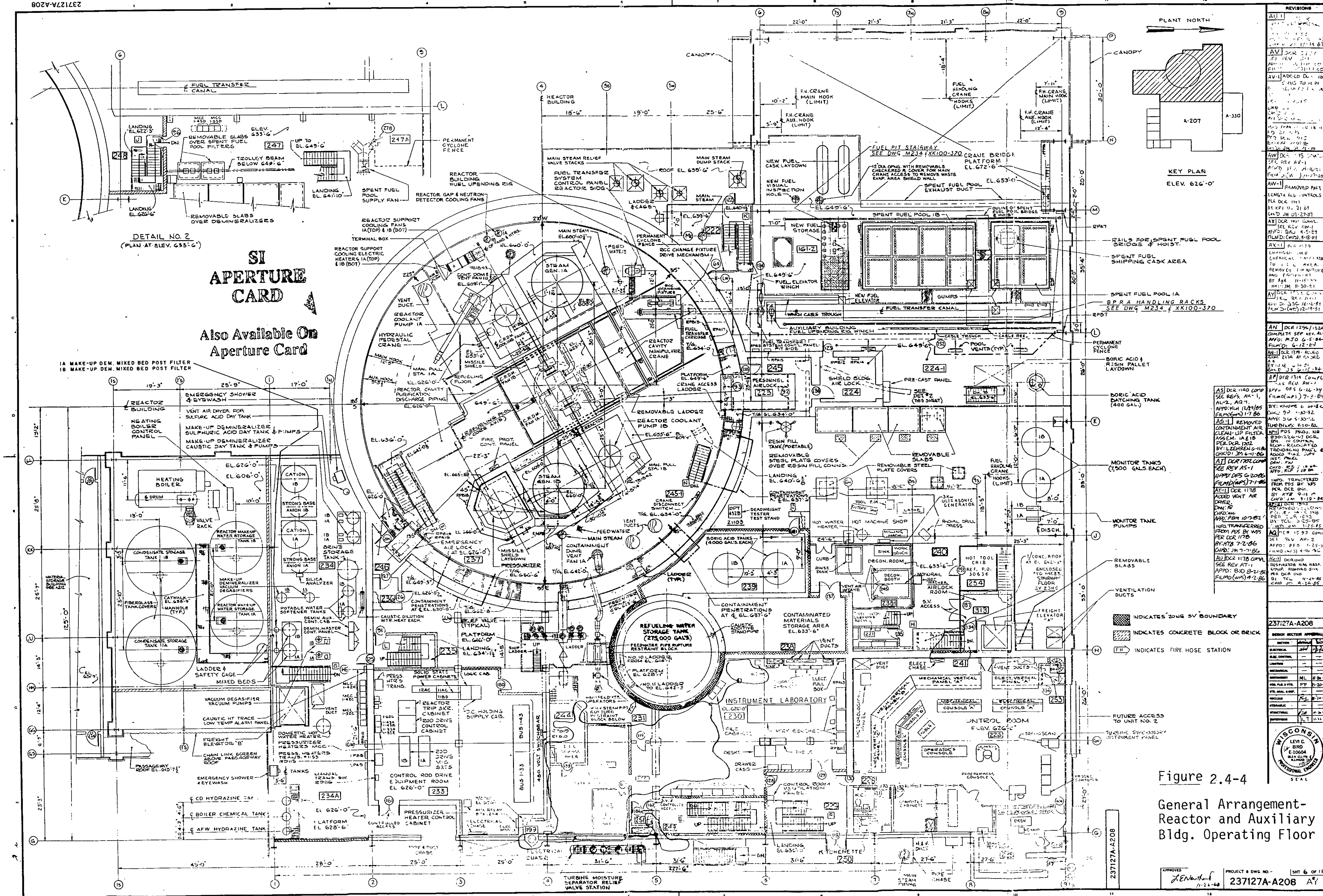


Figure 2.4-4

General Arrangement-
Reactor and Auxiliary
Bldg. Operating Floor

9212090118-03

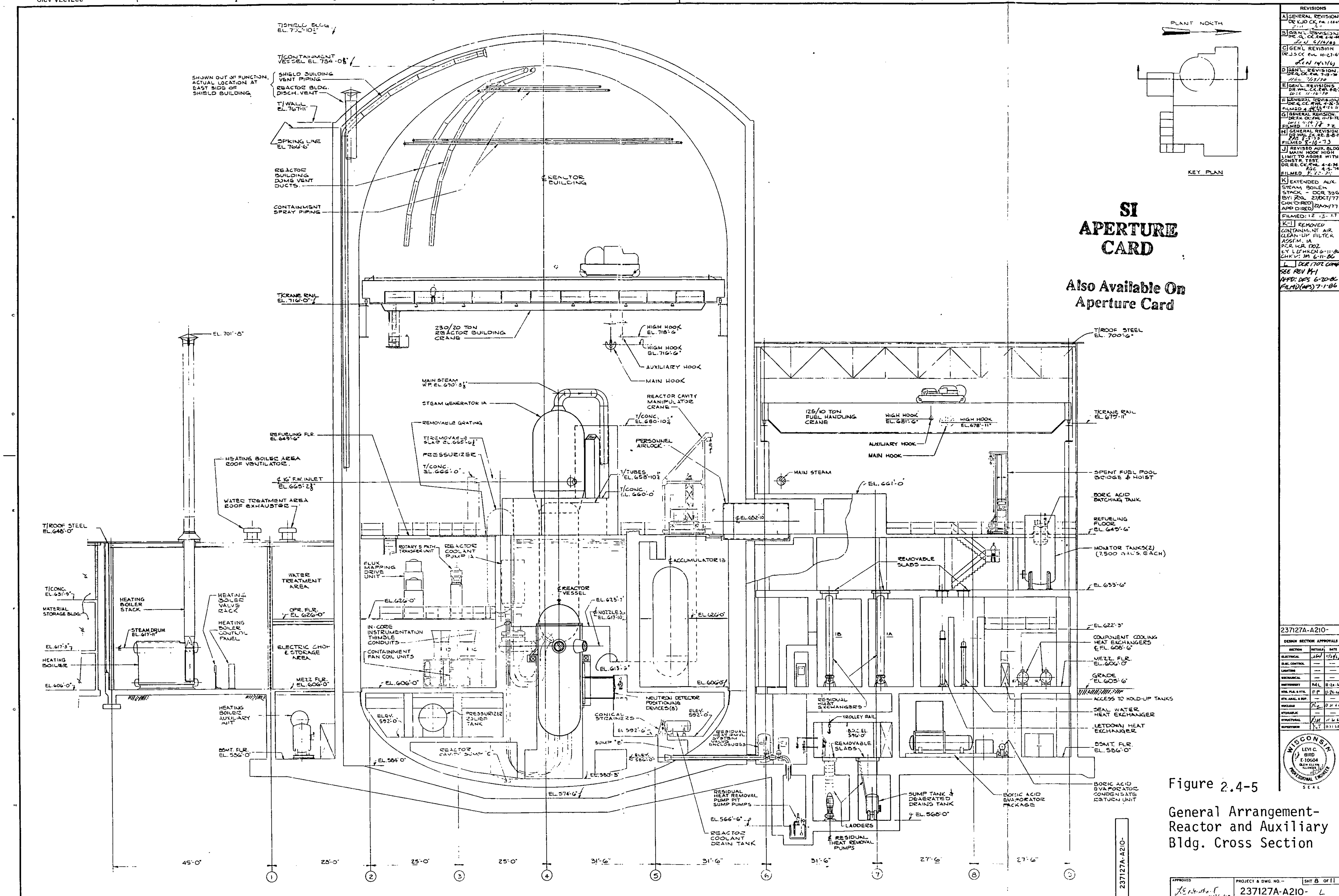


Figure 2.4-5

General Arrangement-Reactor and Auxiliary Bldg. Cross Section

9212090118-04

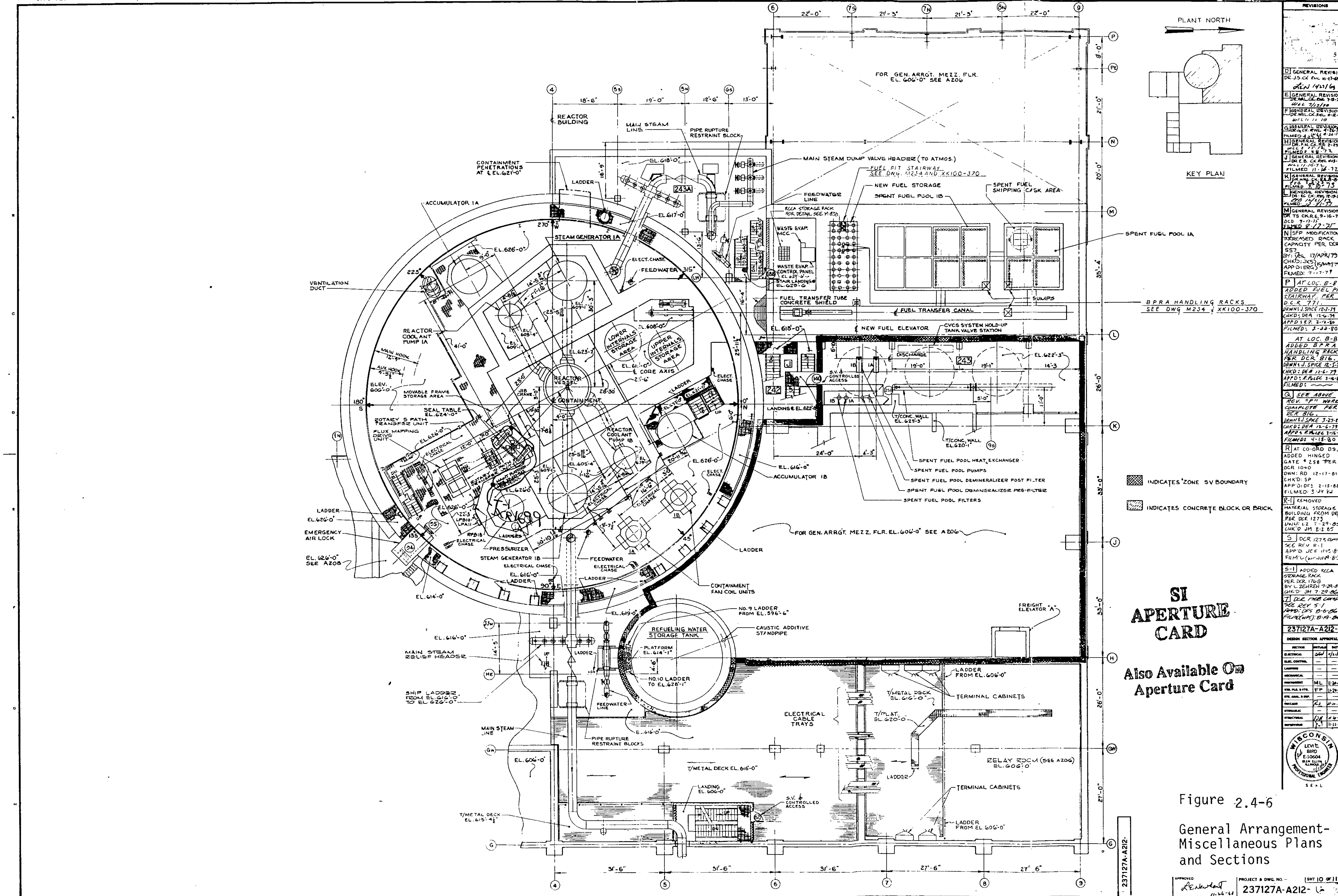


Figure 2.4-6

General Arrangement-
Miscellaneous Plans
and Sections

9212090118-05

3.0 FRONT-END ANALYSIS

This section documents the Level 1 portion of the Kewaunee PRA. The methodology used in the Kewaunee PRA is consistent with the PRA Procedures Guide and several other PRAs and IPEs that have been reviewed by the NRC staff. The organization of the Kewaunee PRA closely resembles the outline provided in NUREG-1335. There are, however, some slight differences. The organization of sections 3.1 and 3.3 is slightly different then that outlined in NUREG-1335. Namely, subsections 3.1.2 and 3.1.3 are combined into one Event Tree section, subsections 3.3.1 and 3.3.2 are combined into one Data Analysis section, and subsections 3.3.5 - 3.3.7 are combined into one Core Damage Quantification section.

3.1 Accident Sequence Delineation

3.1.1 Initiation Events

The purpose of the initiating event analysis was to identify a complete list of unique initiating events and then to determine the appropriate initiating event frequency for each event. Several steps were taken to identify and categorize the initiating events; these steps are described below:

1. A Core Damage Logic Diagram for Internal Initiators was developed to systematically categorize all internal initiating events on the basis of similar transient progression or consequences.

2. The initiating event categories were grouped into three categories, LOCAs, transients and special initiating events.

3. LOCAs include all accidents that result in a reduction of primary coolant system water inventory. The category was divided into three subcategories, leak to the secondary system (steam generator tube rupture), leak that bypasses containment (Event V-sequence/interfacing systems LOCA), and leaks within the containment building. The category of leaks within containment was further divided based on the size of the break; this discussion will take place later in this section.

4. In order to determine the specific events modeled for transients and special initiating events, two separate reviews were performed. The first was a review of the transient initiators provided in NUREG/CR-3862[17], and past PRAs. The second was a system by system review to determine which system failures can cause a plant trip.

5. The transient initiators were then grouped into categories based on plant response, signal actuation, systems required for mitigation, and subsequent plant-related effects.

   The initiating events and their calculated frequencies used in the Kewaunee PRA are provided in Table 3.1-1. A summary discussion of each of the 16 initiating events and how the frequencies were determined is provided below. The initiating event frequencies for the internal flooding events modeled are described in section 3.3.8 of this report.

3.1.1.1 Large Break LOCA

The large break LOCA category includes ruptures inside containment in the size range from a doubled ended severance of the largest pipe in the reactor coolant system (RCS) down to a 6 inch equivalent diameter hole. This range is typically used and has been generally accepted in PRA studies for Westinghouse 2 loop plants. The generic LOCA frequency for breaks greater than 6 inches provided in NUREG/CR-4550[9] was used which is 5.0E-4/year. There are no single failures of equipment (other than piping) that would result in a large LOCA, so the NUREG/CR-4550 value was used.

3.1.1.2 Medium Break LOCA

The medium break LOCA range of breaks represents all RCS ruptures inside containment with blowdown rates equivalent to breaks in the range from 6 inch to 2 inch equivalent diameter holes. A stuck open safety valve on the pressurizer also causes a medium LOCA. The medium LOCA category is a transition range between large and small break LOCAs, exhibiting characteristics of both large and small LOCAs depending on the size of break. Several existing PRA studies were reviewed and the methodology for determining the IE frequency for medium LOCAs varied. The method used in the Kewaunee PRA was to sum the random pipe failure frequency for medium LOCAs from WASH-1400[12] (8.0E-4/year) and the calculational frequency for a stuck safety valve at Kewaunee. Kewaunee has never had a stuck open safety valve, so it was conservatively postulated that there was one event. The pressurizer safety failure frequency was determined using plant data for the number of transients per year and generic data for failure to reseat. The result for the medium break LOCA IE frequency was 2.36E-3/year.

3.1.1.3 Small Break LOCA

This category of events comprises breaks inside containment in the range of 2 inch to 3/8 inch equivalent diameter, as well as failures of reactor coolant pump seals, and power operated relief valves (PORVs). Breaks under 3/8" diameter, which are sometimes called very small breaks, can be maintained by normal charging pump flow and do not result in a reactor trip; therefore, they were not modeled. The pipe break frequency for this range of piping, taken from WASH-1400, is 3.0E-3/year. Plant records were reviewed and 2 cases were found in which a pressurizer PORV lifted. In both cases the block valves were operable; however, two failures were assumed in the analysis. Kewaunee has no experience with reactor coolant pump seal failures, so generic Westinghouse experience was used. The result for small break LOCA IE frequency was 5.12E-3/year.

3.1.1.4 Steam Generator Tube Rupture

Although this category can be included in the small LOCA category, it is separated due to its unique effect on the plant. A steam generator tube rupture may result in direct bypass of the containment boundary, if steam generator safety or relief valves fail, and therefore must be analyzed separately. Kewaunee has not had a steam generator tube rupture event. Therefore, historical tube failure rate data for domestic Westinghouse steam generators was used in combination with the number of tubes in the Kewaunee steam generators. The result for the steam generator tube rupture IE frequency was 6.41E-3/year.

3.1.1.5 Reactor Vessel Failure

This event is for loss of coolant accidents beyond the capacity of the ECCS. A generic value from WASH-1400 was used, this value is 3.0E-7/year.

3.1.1.6 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 the containment boundary and a direct release path to the environment.

A comprehensive analysis was performed to determine all of the possible interfacing systems LOCA (ISL) paths that could bypass containment. Each of the ISL paths identified was further analyzed and the IE frequency for a LOCA through each path was quantified. The methodology was based on that described in NSAC/154[18] with data and other input from NUREG/CR-5102[19]. The net result for the interfacing systems LOCA IE frequency was 1.48E-6/year.

3.1.1.7 Transients With Main Feedwater Available

This is a large category of events that result in a reactor trip. The common tie to all the events grouped in this category is that the main feedwater system is available for decay heat removal at the time of the trip. In determining the IE frequency for this event, all of the plants trips at Kewaunee through the end of 1991 were reviewed to determine whether or not main feedwater was available. Using the plant data it was determined that the IE frequency for transients with main feedwater available is 3.0/year.

3.1.1.8 Transients Without Main Feedwater Available

This event is similar to the previous event except that main feedwater is not available. Once again plant data was used and the IE frequency for this event was determined to be 0.14/year.

3.1.1.9 Large Steam/Feedwater Line Break

This event includes main feedwater 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 secondary side isolation and ECCS actuation. The IE frequency for a large steam/feedwater line break is assumed to be similar to the frequency of a large LOCA event. However, since steam and feedwater piping has failed in industry, a factor of 5 was multiplied to the large LOCA frequency. The IE frequency for this event was therefore determined to be 2.5E-3/year.

3.1.1.10 Loss of Offsite Power

This event results from a complete loss of the offsite grid power accompanied by a turbine trip/reactor trip. Following the initial loss of AC power, at least one diesel generator starts and supplies electrical power to a safeguards bus. Events in which both diesel generators fail are included under the station blackout event. Kewaunee has never had a loss of offsite power event so a generic methodology developed in NUMARC 87-00[20] in response to the station blackout rule was used to determine the IE frequency. The NUMARC 87-00 methodology sums the frequencies of four causes of loss of offsite power events: plant centered losses, grid disturbances, severe weather related losses, and extremely severe weather related losses. NSAC-182[21] provides a generic value for the sum of the plant centered losses and grid disturbances. The severe and extremely severe weather related losses frequencies were determined using a methodology from NUMARC 87-00 and weather data specific to the Kewaunee site. The net result for the loss of offsite power IE frequency was 4.36E-2/year.

3.1.1.11 Station Blackout

Three possible initiators were considered for this event in the Kewaunee PRA. The first is the most obvious, a loss of offsite power along with a failure of the onsite emergency buses. The other two are similar, but the event is triggered by either a transient with main feedwater available or a transient without main feedwater available. A fault tree was developed encompassing all three initiators and the resultant IE frequency for the station blackout event was 4.35E-4/year.

3.1.1.12 Anticipated Transient Without Scram or Main Feedwater

This event does not involve an internal initiating event but the consequential failure resulting from another event. A separate category was provided because of the specific plant response and operator action it requires. Only an Anticipated Transient Without Scram (ATWS) without main feedwater is considered because if an ATWS occurs with main feedwater available, the likelihood of core damage is negligible. A fault tree was developed to quantify the failure probability of the reactor protection system. This value was added to the probability of mechanical failure of one or more rod control cluster assemblies. The sum of the two failure probabilities was multiplied by the IE frequency for transients without main feedwater available to determine the IE frequency which was 3.84E-6/year.

3.1.1.13 Loss of Service Water System

This event results from a total loss of service water. This loss of service water results in a loss of cooling to safety-related and secondary plant systems. The safety-related components that lose cooling are the component cooling heat exchangers, safety injection pumps, containment fan coolers, and diesel generators. The important secondary systems that will lose cooling are the feedwater pumps and turbine oil coolers. Reactor trip will occur either because of a loss of the feedwater pumps or a manual trip because of loss of component cooling to the reactor coolant pumps. The service water system is the safety related cooling source to the station and instrument air compressors, but the compressors are normally cooled by the potable water system, which is not a safety system. The loss of service water will also result in the loss of room cooling of many areas in the plant.

The loss of service water IE frequency was developed through quantification of a service water system fault tree with sub-tree linking. The resultant IE frequency was 1.22E-4/year.

3.1.1.14 Loss of Component Cooling Water System

The loss of component cooling results in the loss of cooling to the reactor coolant pumps, the RHR heat exchangers, the RHR pumps, the safety injection pumps, and the containment spray pumps. The loss of component cooling most likely results in a manual reactor trip due to loss of cooling to the reactor coolant pumps.

The loss of component cooling water IE frequency was developed through quantification of a component cooling water fault tree with sub-tree linking. The resultant IE frequency was 1.61E-3/year.

3.1.1.15 Loss of 125V Emergency DC Bus

The loss of either the train A or B safeguards DC bus may cause a reactor trip and a loss of various components. The result of a loss of each bus was analyzed and the major difference is that the loss of the train A safeguards DC bus results in the loss of the automatic start function of both the A motor driven auxiliary feedwater (AFW) pump and the turbine driven AFW pump, whereas the B safeguards DC bus only fails the automatic start of the B motor driven AFW pump. Therefore, the loss of the A safeguards DC bus is modeled to bound the loss of either DC bus. The IE frequency for this event was developed through fault tree quantification with sub-tree linking. The resultant IE frequency was 2.35E-3/year.

3.1.1.16 Loss of Instrument Air

The complete loss of instrument air causes the main feedwater regulating valves to close resulting in a reactor trip. The loss of instrument air also causes air operated valves to fail to their safe position and the loss of the speed controllers for 2 of the 3 charging pumps. The IE frequency for the loss of instrument air was determined by the quantification of an instrument air fault tree with sub-tree linking. Instrument air piping failures were also included because the instrument air system history has shown it to not be a leak-tight system. The resultant IE frequency was 1.07E-4/year.

TABLE 3.1-1

QUANTIFICATION OF INITIATING EVENT CATEGORIES

Initiating Event Categories	Frequencies (Per Year)
1.0 Large LOCA (>6")	5.0E-04
2.0 Medium LOCA (2"-6")	2.36E-03
3.0 Small LOCA (<2")	5.12E-03
4.0 Steam Generator Tube Rupture	6.41E-03
5.0 Reactor Vessel Failure	3.0E-07
6.0 Interfacing Systems LOCA	1.48E-06
7.0 Transients With Main Feedwater	3.0
8.0 Transients Without Main Feedwater	1.4E-01
9.0 Large Steam/Feedwater Line Break	2.5E-03
10.0 Loss of Offsite Power	4.36E-02
11.0 Station Blackout	4.35E-04
12.0 Anticipated Transient Without Scram Or Main Feedwater	3.84E-06
13.0 Loss of Service Water System	1.22E-04
14.0 Loss of Component Cooling Water System	1.61E-03
15.0 Loss of 125V Emergency Bus	2.35E-03
16.0 Loss of Instrument Air	1.07E-04

3.1.2/3 Event Trees

Event trees were developed for each of the initiating events described in Section 3.1.1 of this report. The event trees were developed in accordance with the functional event tree logic summarized in section 3.1.2/3.1. Sections 3.1.2/3.2 through 3.1.2/3.17 provide the event trees, top level event summaries and success criteria for the 16 events. Section 3.1.2/3.18 provides additional information regarding success criteria bases that is not described in the top level event summaries in the previous 16 sections. Section 3.1.2/3.19 provides the reasoning and calculations that support the scalars used in the event trees.

3.1.2/3.1 Event Tree Guidelines

Event trees used in this study are based on a functional event tree logic. Modeled after the initiating event are, in progression: short term cooling, operator actions and long term recirculation cooling. In general, when short term cooling and long term cooling are available, no core damage results. If short term cooling fails, some operator actions may be performed to accomplish the function of short term cooling. Therefore, if these operator actions are successful, early core damage does not occur and long term recirculation precludes later core damage. Conversely, if both short term cooling and operator actions are not available, then early core melt is assumed to follow. The support systems needed for the success of frontline systems are included in the success criterion of each event tree node.

The termination of criticality by control rod insertion (reactor trip) is not shown in the functional event tree. The failure to trip is treated separately as the anticipated transient without scram or main feedwater (AWS) initiating event. Random consequential LOCA events (such as opening of PORVs or reactor coolant pump (RXCP) seal LOCA after a transient) are included in the small LOCA initiating event; inadvertent opening of safety valves in the steam side is included in the steam line break event; and the stuck open safety valve after passing water is included in the steam generator tube rupture (SGR) event. Operator generated LOCAs, such as bleed and feed operation after loss of secondary cooling in a transient event, are modeled explicitly in the event trees.

Short term cooling is normally carried out by either secondary cooling or the emergency core cooling systems (ECCS) when primary integrity is lost. Part of the definition of secondary cooling is the need to relieve steam from the secondary side.

The methods available to relieve steam are dependent upon whether the main steam isolation valves (MSIVs) are open or closed. The MSIVs close on the following main steam isolation signals:

1. Containment pressure Hi Hi.
2. Hi Hi steam flow with safety injection signal.
3. Hi steam flow with safety injection signal with Lo Lo Tave
4. Manual

Main steam isolation occurs for those events that result in high steam flow or high containment pressure - steam line breaks or large LOCAs.

With the MSIVs closed, the following methods of steam relief are available:

1. Atmospheric Relief Valve (PORVs SD-3A/B) - Both PORVs combined have a total relief capacity of 10% of maximum total steam flow.

2. ASME Code Safety Valves - Five safety valves SD-1A(B)1 through SD-1A(B)5 are on each main steam header. The total relief capacity of all 10 safety valves is 110% of rated steam flow.

3. Following the initiating event, procedures will guide the operator to eventually reset SI and control core average temperature using the steam dump valves.

It should be noted that for those IEs that cause main steam isolation (steam line break or large LOCA), short term secondary cooling by relieving steam is not necessary due to the large cooldown of the reactor coolant system (RCS) and resulting low SG pressures.

With the MSIVs open, the following methods of steam relief are available:

1. Atmospheric Relief Valves (PORVs SD-3A/B) - as previously explained.

2. Atmospheric Steam Dump Valves (SD-5A(B)1 through SD-5A(B)3) - Three 8" control valves are located downstream of the MSIVs on each mainsteam line. The six atmospheric steam dump valves are sized to give a capacity of 45% of maximum steam flow.

3. Condenser Steam Dump Valves (SD-11A(B)1 through S-11A(B)3) - Three 8" control valves located downstream of the MSIVs on each main steam line provide steam relief to the condenser. The six condenser steam dump valves are sized to give a capacity of 40% of maximum steam flow.

4. ASME Code Safety Valves (SD-1A(B)1 through SD-1A(B)5) - As previously explained.

For the secondary cooling nodes operator actions to verify and regulate flow to the steam generators are implicit in the definitions. Failure to perform these actions will fail the node. Therefore, these operator actions are modeled in the fault tree analyses for these top events.

Following the short term cooling node is the node for operator actions. The types of actions are different for different initiating events, and are described in detail in the event tree modeling for each initiating event.

The last node is long term cooling (LC) which refers to the continued removal of reactor decay heat for 24 hours following the initiating event. Long term cooling is typically supplied by auxiliary feedwater (or main feedwater if available), by the residual heat removal (RHR) System, or by the ECCS through containment sump recirculation. The LC node encompasses all operator actions and equipment availability needed to perform this function.

In each event tree, certain criteria were established that must be satisfied in order to prevent core melt. These criteria deal with system functions to maintain primary inventory control and remove reactor decay heat. Thus, following all initiating events that do not include a breach of the primary coolant boundary, feedwater is required. If a loss of primary coolant occurs from either an initiating event (LOCA, SGR) or from a consequential failure following an event, then safety injection is required.

In the SGR, there are paths that are termed "Leak". These are neither successes nor do they lead to core melt. The term "Leak" is applied when the SG tube rupture is within the capacity of the charging pumps.

The event tree analysis identifies the results of events effecting reactor and turbine-generator availability and subsequent failures of safeguards systems. To preserve core integrity, certain functions must be achieved following an event: shutdown of the reactor and removal of reactor decay heat. Multiple systems and methods are available to carry out these functions, and these systems are explicitly analyzed by the event tree.

However, when different systems fail, it is possible that core damage will occur. For this analysis, once conditions that might yield core damage have been identified by an event tree sequence, core melt is postulated. Neither recovery of systems nor use of emergency non-essential safeguards methods that hypothetically could be attempted by the operators are addressed.

Each event sequence that results in core melt is identified by an appropriate core melt state descriptor. These descriptors are acronyms that indicate the type of event and the time period between initiation of the accident and the onset of core melt. Two sets of characters identify the core melt state. The first identifier is one of the following:

A	=	Large LOCA	-	Characterized by rapid decompression of the RCS and core uncovery.
S	=	Small LOCA	-	Characterized by a small breach of the RCS pressure boundary resulting in a direct release path to the containment environment. Depending on the break size, the RCS decompression can slow down or even stall at relatively high pressure levels.

T	=	Transient	-	Characterized by release of primary coolant through the pressurizer relief and safety valves to the pressurizer relief tank. This release path mitigates the containment pressure response and provides efficient scrubbing of radionuclides from the primary coolant. The RCS pressure may or may not drop, depending on development of the accident sequence.
V	=	Interfacing Systems LOCA Event	-	This is a large LOCA event characterized by a leak path outside of containment. Rapid and large release of radionuclides is expected because the release path bypasses the containment building.
V2	=	Steam Generator Tube Rupture Event	-	This particular SGR event is characterized by stuck open secondary side relief/safety valves(s). Primary coolant is discharged through these valves, bypassing containment. The release is expected to occur in small amounts due to the size of the RCS leak. A leak larger then 1 SG tube is too improbable to be considered.

The second identifier is one of the following:

E	=	Early Core Melt	-	The onset of core melt occurs within the four hour period following the initiating event.
L	=	Late Core Melt	-	The onset of core melt occurs after the four hour period following the initiating event.

A third identifier, always the letter "Y", is added to the core melt state descriptor simply to indicate that containment safeguards operation must later be considered when converting these core melt state descriptors to Plant Damage States. As an example, all core melt descriptors will be prefixed with the acronym "CMS-" followed by the sequence identification scheme outlined above. A Large LOCA having early core melt would be identified as CMS-AEY where the "Y" indicates that the containment safeguards are not yet linked.

Initiating events are designated by IEV-XXX. Success events are indicated by SUC-XXX while failure events are indicated by SYS-XXX.

3.1.2/3.2 Large Break LOCA

INITIATING EVENT - LARGE LOCA (LLO)

A large LOCA is initiated by random ruptures in the RCS from about a 6-inch diameter rupture ranging up to the area of a double-ended rupture of the largest primary system pipe.

ACCUMULATOR INJECTION (ACC)

The SI accumulators comprise a passive system designed to rapidly inject a large volume of water into the RCS cold legs during a large LOCA event. The accumulators are tanks that are partially filled with borated water and pressurized by compressed nitrogen. One accumulator is attached to each loop of the RCS. If the RCS pressure drops below the pressure in the accumulators, borated water is forced through check valves into the cold legs to provide core cooling. Success of the system is the injection from one accumulator into the intact cold leg pipe. Failure of accumulator injection (ACC) results in early core melt.

LOW PRESSURE INJECTION (LI1)

If accumulator injection (ACC) is successful, low pressure safety injection (SI) is addressed. Low pressure SI for the Large LOCA event (LI1) is performed by the RHR pumps. The purpose of this function is to supplement the SI accumulators in refilling the reactor vessel lower plenum and/or reflooding the reactor core. Following the emptying of the accumulators, the RHR pumps continue supplying water to the reactor vessel for core cooling. The RHR pumps automatically start following generation of an SI signal. During power operation, the RHR trains are aligned for low head SI and, upon starting, they take suction from the refueling water storage tank (RWST) and deliver borated water to the reactor vessel. LI1 continues until the water in the RWST is depleted. Generation of the SI signal and availability of the RWST are both modeled directly in the low head SI System fault tree.

Success of LI1 is 1 of 2 low pressure SI trains delivering water to the intact loop.

Because the large LOCA event can result in rapid core uncovery and heat up, it is conservatively assumed that there is insufficient time for the operators to perform any recovery actions to compensate for failure of LI1. Therefore, if LI1 fails, early core melt is assumed.

LOW PRESSURE RECIRCULATION (LR1)

If ACC and LI1 are successful, long term cooling is addressed. During the SI phase of the accident water spilled from the break and water from the containment spray system are collected on the containment floor and in the sump. Long term cooling for the large LOCA event (LR1) is provided by the RHR pumps taking suction from the containment sump and discharging through the residual heat exchangers and back into the reactor coolant system. LR1 starts when the RWST water level decreases to 37%. Success of LR1 is 1 of 2 low pressure recirculation trains delivering flow to the reactor vessel.

Failure of LR1 results in early core melt.

FIGURE 3.1.2/3-1

LARGE LOCA EVENT TREE

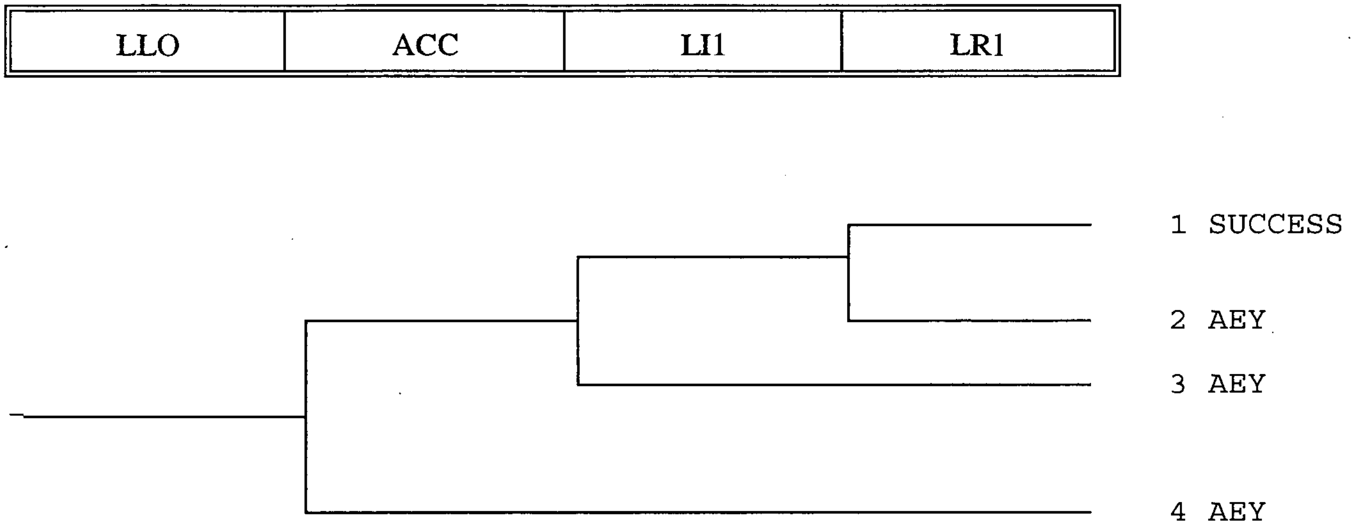


TABLE 3.1.2/3-1

SUCCESS CRITERIA FOR LARGE LOCA

Top Event Description	System Success Criteria	Necessary Operator Actions	Mission Time (hrs)
ACC - ACCUMULATOR INJECTION	1 of 1 accumulator on the intact loop injecting into the cold leg	Confirm operation of system	None
LI1 - LOW PRESSURE SAFETY INJECTION	1 of 2 low pressure SI trains injecting flow into reactor vessel	Confirm operation of system	1.0
LR1 - LOW PRESSURE RECIRCULATION	1 of 2 low pressure SI trains in recirc from containment sump to reactor vessel via residual heat exchangers, sump valve on operable recirculation train open	Manually align low pressure sump recirc on low RWST level, align CCW cooling to residual Hx, confirm operation of system	23.0

3.1.2/3.3 Medium Break LOCA

INITIATING EVENT - MEDIUM LOCA (MLO)

This initiating event comprises those losses of reactor coolant for medium breaks ranging in size from 2.0 inches to 6.0 inches equivalent diameter.

ACCUMULATOR INJECTION (ACC)

The SI accumulators comprise a passive system designed to rapidly inject a large volume of water into the RCS cold legs during a LOCA transient. The accumulators are tanks that are partially filled with borated water and pressurized by compressed nitrogen. One accumulator is attached to each loop of the RCS. If the RCS pressure drops below the pressure in the accumulators, borated water is forced through check valves into the cold legs to provide core cooling. Success of the system is the injection of one accumulator into the intact cold leg pipe. It is assumed that the accumulator attached to the cold leg with the break is unavailable because the accumulator contents are assumed to flow out the cold leg break.

Depending on the sequence of the event, the accumulators inject either because of the LOCA depressurization or because of operator action to depressurize the RCS. The accumulators provide necessary water to cover the core as well as provide additional water for recirculation.

It is assumed that failure of this node and high pressure SI (HI0) results in core melt.

HIGH PRESSURE INJECTION (HI0)

High pressure injection is automatically actuated on an SI signal upon receiving a low pressurizer pressure signal. The SI pumps first take suction from the boric acid tank, then from the RWST. The SI pumps inject into the RCS cold legs.

Success of HI0 is 1 of 2 SI pumps delivering flow to the intact loop cold leg. An SI signal must be successfully generated.

HIGH PRESSURE RECIRCULATION (HR0)

If high pressure injection (HI0) is successful, long term cooling is addressed. Long term cooling is provided by sump recirculation. For LOCAs in which RCS pressure remains above 140 psig, a low pressure SI train is lined up to take suction from the containment sump and discharge to the suction of the SI pumps via the residual heat exchangers. This lineup is referred to as a SI/RHR train in the emergency operating procedures.

Success of HR0 is 1 of 2 SI/RHR trains providing flow to the intact RCS cold leg.

It is assumed that failure of this node and low pressure recirculation (LR2) results in late core melt.

OPERATOR ACTION - COOLDOWN AND DEPRESSURIZE RCS (OP1)

Upon failure of the high pressure SI, RCS inventory can be provided by the low pressure SI system. This requires operator action to cool down and depressurize the RCS by dumping steam from an intact steam generator to maintain a maximum 100 °F/hour cooldown rate. These operator actions are provided in Emergency Operating Procedure ES-1.2, Post LOCA Cooldown and Depressurization, which is entered from E-1, Loss of Reactor or Secondary Coolant. It is assumed that the operators have 15 minutes to initiate this action and that failure of this node results in early core melt because of the unavailability of all ECCS to provide core cooling.

AUXILIARY FEEDWATER (AF0)

If operator actions to cool down and depressurize the RCS are required, auxiliary feedwater (AFW) must be available in order to supply inventory to a steam generator (SG) for the cooldown.

One SG and the associated atmospheric relief valve has adequate capacity to cool down and depressurize the RCS and ensure accumulator injection and subsequent RHR pump injection. Success of AF0 is 1 of 3 AFW pumps injecting to 1 of 2 SGs. It is conservative to assume the minimum feedwater flow (200 gpm) for success.

OPERATOR ACTION - ESTABLISH MAIN FEEDWATER (OM0)

If operator actions to cool down and depressurize the RCS are required and AF0 fails, main feedwater (OM0) is used for secondary cooling. Emergency Operating Procedure FR-H.1, Response to Loss of Secondary Heat Sink, is entered via the critical safety function trees. FR-H.1 instructs the operators to attempt to restore AFW and to line up main feedwater to provide secondary cooling.

If main feedwater is unavailable, the operators are instructed to depressurize the RCS in order to block SI and depressurize a SG in order to provide secondary coolant flow from the condensate system. However, due to the complexity of the operator actions required to establish flow to a SG from the condensate system alone, it would take a significant amount of time to establish this flow and it is not included in the event tree modeling.

Success of OM0 is 1 of 2 main feedwater (MFW) trains delivering flow to at least 1 of 2 SGs with a flow rate of at least 200 gpm.

It is assumed that failure of this node results in early core melt. This because of the inability to provide sufficient inventory to a steam generator to ensure RCS cooldown prior to using low pressure SI for core cooling.

LOW PRESSURE INJECTION (LI2)

Low pressure injection is provided by the RHR pumps which inject borated water from the RWST to the reactor vessel. This system aids in the filling of the reactor vessel and supplying water to complete the core reflooding process. The operators manually re-initiate low pressure injection following RCS cooldown and depressurization.

Success of LI2 is 1 of 2 low pressure SI trains delivering flow to the reactor vessel.

It is assumed that failure of this node will result in early core melt because of the loss of both high pressure and low pressure SI.

LOW PRESSURE RECIRCULATION (LR1)

If LI2 is required to mitigate this event, long term cooling with low pressure recirculation (LR1) is required. During the SI phase of the accident, water spilled from the break and water from the containment spray system is collected on the containment floor and in the sump. Long term cooling is provided by the RHR pumps taking suction from the containment sump and discharging to the reactor vessel via the residual heat exchangers.

Success of LR1 is 1 of 2 low pressure recirculation trains delivering flow to the reactor vessel.

It is assumed that failure of this node results in late core melt.

LOW PRESSURE RECIRCULATION (LR2)

If HI0 is successful, the long term cooling is with HR0 or low pressure recirculation (LR2). For LOCAs in which RCS pressure quickly drops below 140 psig, a low pressure SI train is lined up to the containment sump and discharges to the reactor vessel via the residual heat exchangers.

Success of LR2 is 1 of 2 low pressure recirculation trains delivering flow to the reactor vessel.

It is assumed that failure of this node and HR0 results in late core melt.

FIGURE 3.1.2/3-2

MEDIUM LOCA EVENT TREE

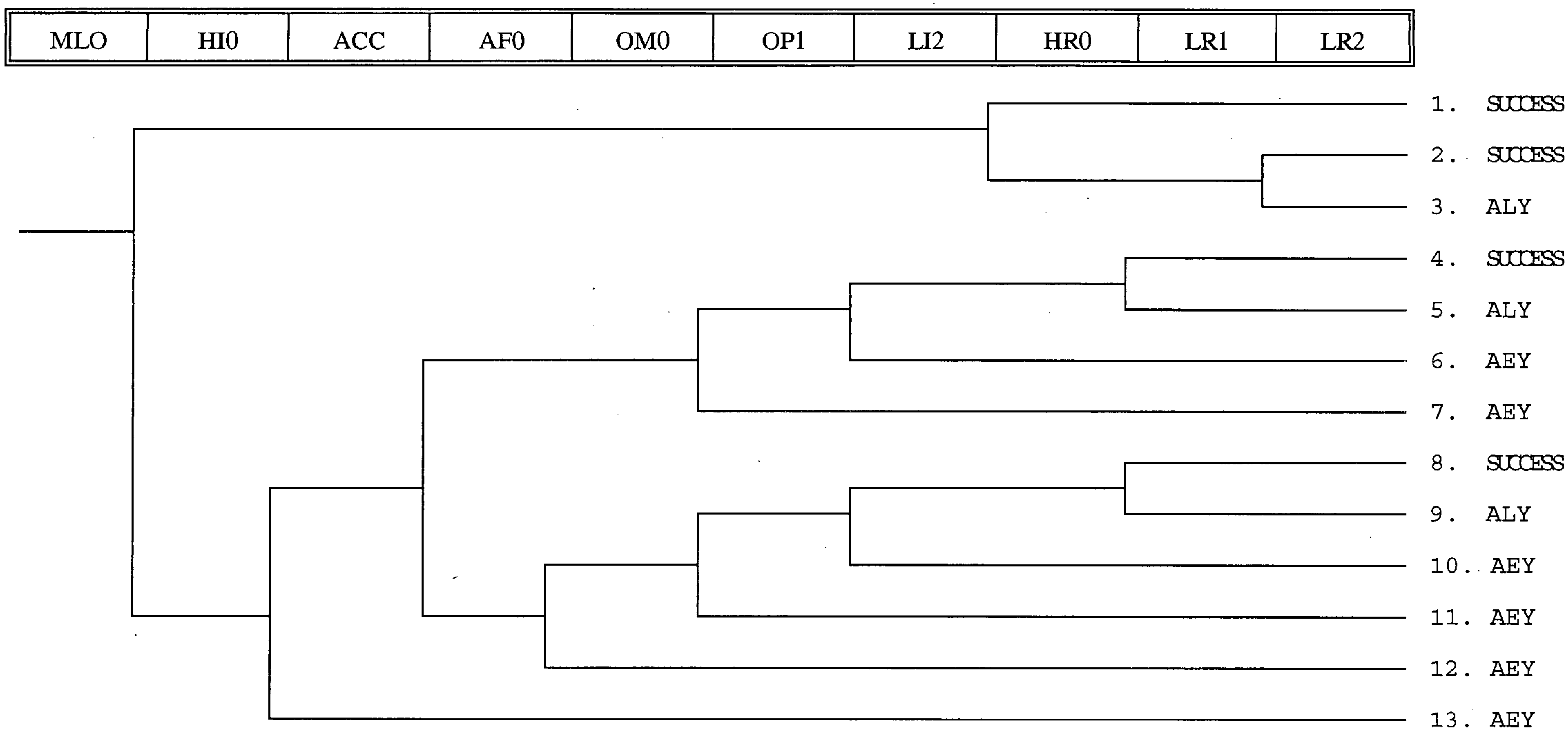


TABLE 3.1.2/3-2

SUCCESS CRITERIA FOR MEDIUM LOCA

Top Event Description	System Success Criteria	Necessary Operator Actions	Mission Time (hrs)
HI0 - HIGH PRESSURE INJECTION	1 of 2 high pressure SI trains injecting contents of BAT and RWST to intact RCS cold leg	Confirm operation of system	3.5
HR0 - HIGH PRESSURE RECIRCULATION	1 of 2 SI/RHR trains delivering flow from containment sump to intact RCS cold leg, sump valves on operable recirc train open	Manually align high pressure containment sump recirculation on low RWST level, align CCW cooling to residual Hx, confirm operation of system. (Note: This action must be successful for LR2 success.)	20.5
ACC - ACCUMULATOR INJECTION	1 of 1 accumulator on the intact loop injecting into the cold leg	Confirm operation of system	None
OP1 - OPERATOR ACTION-COOLDOWN AND DEPRESSURIZE RCS	Operator initiated cooldown started within 15 minutes using at least one SG supplied with feedwater	With EOP ES-1.2, cooldown RCS by dumping steam at max 100°F/hr. Depressurize RCS to inject accumulators and permit initiation of low pressure SI.	Approximately 1 hour (until break flow and low-head SI flow are able to remove decay heat).
AF0 - AUXILIARY FEEDWATER	1 of 3 AFW pumps delivering to at least 1 of 2 steam generators, total flow of at least 200 gpm	Confirm operation of system	Run for 24 hours

TABLE 3.1.2/3-2

SUCCESS CRITERIA FOR MEDIUM LOCA (Continued)

Top Event Description		System Success Criteria	Necessary Operator Actions	Mission Time (hrs)
OM0 -	OPERATOR ACTION-ESTABLISH MAIN FEEDWATER	1 of 2 MFW trains delivering at least 200 GPM to 1 of 2 steam generator.	Manually align and initiate MFW. Confirm operation of system.	Run for 24 hours.
LI2 -	LOW PRESSURE INJECTION	1 of 2 low pressure SI trains injecting flow into reactor vessel.	Manually initiate low pressure SI following RCS cooldown and depressurization, confirm operation of system.	1
LR1 -	LOW PRESSURE RECIRCULATION	1 of 2 low pressure SI trains in recirc from containment sump to reactor vessel via residual heat exchangers, sump valves on operable recirc train open.	Manually align low pressure containment sump recirc on low RWST level, align CCW cooling to residual Hx, confirm operation of system.	23
LR2 -	LOW PRESSURE RECIRCULATION	1 of 2 low pressure SI trains in recirc from containment sump to reactor vessel via residual heat exchangers, sump valves on operable recirc train open.	See HR0	23

3.1.2/3.4 Small Break LOCA

INITIATING EVENT - SMALL LOCA (SLO)

This initiating event comprises those LOCAs ranging in size from 0.375 inch to 2.0 inches equivalent diameter. These breaks are outside the capacity of the chemical and volume control system.

HIGH PRESSURE INJECTION (HI2)

High pressure SI is automatically actuated on an SI signal upon receiving a low pressurizer pressure signal. However, if the SI pumps are not automatically actuated, the operator manually starts the pumps. The SI pumps first take suction from the boric acid tank, then from the RWST. The SI pumps inject into the RCS cold legs.

Success of HI2 is 1 of 2 SI pumps delivering flow to 1 of 2 RCS cold legs.

OPERATOR ACTION - COOL DOWN AND DEPRESSURIZE RCS FOR CHARGING FLOW (ES1)

In EOP ES-1.2, Post LOCA Cooldown and Depressurization, the operator initiates a cooldown at a maximum rate of 100°F/hr using the available intact SGs. Once RHR system entry conditions are established (RCS pressure less than 425 psig, coldest RCS wide range temperature less than 380°F), the RHR system is placed in service to continue the cooldown to cold shutdown (200°F). As the cooldown progresses, the high pressure SI pumps are sequentially stopped (based on specified subcooling and RCS inventory criteria) with the charging pumps able to supply the required makeup. The RCS is also depressurized (using pressurizer spray or a PORV) to increase inventory and to minimize the break flow. Using EOP ES-1.2, it may be possible for very small break cases to depressurize the RCS to near atmospheric pressure and thereby terminate or substantially reduce the break flow. By doing so, the charging flow can be reduced and switchover to high pressure recirculation can be avoided.

Although the ES-1.2 actions appear complex, the operator has a long period of time to perform these actions. With the ES-1.2 actions, it is likely that for break sizes 0.7 inch diameter or smaller (it is assumed that 50% of the breaks will be in this range), the operator is able to avoid switchover for at least the 24 hour time frame assumed for the event tree and fault tree modeling. Success also requires steam dump from at least one of the steam generators (i.e., steam dump to condenser, if available, or operation of the atmospheric steam dump valve). The active SG(s) used for the cooldown also needs a supply of auxiliary feedwater (AF0 success) until the RHR system can be aligned for service. Another function to ensure ES1 success is a means for RCS depressurization by either pressurizer spray or operation of one PORV. Normal spray requires operation of an RXCP. In order to achieve cold shutdown conditions, it is assumed that operation of at least one train of RHR is required. A requirement for operation of at least 2 of 3 charging pumps is also included since this allows the SI pumps to be stopped at reasonable subcooling values and allows makeup control after the SI pumps are secured. Charging pumps are also required for auxiliary spray success.

HIGH PRESSURE RECIRCULATION (HR1)

If HI2 is successful, but ES1 fails, long term cooling is addressed via sump recirculation. For LOCAs in which RCS pressure remains above 140 psig, a low pressure SI train is lined up to take suction from the containment sump and discharge to the suction of the high pressure SI pumps via the residual heat exchangers. This lineup is referred to as a SI/RHR train in the emergency operating procedures.

Success of HR1 is 1 of 2 SI/RHR trains providing flow to 1 of 2 RCS cold legs.

It is assumed that failure of this node and low pressure recirculation (LR2), results in late core melt.

OPERATOR ACTION - COOLDOWN AND DEPRESSURIZE RCS FOR ACC AND LI2 (OP2)

Upon failure of the HI2, RCS inventory can be provided by the accumulators and the low pressure safety injection system. This requires operator action to cool down and depressurize the RCS by dumping steam from an intact steam generator to maintain a maximum 100 °F/hour cooldown rate. These operator actions are provided in Emergency Operating Procedure ES-1.2, Post LOCA Cooldown and Depressurization, which is entered from E-1, Loss of Reactor or Secondary Coolant. It is assumed that the operators have 30 minutes to initiate this action, and that failure of this node results in early core melt because of the inability to provide inventory to the core by high pressure SI, the accumulators or low pressure SI.

ACCUMULATOR INJECTION (ACC)

The accumulators are normally isolated in EOP ES-1.2 after the operator regains control of inventory (pressurizer level) and RCS subcooling. If high pressure safety injection fails, the conditions allowing accumulator isolation are not met, so the accumulators are available to inject their contents if the operator performs the cooldown and depressurization actions prescribed in the emergency procedures.

With high pressure safety injection failed, accumulator injection provides more time for the operators to cool down and depressurize the RCS to the point at which low pressure SI is provided to the core.

It is assumed that failure of this node results in early core melt because of the loss of both high pressure SI and accumulator injection.

AUXILIARY FEEDWATER (AF0)

Auxiliary feedwater (AF0) is required to remove decay heat for a small LOCA. AFW would be actuated on lo-lo steam generator level or by the SI signal.

It is assumed that secondary cooling is required for the entire event. Success of AF0 is 1 of 3 AFW pumps supplying at least 200 gpm to 1 of 2 steam generators for the entire event.

OPERATOR ACTION - ESTABLISH MAIN FEEDWATER (OM0)

If AF0 fails, main feedwater (OM0) is used for secondary cooling. Emergency Operating Procedure FR-H.1, Response to Loss of Secondary Heat Sink, is entered via the critical safety function trees. FR-H.1 instructs the operators to attempt to restore AFW and to line up MFW to provide secondary cooling. If MFW is unavailable, the operators are instructed to depressurize the primary system in order to block SI and to depressurize a steam generator in order to provide secondary coolant flow from the condensate system. However, due to the complexity of the operator actions required to establish flow to a steam generator from the condensate system alone, it would take a significant amount of time to establish this flow and it is not included in the event tree modeling.

Success of OM0 is 1 of 2 MFW trains delivering flow to 1 of 2 SGs for the entire event with a flow rate of at least 200 gpm.

It is assumed that if HI2 and OM0 fail, early core melt occurs. This is due to the unavailability of high pressure SI for bleed and feed.

OPERATOR ACTION - BLEED AND FEED (OB1)

If secondary cooling via AF0 or OM0 is unavailable, the operators are instructed to initiate primary system bleed and feed. Emergency operating procedure FR-H.1, Response to Loss of Secondary Cooling, instructs the operators to initiate bleed and feed if secondary cooling is lost and wide range SG level in either SG drops below 15% (RCS pressure and hot leg temperature increasing for adverse containment) or pressurizer pressure increases above 2335 psig. The operators use the SI pumps for injection and establish an RCS bleed path by opening at least one of two pressurizer PORVs. It is likely that bleed and feed cooling using FR-H.1 would be established by 30 minutes. SG secondary dryout is expected at approximately one hour.

Success of OB1 is 1 of 2 high pressure SI trains delivering flow to 1 of 2 RCS cold legs with 1 of 2 pressurizer PORVs open. Bleed and feed initiation prior to SG dryout with this success criterion is expected to result in effective decay heat removal. For simplicity, it is assumed that bleed and feed initiated by 30 minutes using one SI pump and one pressurizer PORV results in success.

It is assumed that failure of this node results in early core melt due to loss of all secondary cooling.

LOW PRESSURE INJECTION (LI2)

If high pressure injection fails, operator actions must be taken to cool down and depressurize the RCS and use low pressure SI (LI2) to provide core cooling and inventory. Low pressure SI is provided by the RHR pumps which inject borated water from the RWST to the reactor vessel. The operators manually re-initiate low pressure SI following RCS cooldown and depressurization.

Success of this node is 1 of 2 low pressure injection trains delivering flow to the reactor vessel.

It is assumed that failure of this node after failure of HI2 results in early core melt.

LOW PRESSURE RECIRCULATION (LR1)

If LI2 is required to mitigate this event, long term cooling with low pressure recirculation (LR1) is required. During the injection phase of the accident, water spilled from the break is collected on the containment floor and in the sump. Long term cooling is provided by the RHR pumps taking suction from the containment sump and discharging to the reactor vessel via the residual heat exchangers.

Success of LR1 is 1 of 2 low pressure recirculation trains delivering flow to the reactor vessel.

It is assumed that failure of this node will result in late core melt.

LOW PRESSURE RECIRCULATION (LR2)

If HR1 fails, long term cooling is with low pressure recirculation (LR2). LR2 consists of a low pressure SI train lined up to the containment sump and discharging to the reactor vessel via the residual heat exchangers.

Success of LR2 is 1 of 2 low pressure recirculation trains delivering flow to the reactor vessel.

It is assumed that failure of this node and HR1 results in late core melt.

FIGURE 3.1[illegible]3-3

SMALL LOCA E[illegible]T TREE

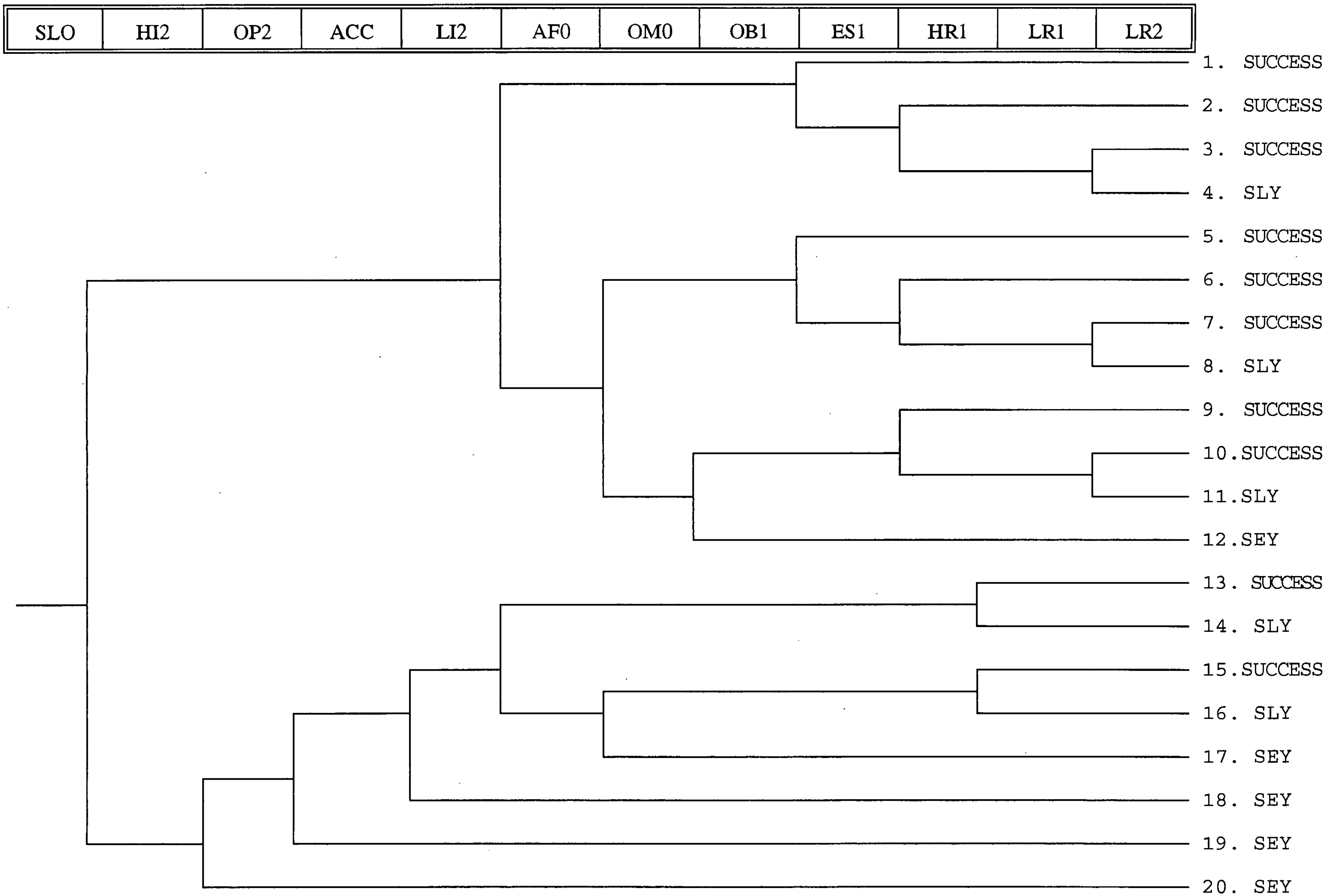


TABLE 3.1.2/3-3

SUCCESS CRITERIA FOR SMALL LOCA

Top Event Description		System Success Criteria	Necessary Operator Actions	Mission Time (hrs)
HI2 -	HIGH PRESSURE INJECTION	1 of 2 high pressure SI trains injecting contents of BAT and RWST to 1 of 2 RCS cold legs	Confirm operation of system. If pumps not automatically started, manually start.	3.5
HR1 -	HIGH PRESSURE RECIRCULATION	1 of 2 SI/RHR trains delivering flow from containment sump to 1 of 2 RCS cold legs, sump valve on operable recirc. train open.	Manually align high pressure containment sump recirculation on low RWST level (may include re-start of RHR pump), align CCW cooling to residual Hx, confirm operation of system.	20.5
OP2 -	OPERATOR ACTION-COOLDOWN AND DEPRESSURIZE RCS FOR ACC AND LI2	Operator initiated cooldown started within 30 minutes using at least one SG supplied with feedwater.	Per EOP ES-1.2, cooldown RCS by dumping steam at max 100 F/hr. Depressurize RCS to inject accumulators and permit initiation of low pressure SI.	Potentially 24 hours (steam relief needed for decay heat removal).
AF0 -	AUXILIARY FEEDWATER	1 of 3 AFW pumps delivering to at least 1 of 2 steam generators, total flow of at least 200 gpm.	Confirm operation of system.	24

TABLE 3.1.2/3-3

SUCCESS CRITERIA FOR SMALL LOCA (Continued)

Top Event Description	System Success Criteria	Necessary Operator Actions	Mission Time (hrs)
OM0 - OPERATOR ACTION-ESTABLISH MAIN FEEDWATER	1 of 2 MFW trains delivering at least 200 GPM to 1 of 2 steam generators.	Manually align and initiate MFW. Confirm operation of system.	24
LI2 - LOW PRESSURE INJECTION	1 of 2 low pressure SI trains injecting flow into reactor vessel.	Manually initiate low pressure SI following RCS cooldown and depressurization, confirm operation of system.	1
LR1 - LOW PRESSURE RECIRCULATION	1 of 2 low pressure SI trains in recirc from containment sump to reactor vessel via residual heat exchangers, sump valve on operable recirc train open	Manually align low pressure containment sump recirc on low RWST level (may include re-start of RHR pump), align CCW cooling to residual Hx, confirm operation of system.	23
LR2 - LOW PRESSURE RECIRCULATION	1 of 2 low pressure SI trains in recirc from containment sump to reactor vessel via residual heat exchangers, sump valve on operable recirc train open	Manually align low pressure containment sump recirc on low RWST level (may include re-start of RHR pump), align CCW cooling to residual Hx, confirm operation of system.	23

TABLE 3.1.2/3-3

SUCCESS CRITERIA FOR SMALL LOCA (Continued)

Top Event Description		System Success Criteria	Necessary Operator Actions	Mission Time (hrs)
OB1 -	OPERATOR ACTION - BLEED AND FEED	1 of 2 high pressure SI trains delivering flow to 1 of 2 RCS cold legs; 1 of 2 pressurizer PORVs open (bleed and feed initiated prior to secondary dryout - assume at 30 minutes).	Manually open PORVs and block valves, verify SI pumps running (see FR-H.1)	Run for 24 hours.
ACC -	ACCUMULATOR INJECTION	1 of 1 accumulator on the intact loop injecting into the cold leg.	Confirm operation of system.	None
ES1 -	OPERATOR ACTION - COOLDOWN AND DEPRESSURIZE RCS FOR CHARGING FLOW	Cooldown and depressurize RCS to near atmospheric pressure to avoid depleting RWST (consider for very small breaks).	Cooldown RCS using SGs, depressurize RCS using spray or one pressurizer PORV, reduce SI by stopping high pressure SI pumps, operate 2 of 3 charging pumps for makeup, align RHR system for cooldown to cold shutdown.	24

3.1.2/3.5 Steam Generator Tube Rupture

INITIATING EVENT - STEAM GENERATOR TUBE RUPTURE (SGR)

The steam generator tube rupture is initiated by a random or consequential rupture of a steam generator tube ranging from a small tube leak up to a double-ended break of a single tube.

AUXILIARY FEEDWATER (AF1)

Auxiliary feedwater (AF1) is required to remove decay heat for a SGR. AFW is actuated on lo-lo steam generator level or by the SI signal. Because of KNPP's design, only 2 AFW pumps, 1 motor driven and the turbine driven, can provide flow to 1 SG once the other SG is isolated.

It is assumed that secondary cooling is required for the entire event. Success of AF1 is 1 of 2 AFW pumps supplying at least 200 gpm to the intact steam generator for the entire event.

OPERATOR ACTION - ESTABLISH MAIN FEEDWATER (OM1)

If AF1 fails, main feedwater (OM1) is used for secondary cooling. Emergency Operating Procedure FR-H.1, Response to Loss of Secondary Heat Sink, is entered via the critical safety function trees. FR-H.1 instructs the operators to attempt to restore AFW and to line up MFW to provide secondary cooling. If MFW is unavailable, the operators are instructed to depressurize the primary system in order to block SI and then to depressurize the intact SG in order to provide secondary coolant flow from the condensate system. However, due to the complexity of the operator actions required to establish flow to a steam generator from the condensate system alone, it would take a significant amount of time to establish this flow and it is not included in the event tree modeling.

Success of OM1 is 1 of 2 MFW trains delivering at least 200 gpm flow to the intact steam generator for the entire event.

It is assumed that if OM1 fails, early core melt always occurs.

HIGH PRESSURE INJECTION (HI1)

High pressure injection (HI1) requires the automatic or manual actuation of SI and operation of at least one high pressure SI train delivering water to the RCS.

Failure of high pressure SI for the SGR event does not necessarily mean that core uncovery will occur. The operator must perform the actions in EOPs E-3 (OS1 and OS2) or ECA-3.1/3.2 (EC3) (if the RCS reaches saturation) to cool down and depressurize the RCS to less than the ruptured SG pressure before RCS inventory loss through the SGR results in core uncovery and subsequent core damage. If the SG is not isolated, it is assumed that depressurization to the ruptured SG pressure will occur rapidly due to the break flow and no SI flow, and the actions in ECA-3.1/3.2 need to be performed in order to ensure long term cooling. These actions are represented by node EC3. In the case where HI1 fails and SGs are successfully isolated, the nodes OS1 and OS2 are used, even though a momentary loss of subcooling may cause operators to use ECA-3.1/3.2 instead of E-3. Since the actions of ECA-3.1/3.2 and E-3 are very similar, the use of OS1 and OS2 is appropriate.

Success of HI1 is 1 of 2 high pressure safety injection trains delivering flow to 1 of 2 RCS cold legs.

It is assumed that the failure of HI1 along with a loss of all feedwater (AF1 and OM1) will result in early core melt.

STEAM GENERATOR ISOLATION BY MSIV CLOSURE (ISO)

In the normal EOP E-3 Steam Generator Tube Rupture recovery, the ruptured SG is isolated from the intact SG by closure of an MSIV. Other paths to and from the ruptured SG also require isolation (e.g., blowdown, steam supply to the turbine-driven AFW pump, etc.). Isolation of these paths, however, is not as crucial to the recovery as main steam isolation. It is preferable to close the MSIV for the ruptured SG since this gives the operator the option of using steam dump to condenser, if available, for the subsequent cooldown using the intact SGs. Should the MSIV for the ruptured SG fail to close, the MSIV for the intact SG is closed and the corresponding SG PORV used for the cooldown. Since the initial cooldown is limited (i.e., to about 500°F), only one SG is required for the cooldown. Therefore, success for the ISO function is determined by the ability to close at least one MSIV on any SG. For a design basis SGR, it is assumed that the operator must identify the ruptured SG and perform the ISO isolation function within 15 minutes for ISO to be successful.

If this node fails, the recovery actions in ECA-3.1 or ECA-3.2 (EC3) need to be addressed.

OPERATOR ACTION TO COOL DOWN AND DEPRESSURIZE THE RCS AND TERMINATE SAFETY INJECTION BEFORE RUPTURED SG OVERFILLS (OS1)

Success of this action requires the operator to successfully complete the actions in EOP E-3, Steam Generator Tube Rupture, to stabilize RCS pressure less than the ruptured SG pressure before the ruptured SG fills due to the addition of AFW and break flow. This normally requires three different high level operator actions: initial cooldown, RCS depressurization, and SI termination. The initial cooldown is performed using the intact SG supplied with feedwater, which has been isolated from the ruptured SG. The RCS depressurization is accomplished using normal or auxiliary spray, if available, or by opening one pressurizer PORV (and its associated block valve, if necessary). Of the three high level actions for OS1, the initial cooldown is the most essential one to model in the fault tree analysis. This is because success for SI termination is easy to demonstrate and the leak eventually causes the RCS to depressurize to the ruptured SG pressure if the SI pumps are secured and the intact SG is maintained at a lower pressure than the ruptured SG. These actions should be completed in about 30 minutes to prevent SG overfill for a design basis SGR event. This assumes simultaneous completion of all three actions at 30 minutes. The expected SG overfill time if the cooldown and depressurization are performed earlier could be significantly longer since the break flow would be reduced while these actions are being completed.

Success of this node requires operator action to cool down and depressurize the RCS in order to stop the primary to secondary leak. These actions must be completed within 30 minutes. Success of OS1 results in no core melt if the RCS cooldown is successfully stopped.

OPERATOR ACTION TO COOL DOWN AND DEPRESSURIZE THE RCS AND TERMINATE SAFETY INJECTION AFTER RUPTURED SG OVERFILLS (OS2)

This event models the same actions as OS1. It is assumed, however, that the ruptured SG overfills prior to the completion of these actions. The 1982 Ginna event is an example of this case. For this SGR, the ruptured SG overfilled and one of the safety valves briefly opened, possibly several times. Upon SI termination, the safety valve did re-seat and the recovery proceeded normally.

Success of this node requires operator action to cooldown and depressurize the RCS and stop the primary to secondary leak after the ruptured SG has overfilled. Failure of this node results in late core melt.

OPERATOR ACTION TO STOP THE RCS DEPRESSURIZATION (OSD)

Nodes OS1 and OS2 require a depressurization of the RCS. If the pressurizer PORVs are used for this purpose, there is a possibility that they do not close, in which case RCS inventory is lost through the PORVs.

Success of this node requires that either the pressurizer sprays are used for depressurization or the one PORV used for depressurization is successfully closed.

Success of OSD always results in no core melt. Failure of OSD and either HI1 or high pressure recirculation (HR1) always results in core melt.

INTEGRITY MAINTAINED OR RESTORED IN RUPTURED STEAM GENERATOR (SSV)

If one of the secondary relief valves sticks opens following overfill of the ruptured SG, the SGR recovery strategy becomes somewhat more complicated. The operator transitions to EOP ECA-3.1, SGTR With Loss of Reactor Coolant - Subcooled Recovery Desired Saturated Recovery Desired, and possibly to ECA-3.2, SGTR With Loss of Reactor Coolant. Success is defined as all 5 safety valves and the PORV closing to maintain or restore secondary integrity after the E-3 Steam Generator Tube Rupture actions are complete.

It is assumed that if SSV fails and high pressure SI is available, recovery actions in ECA-3.1 or ECA-3.2 (EC4) must be addressed. If SSV fails and high pressure SI is unavailable, early core melt is assumed.

OPERATOR ACTIONS TO COOL DOWN AND DEPRESSURIZE THE RCS WITH ECA-3.1/3.2 (EC3)

The ECA-3.1 and possibly ECA-3.2 recovery actions need to be followed if the ruptured SG can not be isolated from the intact SG used for cooldown (ISO fails).

In ECA-3.1, the operator initiates a cooldown at a maximum rate of 100°F/hr using the intact SG. Once RHR entry conditions are established (RCS pressure less than 425 psig and coldest RCS wide range temperature less than 380°F), the RHR system can be placed in service to continue the cooldown to cold shutdown (200°F). As the cooldown progresses, the high pressure SI pumps are sequentially stopped (based on specified subcooling and RCS inventory criteria) until the charging pumps are able to supply the required makeup. The RCS is also depressurized either using pressurizer spray or a PORV to minimize the break flow to the ruptured SG and the environment. If RWST level decreases to below 52% or narrow range level in the ruptured SG increases to above 92%, step 13 of ECA-3.1 instructs the operators (or Emergency Director) to determine if a transition to ECA-3.2, SGTR With Loss of Reactor Coolant - Saturated Recovery Desired, is appropriate.

In ECA-3.2, the subcooling and inventory criteria are relaxed to allowed a more expedited recovery. The ultimate objective of the ECA-3.1/3.2 recovery is to depressurize the RCS and ruptured SG to near atmospheric pressure and thereby terminate the leak. Although the actions appear complex, the operator has roughly 6 to 10 hours for them to be successful for a design basis SGR provided high pressure SI is available. This is the approximate RWST depletion time for a break with an average injection flow requirement ranging from 300 to 600 gpm.

For the success for EC3, successful RHR system operation is required. A requirement for operation of at least 2 of 3 charging pumps is also included since this would allow the SI pumps to be stopped at reasonable subcooling values and would allow makeup control after the SI pumps are secured. Charging pumps are also required for auxiliary spray success.

It is assumed that the failure of EC3 always results in core melt.

OPERATOR ACTIONS TO COOL DOWN AND DEPRESSURIZE THE RCS WITH ECA-3.1/3.2 (EC4)

In addition to the scenario described above for EC3, ECA-3.1 and possibly ECA-3.2 recovery actions need to be followed if SG overflow occurs and a secondary side relief valve on the ruptured SG sticks open (OS1 and SSV fail).

EC4 represents the same operator actions as EC3. Since it follows the failure of OS1 however, it has a higher failure probability due to operator dependence (refer to Section 3.3.3 of this report for more details).

It is assumed that failure of EC4 always results in core melt.

HIGH PRESSURE RECIRCULATION (HR1)

If OSD fails, then recirculation is eventually required to maintain inventory. Success requires switchover from the RWST to the containment sump, associated valve realignments, plus operation of one of the low-pressure SI (RHR) pumps to feed either the reactor vessel or the suction to the high pressure SI pump(s). Most likely, switchover will be required several hours after RCS depressurization, so high pressure recirculation is conservatively assumed to be required. Two pressurizer PORVs have sufficient capacity to maintain RCS pressure near the shutoff head pressure of the RHR pumps (approximately 200 psig) at this time, so it may be possible to relax the requirement that high pressure SI be available for this case.

Failure of HR1 results in late core melt.

FIGURE 3.1.2/3-4 STEAM GENERATOR TUBE RUPTURE EVENT TREE

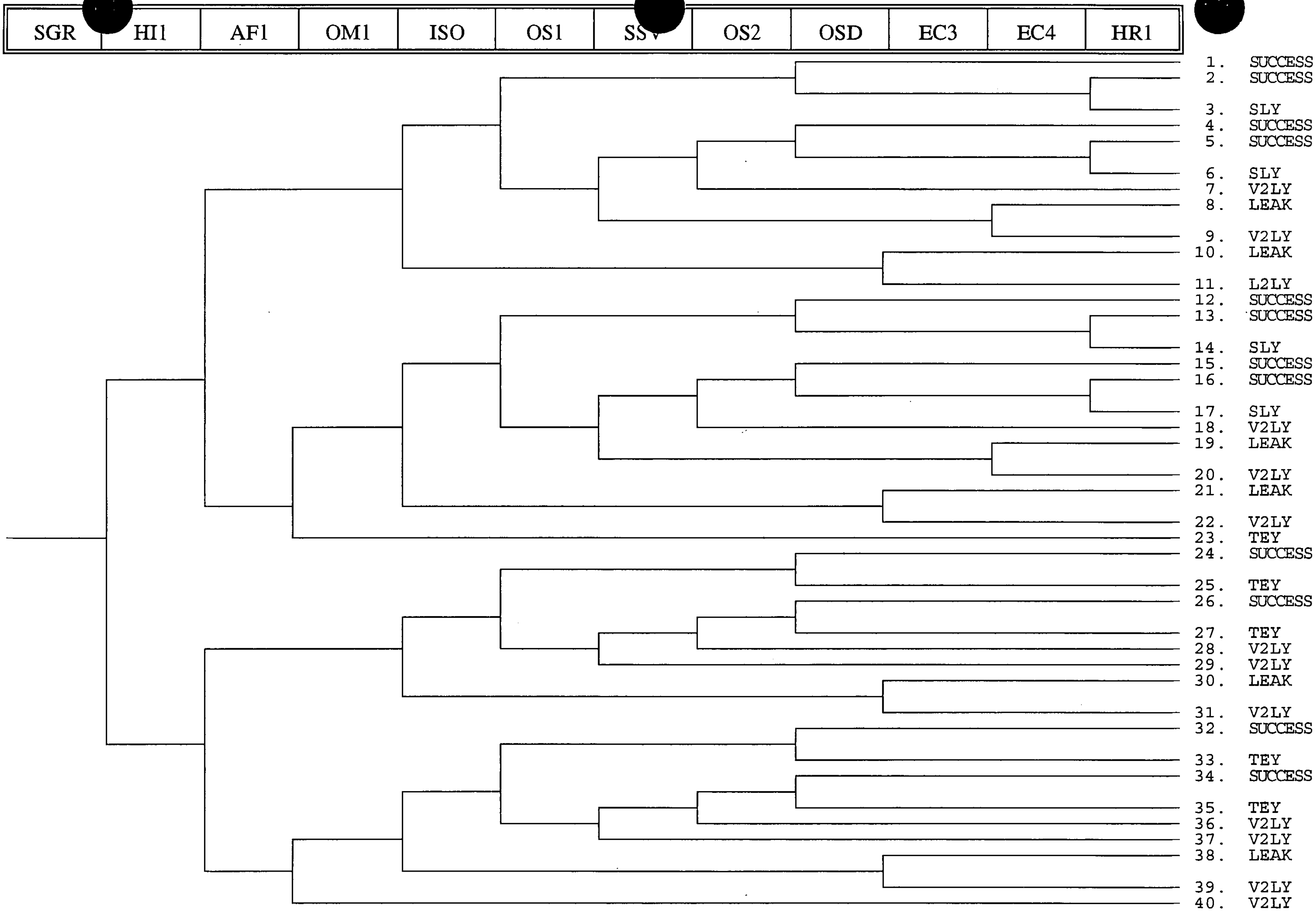


TABLE 3.1.2/3-4

SUCCESS CRITERIA FOR SGR

Top Event Description	System Success Criteria	Necessary Operator Actions	Mission Time (hrs)
HI1 - HIGH PRESSURE INJECTION	1 of 2 high pressure SI trains injecting contents of BAT and RWST to 1 of 2 RCS cold legs	Confirm operation of system	3.5
AF1 - AUXILIARY FEEDWATER	1 of 2 AFW pumps delivering 200 GPM to intact steam generator.	Confirm operation of system	24
OM1 - OPERATOR ACTION-ESTABLISH MAIN FEEDWATER	1 of 2 MFW trains delivering at least 200 GPM to intact steam generator.	Manually align and initiate MFW. Confirm operation of system.	24
ISO - STEAM GENERATOR ISOLATION BY MSIV CLOSURE	Isolation of ruptured SG - Closure of 1 of 2 MSIVs (within about 15 minutes for a design basis SGTR).	Diagnose ruptured SG, close MSIV on ruptured SG or close MSIV on intact SG and use intact SG PORVs for initial cooldown.	None
OS1 - OPERATOR ACTION TO COOLDOWN AND DEPRESSURIZE THE RCS AND TERMINATE SI BEFORE RUPTURED SG OVERFILLS	Stabilize RCS pressure with ruptured SG pressure before overfill of the ruptured SG (within about 30 minutes for a design basis SGTR)	Cool down the RCS by dumping steam from the intact SG. Depressurize the RCS using pressurizer spray or PORVs in order to stop the leak and maintain subcooling. Stop all SI pumps.	None
OS2 - OPERATOR ACTION TO COOLDOWN AND DEPRESSURIZE THE RCS AND TERMINATE SI AFTER RUPTURED SG OVERFILLS	Stabilize RCS pressure with ruptured SG pressure after overfill of the ruptured SG (assume approximately 60 minutes)	Cool down the RCS by dumping steam from the intact SG. Depressurize the RCS using pressurizer spray or PORVs in order to stop the leak and maintain subcooling. Stop all SI pumps.	None

TABLE 3.1.2/3-4

SUCCESS CRITERIA FOR SGR (Continued)

Top Event Description	System Success Criteria	Necessary Operator Actions	Mission Time (hrs)
OSD - OPERATOR ACTION TO STOP THE RCS DEPRESSURIZATION	Either depressurization was accomplished with pressurizer spray valves, or close 1 of 1 open pressurizer PORV's	Close PORV.	None
SSV - INTEGRITY MAINTAINED OR RESTORED IN RUPTURED SG	No secondary relief valves for for the ruptured SG open.	None.	None
EC3 - OPERATOR ACTIONS TO COOLDOWN AND DEPRESSURIZE THE RCS PER ECA-3.1/3.2	Depressurize RCS and ruptured SG to near atmospheric pressure prior to draining RWST (6-10 hour time frame).	Cooldown using intact SG or possibly ruptured SG, depressurize RCS using spray or one Pzr PORV, reduce SI by stopping high pressure SI pumps, operate 2 of 3 charging pumps for makeup, align RHR system for cooldown to cold shutdown.	None
EC4 - OPERATOR ACTIONS TO COOLDOWN AND DEPRESSURIZE THE RCS PER ECA-3.1/3.2	Depressurize RCS and ruptured SG to near atmospheric pressure prior to draining RWST (6-10 hour time frame).	Cooldown using intact SG or possibly ruptured SG, depressurize RCS using spray or one Pzr PORV, reduce SI by stopping high pressure SI pumps, operate 2 of 3 charging pumps for makeup, align RHR system for cooldown to cold shutdown.	None

TABLE 3.1.2/3-4

SUCCESS CRITERIA FOR SGR (Continued)

Top Event Description	System Success Criteria	Necessary Operator Actions	Mission Time (hrs)
HR1 - HIGH PRESSURE RECIRCULATION	1 of 2 SI/RHR trains delivering flow from containment sump to 1 of 2 RCS cold legs, sump valve on operable recirc train open.	Manually align high pressure containment sump recirculation on low RWST level (may include re-start of RHR pump), align CCW to residual Hx, confirm operation of system.	20.5

3.1.2/3.6 Vessel Failure

VESSEL FAILURE (VEF)

This initiator includes any LOCA that is beyond the capability of the ECCS.

FIGURE 3.1.2/3-5

VESSEL FAILURE EVENT TREE

VEF

1. AEY

3.1.2/3-7 Interfacing Systems LOCA

INITIATING EVENT - INTERSYSTEM LOCA (ISL)

This initiating event frequency is the summation of all the frequencies associated with the credible high pressure/low pressure flowpaths that lead outside of containment.

RHR SYSTEM BREAK (BR)

This node identifies the magnitude of the RHR system break, i.e. whether it is a piping failure or a pump seal failure.

Success of BR is no failure of the RHR piping. With no failure of the RHR piping it is expected that prolonged exposure of the pumps seals to high temperature and pressure results in seal leakage. The success path of this node therefore assumes the common mode failure of the pump seals in both RHR trains. This results in a maximum break area of 0.1 ft^2. It is further assumed that the seal leakage results in a loss of both RHR pumps.

Failure of this node is assumed to be the limiting case of a circumferential rupture of the 10 inch common pump suction piping outside of containment. This would result in a rapid loss of primary coolant inventory outside containment.

OPERATOR ACTION - ISOLATE RHR PUMPS (OIP)

This node determines whether or not the operators are successful in manually closing valves RHR-4A and 4B to isolate both RHR pumps assuming that each pump's seal is leaking. Because of the length of time associated with this action, it is assumed that by the time this isolation is complete both RHR pumps are inoperable and RCS pressure is low enough, due to the pressure relief provided by the RHR relief valves, to not cause RHR piping failure. If this node is unsuccessful, core damage occurs unless a water source is available once the RWST is depleted.

The success of OIP is 2 of 2 RHR pump manual isolation valves closed.

HIGH PRESSURE INJECTION (HI4)

High pressure injection is automatically actuated on an SI signal upon receiving a low pressurizer pressure signal. The SI pumps first take suction from the boric acid tank, then from the RWST. The SI pumps inject into the RCS cold legs.

Success of HI4 is 1 of 2 SI pumps delivering flow to 1 of 2 cold legs. An SI signal must be successfully generated. Failure of this node results in early core melt.

RHR RELIEF VALVES CLOSE (RVC)

This node models the success of the RHR relief valves to close once RCS/RHR pressure is below the relief setpoint of both valves (approximately 480 psig).

Success of RVC is 2 of 2 relief valves closed. Failure of this node results in a failure to isolate the LOCA through the relief valves. This results in early core melt unless ECCS flow is minimized.

AUXILIARY FEEDWATER (AF0)

Auxiliary feedwater (AF0) is required to remove decay heat for an IS LOCA. AFW would be actuated on lo-lo steam generator level or by the SI signal.

It is assumed that secondary cooling is required for the entire event. Success of AF0 is 1 of 3 AFW pumps supplying at least 200 gpm to 1 of 2 steam generators for the entire event.

OPERATOR ACTION - ESTABLISH MAIN FEEDWATER (OM0)

If AF0 fails, Main Feedwater (OM0) is used for secondary cooling. Emergency Operating Procedure FR-H.1, Response to Loss of Secondary Heat Sink, is entered via the critical safety function trees. FR-H.1 instructs the operators to attempt to restore AFW and to line up MFW to provide secondary cooling. If MFW is unavailable, the operators are instructed to depressurize the primary system in order to block SI and to depressurize a steam generator in order to provide secondary coolant flow from the condensate system. However, due to the complexity of the operator actions required to establish flow to a steam generator from the condensate system alone, it would take a significant amount of time to establish this flow and it is not included in the event tree modeling.

Success of OM0 is 1 of 2 MFW trains delivering flow to 1 of 2 SGs for the entire event with a flow rate of at least 200 gpm.

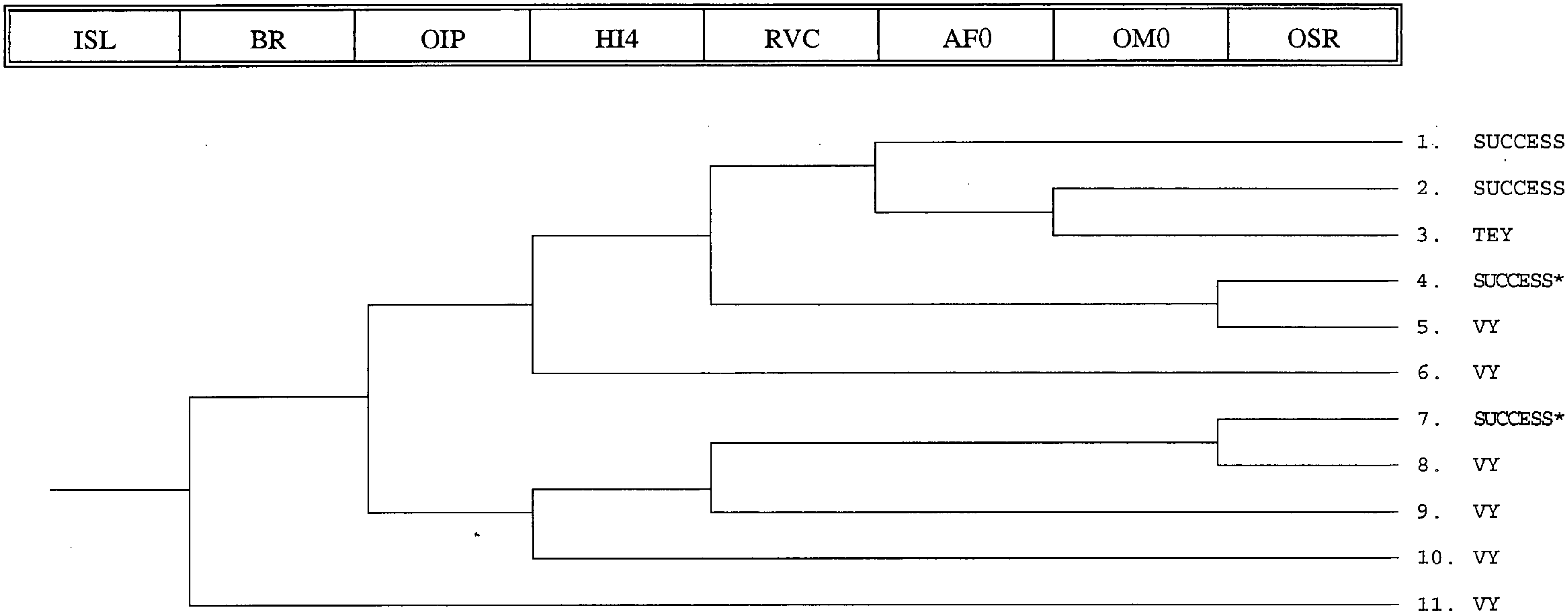
OPERATOR ACTION - MINIMIZE ECCS FLOW (OSR)

This node models the operator actions necessary to minimize ECCS flow upon recognition of loss of recirculation capability. These actions are directed by Emergency Operating Procedure ECA-1.1, Loss of Emergency Coolant Recirculation, which is entered from either Step 16 of E-1, Loss of Reactor or Secondary Coolant, or Step 3 of ECA-1.2, LOCA Outside Containment.

It is assumed that failure of this node will result in early core melt as the RWST would be depleted thus eliminating ECCS injection into the RCS.

FIGURE 3.1.2/3-6

INTERFACING SYSTEMS LOCA EVENT TREE



*Success in this case refers to core melt being delayed.

TABLE 3.1.2/3-6

SUCCESS CRITERIA FOR INTERFACING SYSTEMS LOCA

Top Event Description		System Success Criteria	Necessary Operator Actions	Mission Time (hrs)
BR -	RHR SYSTEM BREAK	RHR system piping does not fail, but both RHR pump seal quickly develop excess leakage.	None	None
OIP -	OPERATOR ACTION ISOLATE RHR PUMPS	2 of 2 RHR pump manual isolation valves closed.	Operator locally closes valves RHR-4A and 4B	None
HI4 -	HIGH PRESSURE INJECTION	1 of 2 high pressure SI trains injecting contents of BAT and RWST into 1 of 2 legs.	Confirm operation of system.	24
RVC -	RHR RELIEF VALVES CLOSED	2 of 2 RHR relief valves closed when RCS pressure drops below relief valve setpoints.	None	None
AF0 -	AUXILIARY FEEDWATER	1 of 3 AFW pumps delivering to at least 1 of 2 steam generators, total flow of at least 200 gpm	Confirm operation of system	Run for 24 hours

TABLE 3.1.2/3-6

SUCCESS CRITERIA FOR INTERFACING SYSTEMS LOCA (Continued)

OM0 -	OPERATOR ACTION ESTABLISH MAIN FEEDWATER	1 of 2 MFW trains delivering at least 200 GPM to 1 of 2 steam generator.	Manually align and initiate MFW. Confirm operation of system.	Run for 24 hours.
OSR -	OPERATOR ACTION -MINIMIZE ECCS FLOW	High pressure SI flow reduced to amount needed to remove decay heat.	With EOP ECA-I.1, operators reduce SI flow by manually throttling SI-7A or SI-7B.	None

3.1.2/3.8 Transients With Main Feedwater

INITIATING EVENT - TRANSIENTS WITH MAIN FEEDWATER (TRA)

This initiating event includes all transients except those that result in total loss of the main feedwater system and certain special initiating events described in section 3.1.1. In Emergency Operating Procedure ES-0.1, Reactor Trip Response, the main feed regulating valves are expected to close on low TAVE (554°F). Since the steam dump system (to condenser or atmosphere) automatically controls to no-load TAVE and the operator is instructed to initially verify and maintain RCS temperature near no-load (547°F), it is expected that for most transients, the main feed regulating valves will automatically shut within minutes after reactor trip. However, MFW is available provided an SI actuation signal or high SG level signal are not present. The event tree developed here, therefore, applies to those transients in which MFW can be used or recovered by operator action.

POWER AVAILABLE (OSP)

If offsite power is lost at any time during the transient, the emergency diesel generators (EDGs) are designed to automatically start and come up to speed within 10 seconds.[14] If the EDG fails to start the first time, several restarts will be attempted automatically. Operator action is then required for additional start attempts. Success for the OSP top event is to have emergency AC power (either from the EDGs or from offsite power) available to at least one of the two 4.16 kV emergency buses. If onsite power is lost, the failure to have AC power to these buses may be due to failure of the EDGs to start or to run, failure of the buses to shed loads, or failure of the EDGs to load. For a loss of offsite power with successful reactor trip, provision of emergency AC power could be delayed for as long as 30 minutes, i.e., a limiting time for SG secondary dryout. For event tree modeling, it is assumed that AC power is required for 24 hours. During this time, a fuel oil immersion pump would be required to operate periodically to replenish the 850 gallon day tanks for the EDGs.

AUXILIARY FEEDWATER (AF3)

This top event models the availability of the AFW system to remove decay heat. Success of the AFW system requires operation of at least one of three AFW pumps delivering a total flow of 200 gpm to at least one SG. The AFW pumps are started on low SG levels or by manual actuation. The success criterion assumed here is the same as that used for the loss of normal feedwater event (plus several other transients) described in the Kewaunee USAR.[14]

Success for AF3 also assumes adequate steam relieving capability. This can be achieved by operation of steam dump to condenser, if available, or by operation of one of the relief valves (one SG PORV, 5 safety valves) for each active SG.

OPERATOR ACTION - ESTABLISH MAIN FEEDWATER (OM2)

This top event models the availability of the MFW system to remove decay heat via the SG if AFW is unavailable. As explained above, success of the MFW system requires one source of MFW delivering to at least one SG. The following paragraphs describe the additional actions needed to ensure availability of this source of MFW.

In ES-0.1, Reactor Trip Response, the operator is directed to verify or control TAVE near 547°F and pressurizer pressure near 2235 psig. Based on these conditions, the main feed regulating valves are expected to shut (based on TAVE <554°F); the operator also places the main feed regulating valve controllers in manual and verifies valve closure. Since pressure is maintained above 1815 psig, it is not likely that SI actuation would occur. Thus, MFW is still available for secondary cooling if needed. ES-0.1 therefore directs the operator to verify 200 gpm total feedwater flow using either MFW or AFW. ES-0.1 further directs the operator to maintain SG narrow range levels between 4% and 50%, so it is not likely that MFW is stopped due to high SG level.

If MFW is needed after feedwater isolation because AFW fails, it is most likely restored using FR-H.1, Response to Loss of Secondary Heat Sink. This procedure is entered if the secondary inventory in both SGs is low and the total feedwater flow is less than 200 gpm. To recover feedwater, the operator first attempts to establish MFW to at least one SG. This requires operation of a condensate pump and a MFW pump. If SI has been actuated (not expected for this event) it is necessary to reset the SI signal and the feed regulating bypass valve lockout from SI.

If MFW is unavailable, the operators are instructed to depressurize the RCS in order to block SI and depressurize a SG in order to provide secondary coolant flow from the condensate system. However, due to the complexity of the operator actions required to establish flow to a SG from the condensate system alone, it would take a significant amount of time to establish this flow and it is not included in the event tree modeling.

Success of OM2 is 1 of 2 MFW trains delivering a flow of at least 200 gpm to 1 of 2 SGs.

OPERATOR ACTION, BLEED AND FEED (OB2)

If AF3 and OM2 fail, the operators are instructed to initiate primary system bleed and feed. Emergency operating procedure FR-H.1, Response to Loss of Secondary Cooling, instructs the operators to initiate bleed and feed if secondary cooling is lost and wide range steam generator level in either steam generator drops below 15% or pressurizer pressure increases above 2335 psig. The operators start at least one high pressure SI pump and establish an RCS bleed path by opening at least one of two pressurizer PORVs. It is likely that bleed and feed cooling, according to FR-H.1 instructions, would be established by 30 minutes. SG secondary dryout is expected at approximately one hour.

Success of this node is 1 of 2 high pressure SI trains delivering flow to 1 of 2 RCS cold legs with at least one pressurizer PORV open. Bleed and feed initiation prior to SG dryout with this success criterion is expected to result in effective decay heat removal. For simplicity, it is assumed that bleed and feed initiated by 30 minutes using one high pressure SI pump and one pressurizer PORV results in success.

It is assumed that failure of this node results in early core melt due to loss of all secondary cooling.

HIGH PRESSURE RECIRCULATION (HR1)

If OB2 is successful, long term cooling is addressed. Long term cooling is provided by sump recirculation. If RCS pressure remains above 140 psig, a low pressure SI train is lined up to take suction from the containment sump and discharge to the suction of the high pressure safety injection pumps via the residual heat exchangers. This lineup is referred as a SI/RHR train in the Emergency Operating Procedures.

Success of HR1 is 1 of 2 SI/RHR trains providing flow to 1 of 2 RCS cold legs.

It is assumed that failure of this mode will result in late core melt.

CHARGING PUMP OPERATION (CHG)

The operator must maintain a minimum amount of charging flow to supply RXCP seal injection. One charging pump will provide adequate RXCP seal cooling and thereby prevent a small LOCA due to seal degradation following loss of all seal cooling. Continued post-trip operation of the charging pump plus operator training ensure that seal injection is maintained with little or no interruption following reactor trip. In addition, Step 4 of EOP ES-0.1, Reactor Trip Response, explicitly directs the operator to verify or establish charging flow. It is highly probable that Step 4 of EOP ES-0.1 is reached within 10 minutes following reactor trip since the EOP E-0 to EOP ES-0.1 transition occurs very quickly (via Step 4 of EOP E-0). Based on the expected RXCP seal response to the loss of all cooling described in Section 10.1 of WCAP-10541, Rev. 2[24], a normal seal flow requirement of 3 to 5 gpm per pump (less than 10 gpm total) is expected if seal injection is restored by 10 minutes, i.e., prior to the transient heatup phase. Even if seal injection is delayed until about 30 minutes, the seal leakage rate is expected to be less than 21 gpm per pump or 42 gpm total. This is still within the capacity of one of the 60 gpm positive-displacement charging pumps.

Based on the above description, success for the CHG top event requires either

1. continued operation of 1 of 3 charging pumps for seal injection, or

2. operator action (based on training or the EOPs) to start a charging pump for seal injection within 30 minutes following reactor trip.

COMPONENT COOLING WATER (CCW)

If CHG fails, RXCP seal injection is lost and seals are cooled by RCS water flowing past the thermal barrier which is cooled by CCW. Without this cooling, it is assumed that the RXCP seals fail, resulting in a small LOCA. It is conservatively assumed that core melt results form this LOCA.

FIGURE 3.1.2/3-7

TRANSIENTS WITH MAIN FEEDWATER EVENT TREE

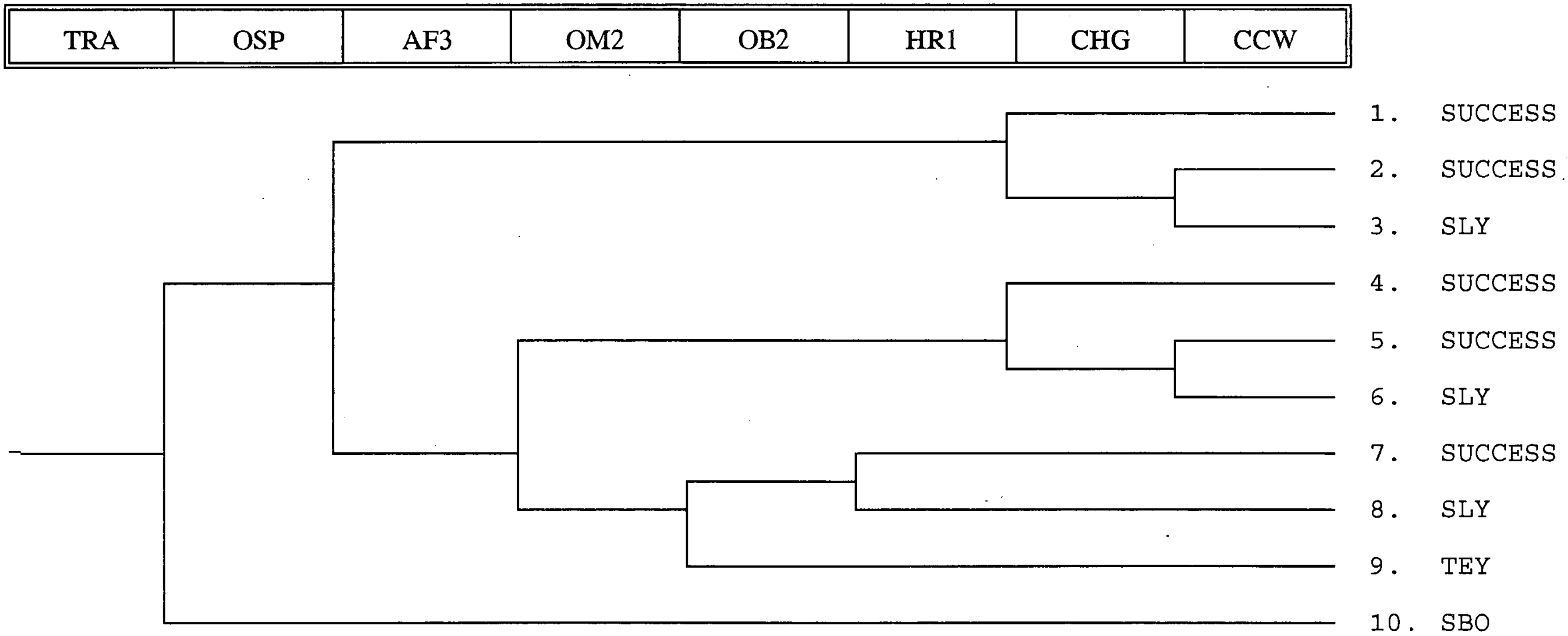


TABLE 3.1.2/3-7

SUCCESS CRITERIA FOR TRANSIENTS WITH MAIN FEEDWATER

Top Event Description	System Success Criteria	Necessary Operator Actions	Mission Time (Hrs)
OSP - ONSITE EMERGENCY AC POWER	Emergency AC Power available to at least one of two 4.16 kV safeguards buses.	May have to manually start diesel.	24
AF3 - AUXILIARY FEEDWATER	1 of 3 AFW pumps delivering at least 200 GPM to at least 1 of 2 steam generators.	If AFW not automatically initiated by event, manually initiate AFW. Confirm operation of system.	24
OM2 - OPERATOR ACTION - ESTABLISH MAIN FEEDWATER	1 of 2 MFW trains delivering at least 200 GPM to at least 1 of 2 steam generators.	If AFW automatically initiated by event but is not available, manually align and initiate MFW. Confirm operation of system.	24
OB2 - OPERATOR ACTION - BLEED AND FEED	1 of 2 high pressure SI trains delivering flow to 1 of 2 RCS cold legs, 1 of 2 pressurizer PORVs open (bleed and feed initiated prior to secondary dryout - assume at 30 minutes).	Manually open PORVs and block valves, start SI pumps (see FR-H.1).	Run for 24 hours.

TABLE 3.1.2/3-7

SUCCESS CRITERIA FOR TRANSIENTS WITH MAIN FEEDWATER (Continued)

HR1 -	HIGH PRESSURE RECIRCULATION	1 of 2 SI/RHR trains delivering flow from the containment sump to 1 of 2 RCS cold legs, sump valve on operable recirc train open.	Manually align high pressure containment sump recirculation on low RWST level (may include re-start of RHR pump), align CCW cooling to RHR Hx, confirm operation of system.	20.5 hours
CHG -	CHARGING PUMP OPERATION	1 of 3 charging pumps continues to operate after initiating event or 1 of 3 charging pumps started within 30 minutes after reactor trip for RXCP seal injection.	If a charging pump continues to operate after initiating event, verify RXCP seal injection. Manually start at least 1 charging pump, if none operating, within 30 minutes after reactor trip and establish RXCP seal injection.	24 hours
CCW -	COMPONENT COOLING WATER	1 of 2 CCW pumps delivering flow to the RXCP thermal barrier.	Verify operation of system.	24 hours

3.1.2/3-9 Transients Without Main Feedwater

INITIATING EVENT - TRANSIENTS WITHOUT MAIN FEEDWATER (TRS)

This initiating event includes all transients (other than loss of offsite power and some special initiators) that result in total loss of main feedwater. It is assumed that main feedwater is not recoverable.

POWER AVAILABLE (OSP)

If offsite power is lost at any time during the transient, the emergency diesel generators (EDGs) are designed to automatically start and come up to speed within 10 seconds.[14] If the EDG fails to start the first time, several restarts will be attempted automatically. Operator action is then required for additional start attempts. Success for the OSP top event is to have emergency AC power (either from the EDGs or from offsite power) available to at least one of the two 4.16 kV emergency buses. If onsite power is lost, the failure to have AC power to these buses may be due to failure of the EDGs to start or to run, failure of the buses to shed loads, or failure of the EDGs to load. For a loss of offsite power with successful reactor trip, provision of emergency AC power could be delayed for as long as 30 minutes, i.e., a limiting time for SG secondary dryout. For event tree modeling, it is assumed that AC power is required for 24 hours. During this time, a fuel oil immersion pump is required to operate periodically to replenish the 850 gallon day tanks for the EDGs.

AUXILIARY FEEDWATER (AF3)

This node models the availability of the AFW system to remove decay heat. Success of the AFW system requires operation of at least one of three AFW pumps delivering a total flow of 200 gpm to at least one SG. The AFW pumps start on low SG levels or by manual actuation. The success criterion assumed here is the same as that used for the loss of normal feedwater event (plus several other transients) described in the Kewaunee USAR.[14]

Success for AF3 also assumes adequate steam relieving capability. This can be achieved by operation of steam dump to condenser, if available, or by operation of one of the relief valves (one SG PORV, five safety valves) for each active SG.

OPERATOR ACTION, BLEED AND FEED (OB2)

If AF3 fails, the operators are instructed to initiate primary system bleed and feed. Emergency operating procedure FR-H.1, Response to Loss of Secondary Cooling, instructs the operators to initiate bleed and feed if secondary cooling is lost and wide range steam generator level in either steam generator drops below 15% or pressurizer pressure increases above 2335 psig. The operators actuate SI and establish a RCS bleed path by opening at least one of two pressurizer PORVs. It is likely that bleed and feed cooling, according to FR-H.1 instructions, would be established by 30 minutes. SG dryout is expected soon after this time if both SGs are at the lo-lo level setpoint at the time of reactor trip. Under these conditions, it is conservatively estimated that the SGs boil dry at approximately 35 minutes.

Success of OB2 is 1 of 2 high pressure SI trains delivering flow to 1 of 2 RCS cold legs with 1 of 2 pressurizer PORVs open. Bleed and feed initiation prior to SG dryout with this success criterion is expected to result in effective decay heat removal. For simplicity, it is assumed that bleed and feed initiated by 30 minutes using one high pressure SI pump and one pressurizer PORV results in success.

It is assumed that failure of this node results in early core melt due to loss of all secondary cooling.

HIGH PRESSURE RECIRCULATION (HR1)

If OB2 is successful, long term cooling is addressed. Long term cooling is provided by sump recirculation. If RCS pressure remains above 140 psig, a low pressure safety injection train is lined up to take suction from the containment sump and discharge to the suction of the SI pumps via the residual heat exchangers. This lineup is referred as a SI/RHR train in the Emergency Operating Procedures.

Success of this HR1 is 1 of 2 SI/RHR trains providing flow to 1 of 2 RCS cold legs.

It is assumed that failure of this node results in late core melt.

CHARGING PUMP OPERATION (CHG)

The operator must maintain a minimum amount of charging flow to supply RXCP seal injection. One charging pump will provide adequate RXCP seal cooling and thereby prevent a small LOCA due to seal degradation following loss of all seal cooling. Continued post-trip operation of the charging pump plus operator training ensure that seal injection is maintained with little or no interruption following reactor trip. In addition, Step 4 of EOP ES-0.1, Reactor Trip Response, explicitly directs the operator to verify or establish charging flow. It is highly probable that Step 4 of EOP ES-0.1 is reached within 10 minutes following reactor trip since the EOP E-0 to EOP ES-0.1 transition occurs very quickly (via Step 4 of EOP E-0). Based on the expected RXCP seal response to the loss of all cooling described in Section 10.1 of WCAP-10541, Rev. 2[24], a normal seal flow requirement of 3 to 5 gpm per pump (less than 10 gpm total) is expected if seal injection is restored by 10 minutes, i.e., prior to the transient heatup phase. Even if seal injection is delayed until about 30 minutes, the seal leakage rate is expected to be less than 21 gpm per pump or 42 gpm total. This is still within the capacity of one of the 60 gpm positive-displacement charging pumps.

Based on the above description, success for the CHG top event requires either

1. continued operation of 1 of 3 charging pumps for seal injection, or
2. operator action (based on training or the EOPs) to start a charging pump for seal injection within 30 minutes following reactor trip.

COMPONENT COOLING WATER (CCW)

If CHG fails, RXCP seal injection is lost and seals are cooled by RCS water flowing past the thermal barrier which is cooled by CCW. Without this cooling, it is assumed that the RXCP seals fail, resulting in a small LOCA. It is conservatively assumed that core melt results from this LOCA.

FIGURE 3.1.2/3-8

TRANSIENTS WITHOUT MAIN FEEDWATER EVENT TREE

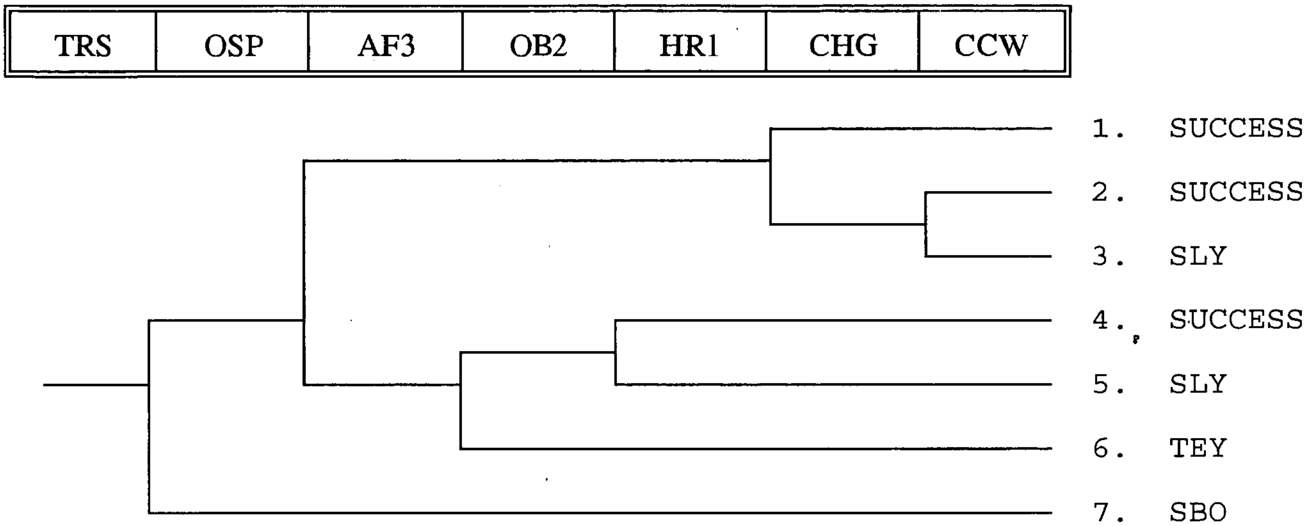


TABLE 3.1.2/3-8

SUCCESS CRITERIA FOR TRANSIENTS WITHOUT MAIN FEEDWATER

Top Event Description	System Success Criteria	Necessary Operator Actions	Mission Time (hrs)
OSP - POWER AVAILABLE	Emergency AC Power available to at least one of two 4.16 kV safeguards buses.	May have to manually start diesel.	24
AF3 - AUXILIARY FEEDWATER	1 of 3 AFW pumps delivering at least 200 GPM to at least 1 of 2 steam generators.	If AFW not automatically initiated by event, manually initiate AFW. Confirm operation of system.	24
OB2 - OPERATOR ACTION - BLEED AND FEED	1 of 2 high pressure SI trains delivering flow to 1 of 2 RCS cold legs; 1 of 2 pressurizer PORVs open (bleed and feed initiated prior to secondary dryout - assume at 30 minutes).	Manually open PORVs and block valves, start SI pumps (see FR-H.1).	Run for 24 hours
HR1 - HIGH PRESSURE RECIRCUL-ATION	1 of 2 SI/RHR trains delivering flow from the containment sump to 1 of 2 RCS cold legs, sump valve on operable recirc train open.	Manually align high pressure containment sump recirculation on low RWST level (may include re-start of RHR pump), align CCW to RHR Hx, confirm operation of system.	20.5

TABLE 3.1.2/3-8

SUCCESS CRITERIA FOR TRANSIENTS WITHOUT MAIN FEEDWATER (Continued)

Top Event Description	System Success Criteria	Necessary Operator Actions	Mission Time (hrs)
CHG - CHARGING PUMP OPERATION	1 of 3 charging pumps continues to operate after initiating event or 1 of 3 charging pumps started within 30 minutes after reactor trip for RXCP seal injection	If a charging pump continues to operate after initiating event, verify RXCP seal injection. Manually start at least 1 charging pump, if none operating, within 30 minutes after reactor trip and establish RXCP seal injection.	24 hours
CCW - COMPONENT COOLING WATER	1 of 2 CCW pumps delivering flow to the RXCP thermal barrier.	Verify operation of system.	24 hours

3.1.2/3.10 Large Steam Line/Feed Line Break

INITIATING EVENT - LARGE STEAM LINE/FEED LINE BREAK (SLB)

This initiator covers large steam line breaks and large feed line breaks.

REACTOR POWER ABOVE 10% (PWR)

The reactor cooldown that occurs as a result of a SLB is more severe at low powers due to the greater stored energy in the SGs. At 10% power, a return to criticality cannot occur even if both steam generators blow down and insert the maximum positive reactivity expected during a fuel cycle.[25]

Success of PWR requires that the reactor is operating at greater than 10% power when the steam line break occurs.

RCS BORATION WITH BAT (HI3)

High pressure SI (HI3) pumps and the boric acid tank (BAT) provide concentrated boric acid to the primary system to add negative reactivity to ensure core shutdown. Safety injection is automatically actuated on an SI signal, which is generated by low steam line pressure, low primary pressure, or high containment pressure.

Success of HI3 is 1 of 2 SI pumps injecting the contents of one BAT into 1 of 2 RCS cold legs. This success is independent of whether the most reactive control rod sticks out following reactor trip.

MAIN STEAM ISOLATION (IS1)

Main steam and feedwater isolation (IS1) are necessary to stop the cooldown. To allow for an arbitrary break location, success of IS1 requires closure of both of the two MSIVs and isolation of both feedwater lines. Operator action is not required for IS1 success.

AUXILIARY FEEDWATER (AF1)

Auxiliary feedwater (AF1) is required to remove decay heat. AFW is actuated on lo-lo steam generator level or by the SI signal. Because of KNPP's design, only 2 AFW pumps, 1 motor driven and the turbine driven, can provide flow to 1 SG once the other SG is isolated. Therefore, once the faulted SG is isolated, only 2 AFW pumps will be available.

Secondary cooling success also requires dumping steam from the intact SG. Since steam dump to condenser is not readily available after the MSIVs close, the operator uses the SG PORV on the intact SG. The secondary safety valves (5 per SG) are also available for steam relief if the PORV can not be operated.

It is assumed that secondary cooling is required for the entire event. Success of AF1 is 1 of 2 AFW pumps supplying at least 200 gpm to the non-faulted steam generator for the entire event.

OPERATOR ACTION - ESTABLISH MAIN FEEDWATER (OM1)

If AF1 fails, main feedwater (OM1) is used for secondary cooling. Emergency Operating Procedure FR-H.1, Response to Loss of Secondary Heat Sink, is entered via the critical safety function trees. FR-H.1 instructs the operators to attempt to restore AFW and to line up MFW to provide secondary cooling. If MFW is unavailable, the operators are instructed to depressurize the primary system in order to block SI and then to depressurize an intact steam generator in order to provide secondary coolant flow from the condensate system. However, due to the complexity of the operator actions required to establish flow to a steam generator from the condensate system alone, it would take a significant amount of time to establish this flow and it is not included in the event tree modeling.

Success of OM1 is 1 of 2 MFW trains delivering at least 200 gpm flow to the non-faulted steam generator for the entire event.

OPERATOR ACTION - BLEED AND FEED (OB4)

If AF1 and MF1 fail, the operators are instructed to initiate primary system bleed and feed. Emergency operating procedure FR-H.1, Response to Loss of Secondary Cooling, instructs the operators to initiate bleed and feed if secondary cooling is lost and wide range SG level in either SG drops below 15% or pressurizer pressure increases above 2335 psig. The operators use the SI pumps for injection and establish an RCS bleed path by opening at least one of two pressurizer PORVs. Bleed and feed initiated with EOP FR-H.1 is performed at a comparatively early time if all SGs loose some of their inventory prior to MSIV closure. Most likely, the loss of heat sink symptom is not reached early, and bleed and feed is not performed until after the RCS and intact SG heat up to no-load and additional inventory is boiled from the intact SG. Since the residual heat is not event dependent (precluding any significant nuclear heat due to a return to criticality) and since the RCS gains additional inventory and heat capacity due to addition of cold SI water, the intact SG dryout times for the secondary break transients exceed the one hour SG dryout time previously noted for the other transient events. Thus, it is appropriate to apply the same success criterion for the steamline break as the other transient cases.

Success of OB4 is 1 of 2 high pressure safety injection trains delivering flow from the RWST to 1 of 2 RCS cold legs with 1 of 2 pressurizer PORVs open. Bleed and feed initiation prior to intact SG dryout with this success criterion is expected to result in effective decay heat removal. For simplicity, it is assumed that bleed and feed initiated by 30 minutes results in successful recovery.

With a secondary heat sink available, the only safety injection required following a large secondary side break is the contents of a BAT. However, with no secondary heat sink available, safety injection from the RWST is required for a successful bleed and feed recovery. Top event OB4 includes the automatic transition of the suction of the SI pumps from the BAT to the RWST. The SI pumps are running since the Emergency Operating Procedures direct the operator to leave them running if a secondary heat sink is unavailable. Thus, no manual action by the operator is required for top event OB4 to either transfer the SI pumps' suction to the RWST or to start the SI pumps. He is required to manually open at least one pressurizer PORV to provide an RCS bleed path.

It is assumed that failure of this node results in early core melt due to loss of all secondary cooling.

HIGH PRESSURE RECIRCULATION (HR1)

If OB4 is successful, long term cooling is addressed. Long term cooling is provided by sump recirculation. If RCS pressure remains above 140 psig, a low pressure SI train is lined up to take suction from the containment sump and discharge to the suction of the high pressure SI pumps via the residual heat exchangers. This lineup is referred as a SI/RHR train in the Emergency Operating Procedures.

Success of HR1 requires one SI/RHR train to provide flow to one of two RCS cold legs.

It is assumed that failure of this node results in late core melt.

FIGURE 3.1.2/3-9

STEAM LINE BREAK EVENT TREE

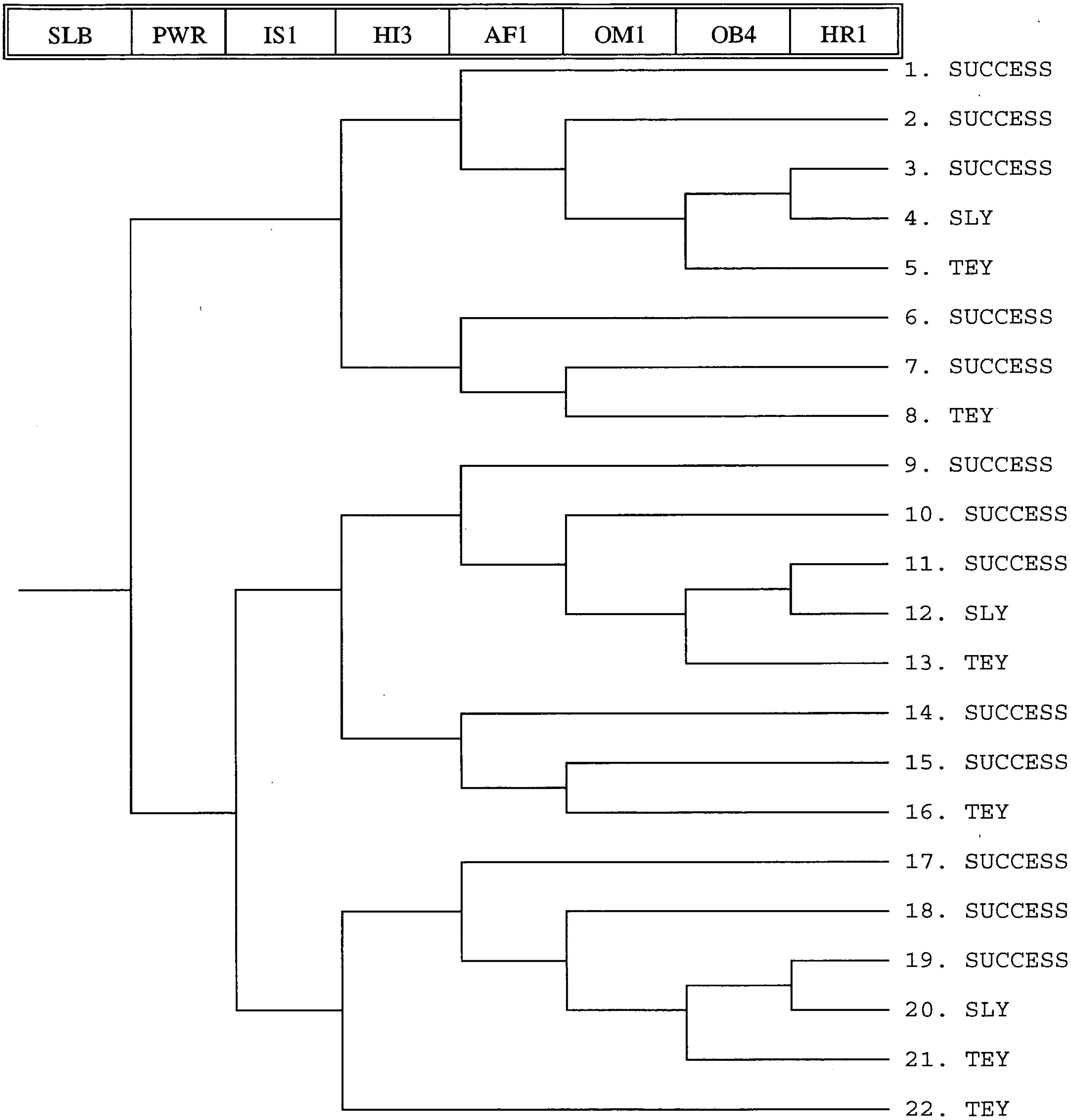


TABLE 3.1.2/3-9

SUCCESS CRITERIA FOR LARGE STEAMLINE/FEEDLINE BREAK

Top Event Description		System Success Criteria	Necessary Operator Actions	Mission Time (hrs)
HI3 -	RCS BORATION WITH BAT	1 of 2 high pressure SI trains inject the contents of one BAT into 1 of 2 RCS cold legs.	Confirm operation of system.	3.5
PWR -	REACTOR POWER ABOVE 10%	Reactor operating at or above 10% power.	None	None
IS1 -	MAINSTEAM ISOLATION	Isolation of faulted SG - closure of any 2 of 2 MSIVs and 2 of 2 feedwater lines.	Verify line isolation	None
AF1 -	AUXILIARY FEEDWATER	1 of 2 AFW pumps delivering at least 200 GPM to intact steam generator.	Confirm operation of system.	24
OM1 -	OPERATOR ACTION - ESTABLISH MAIN FEEDWATER	1 of 2 MFW trains delivering at least 200 GPM to intact generator.	Manually align and initiate MFW. Confirm operation of system.	24

TABLE 3.1.2/3-9

SUCCESS CRITERIA FOR LARGE STEAMLINE/FEEDLINE BREAK (Continued)

Top Event Description	System Success Criteria	Necessary Operator Actions	Mission Time (hrs)
OB4 - OPERATOR ACTION - BLEED AND FEED	1 of 2 high pressure SI trains delivering flow to 1 of 2 RCS cold legs; 1 of 2 pressurizer PORVs open (bleed and feed initiated prior to secondary dryout - assume at 30 minutes).	Manually open PORVs and block valves, verify SI pumps running (see FR-H.1) with suction aligned to RWST	Run for 24 hours
HR1 - HIGH PRESSURE RECIRCUL-ATION	1 of 2 SI/RHR trains delivering flow from the containment sump to 1 of 2 RCS cold legs, sump valve on operable recirc train open.	Manually align high pressure containment sump recirculation on low RWST level (may include re-start of RHR pumps), align CCW to RHR Hx, confirm operation of system.	20.5

3.1.2/3.11 Loss of Offsite Power

INITIATING EVENT - LOSS OF OFFSITE POWER (LSP)

The initiator is a Loss of Offsite Power.

POWER AVAILABLE (OSP)

Upon an LSP, the emergency diesel generators (EDGs) are designed to automatically start and come up to speed within 10 seconds.[14] If the EDG fails to start the first time, several restarts will be attempted automatically. Operator action is then required for additional start attempts. Success for the OSP top event is to have emergency AC power available to at least one of the two 4.16 kV emergency buses. The failure to have AC power to these buses may be due to failure of the EDGs to start or to run, failure of the buses to shed loads, or failure of the EDGs to load. For a loss of offsite power event with successful reactor trip, provision of emergency AC power could be delayed for as long as 30 minutes, i.e., a limiting time for SG secondary dryout. For event tree modeling, it is assumed that onsite emergency AC power is required for 24 hours. During this time, a fuel oil transfer pump is required to operate periodically to replenish the 850 gallon day tanks for the EDGs.

AUXILIARY FEEDWATER (AF3)

Auxiliary Feedwater (AF3) starts in response to an automatic actuation signal or in response to operator action. The motor driven pumps start on a blackout signal or from both FW pump breakers open. The turbine driven AFW pump starts on a bus 1 and 2 undervoltage signal. The pumps could also be started by manual actuation. Success of AF3 is 1 of 3 AFW pumps delivering at least 200 GPM flow to 1 of 2 SGs for 24 hours. The availability of the different AFW pumps and their trains is dependent on the availability of power to the vital buses from the diesel generators. This availability is modeled in the AF3 fault tree.

OPERATOR ACTION - BLEED AND FEED (OB5)

If AF3 fails, the operators are instructed to initiate primary system bleed and feed (OB2). Emergency operating procedure FR-H.1, Response to Loss of Secondary Cooling, instructs the operators to initiate bleed and feed if secondary cooling is lost and wide range steam generator level in either SG drops below 15% or pressurizer pressure increases above 2335 psig. The operators would actuate SI and establish a RCS bleed path by opening at least one of two pressurizer PORVs. Success of OB2 is dependent on the availability of emergency onsite power supplies to the vital buses. It is likely that bleed and feed cooling would be established according to FR-H.1 instructions by 30 minutes. SG secondary dryout is expected at approximately one hour.

Success of OB5 is 1 of 2 high pressure SI trains delivering flow to 1 of 2 RCS cold legs with 1 of 2 pressurizer PORVs open. Bleed and feed initiation prior to SG dryout with this success criterion is expected to result in effective decay heat removal. For simplicity, it is assumed that bleed and feed initiated by 30 minutes using one high pressure SI pump and one pressurizer PORV will result in success.

It is assumed that failure of this node results in early core melt due to loss of all secondary cooling.

HIGH PRESSURE RECIRCULATION (HR1)

If OB5 is successful, long term cooling (HR1) is addressed. Long term cooling is provided by sump recirculation. If RCS pressure remains above 140 psig, a low pressure SI train is lined up to take suction from the containment sump and discharge to the suction of the high pressure SI pumps via the residual heat exchangers. This line up is referred as an SI/RHR train in the Emergency Operating Procedures. Success of HR1 is dependent on the availability of emergency onsite power supplies to the vital buses.

Success of HR1 is 1 of 2 SI/RHR trains providing flow to 1 of 2 RCS cold legs.

It is assumed that failure of this node results in late core melt.

CHARGING PUMP OPERATION (CHG)

The operator must maintain a minimum amount of charging flow to supply RXCP seal injection. One charging pump will provide adequate RXCP seal cooling and thereby prevent a small LOCA due to seal degradation following loss of all seal cooling. Continued post-trip operation of the charging pump plus operator training ensure that seal injection is maintained with little or no interruption following reactor trip. In addition, Step 4 of EOP ES-0.1, Reactor Trip Response, explicitly directs the operator to verify or establish charging flow. It is highly probable that Step 4 of EOP ES-0.1 is reached within 10 minutes following reactor trip since the EOP E-0 to EOP ES-0.1 transition occurs very quickly (via Step 4 of EOP E-0). Based on the expected RXCP seal response to the loss of all cooling described in Section 10.1 of WCAP-10541, Rev. 2[24], a normal seal flow requirement of 3 to 5 gpm per pump (less than 10 gpm total) is expected if seal injection is restored by 10 minutes, i.e., prior to the transient heatup phase. Even if seal injection is delayed until about 30 minutes, the seal leakage rate is expected to be less than 21 gpm per pump or 42 gpm total. This is within the capacity of one of the 60 gpm positive-displacement charging pumps.

Based on the above description, success for the CHG top event requires either

1. continued operation of 1 of 3 charging pumps for seal injection, or

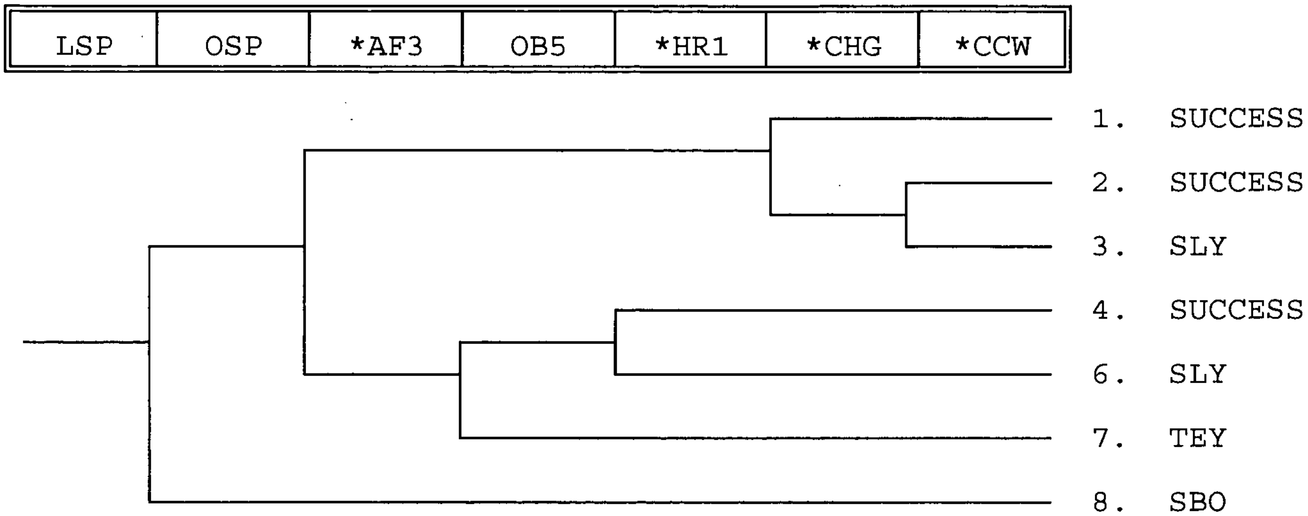
2. operator action (based on training or the EOPs) to start a charging pump for seal injection within 30 minutes following reactor trip.

COMPONENT COOLING WATER (CCW)

If CHG fails, RXCP seal injection is lost and seals are cooled by RCS water flowing past the thermal barrier, which is cooled by CCW. Without this cooling, it is assumed that the RXCP seals fail, resulting in a small LOCA. It is conservatively assumed that core melt results from this LOCA.

FIGURE 3.1.2/3-10

LOSS OF OFFSITE POWER EVENT TREE



*These nodes are used in other event trees, they are conditional here due to the initiator.

TABLE 3.1.2/3-10

SUCCESS CRITERIA FOR LOSS OF OFFSITE POWER

Top Event Description	System Success Criteria	Necessary Operator Actions	Mission Time (hrs)
OSP - POWER AVAILABLE	AC Power available to at least one of two 4.16 kV safeguards buses.	May have to manually start diesel.	24
AF3 - AUXILIARY FEEDWATER	1 of 3 AFW pumps delivering at at least 200 GPM to at least 1 of 2 steam generators.	If AFW not automatically initiated by event, manually initiate AFW. Confirm operation of system.	24
OB5 - OPERATOR ACTION - BLEED AND FEED	1 of 2 high pressure SI trains delivering flow to 1 of 2 RCS cold legs, 1 of 2 pressurizer PORVs open (bleed and feed initiated prior to secondary dryout - assume at 30 minutes).	Manually open PORVs and block valves, start SI pumps (FR-H.1).	Run for 24 hours
HR1 - HIGH PRESSURE RECIRCUL-ATION	1 of 2 SI/RHR trains delivering flow from containment sump to 1 of 2 RCS cold legs, sump valve on operable recirculation train open.	Manually align high pressure containment sump recirculation on low RWST level (may include re-start of RHR pump), align CCW to residual Hx, confirm operation of system.	20.5

TABLE 3.1.2/3-10

SUCCESS CRITERIA FOR LOSS OF OFFSITE POWER (Continued)

Top Event Description	System Success Criteria	Necessary Operator Actions	Mission Time (hrs)
CHG - CHARGING PUMP OPERATION	1 of 3 charging pumps continues to operate after initiating event or 1 of 3 charging pumps started within 30 minutes after reactor trip for RXCP seal injection	If a charging pump continues to operate after initiating event, verify RXCP seal injection. Manually start at least 1 charging pump, if none operating, within 30 minutes after reactor trip and establish RXCP seal injection.	24 hours
CCW - COMPONENT COOLING WATER	1 of 2 CCW pumps delivering flow to the RXCP thermal barrier.	Verify operation of system.	24 hours

3.1.2/3.12 Station Blackout

INITIATING EVENT - STATION BLACKOUT (SBO)

The SBO event tree is entered via transitions from the TRA, TRS, and LSP event tree. The SBO event starts out as a transient, with grid power eventually lost, or as a LSP event initiated by a loss of grid power from the high voltage distribution lines serving the station from the switchyard. In addition, the SBO event includes failure of onsite AC power from the emergency diesel generators.

CHARGING PUMP OPERATION (CHB)

With loss of CCW cooling to the RXCP thermal barrier, the operator must maintain a minimum amount of charging flow to supply RXCP seal injection. One charging pump will provide adequate RXCP seal cooling and thereby prevent a small LOCA due to seal degradation following loss of all seal cooling. Continued post-trip operation of the charging pump plus operator training ensure that seal injection is maintained with little or no interruption following reactor trip. In the case of an SBO, the operators are instructed to strip bus 52 of all loads, start the TSC diesel generator, power bus 52 through bus 46, and start one of the two charging pumps (A or C) that are powered from bus 52. Based on the expected RXCP seal response to the loss of all cooling described in WCAP-10541, Rev. 2[24], a normal seal flow requirement of 3 to 5 gpm per pump (less than 10 gpm total) is expected if seal injection is restored by 10 minutes, i.e., prior to the transient heatup phase. Even if seal injection is delayed until about 30 minutes, the seal leakage rate is expected to be less than 21 gpm per pump or 42 gpm total. This is within the capacity of one of the 60 gpm positive-displacement charging pumps.

Based on the above description, success of this node is 1 of 2 charging pumps supplying the minimum flow needed for seal injection.

TURBINE DRIVEN AUXILIARY FEEDWATER PUMP (AF2)

Auxiliary feedwater (AF2) is required to remove decay heat from the RCS. Only the turbine-driven AFW pump is available if all AC power is lost. For this event, the turbine-driven AFW pump is actuated by one of the following signals:

1. Loss of power to electrical buses 1 and 2

2. Lo-lo SG level in both steam generators

3. Manual actuation

Success of the turbine-driven AFW pump requires that it supply flow of at least 200 gpm to at least 1 SG. Steam relief from one of the active SGs is also implied, either via the SG PORV or one of the five safety valves on each SG. This is consistent with the requirements for the transient and LSP events described in the previous sections.

POWER RESTORED IN 2 HOURS (AC2)

Analyses for Kewaunee show that, if the turbine-driven AFW pump fails to start, power must be restored in 2 hours or less to prevent significant core damage. If AC power is recovered at this time, SI is required to restore RCS inventory in order to prevent core damage. Success of AC2 is restoration of AC power in 2 hours. If AC power is not recovered within two hours after the turbine-driven pump fails, core damage is assumed.

OPERATOR ACTION - RCS COOLDOWN (OCD)

Emergency Operating Procedure ECA-0.0, Loss of All AC Power, instructs the operator to depressurize the intact steam generators to 300 psig by locally dumping steam at the maximum rate using the steam generator PORVs. By establishing AFW to and using the associated SG PORV for one or both of the steam generators, it is possible to cool the RCS to around 410°F within a one hour time period. By reducing the temperature to 410°F or less, the RXCP seal leak rate is significantly reduced because the cooldown results in an RCS depressurization that causes most of the contents of the accumulators to be injected.

POWER RESTORED IN X HOURS (ACX)

This top event is the probability that AC power is restored in X hours, where X depends on the success or failure of OCD (the RCS cooldown). As explained previously, if the cooldown is not successful, power must be restored within two hours after auxiliary feedwater is lost. If the cooldown is successful, power recovery could be delayed until approximately 3 hours after the safeguards batteries (and auxiliary feedwater) are assumed to be lost at 8 hours. Analyses on similar plants have shown a successful RCS cooldown delays core uncovery for 2 hours following SG dryout.[24] Analysis for Kewaunee has shown that SG dryout occurs around one hour following termination of auxiliary feedwater. Thus, the AC power restoration times correspond roughly with the times to secondary dryout, when recovery is still possible using one SI pump. These times are as follows:

> If the cooldown is successful, AFW continues for 8 hours, and power is restored at any time before X=11 hours, core uncovery can be averted (assuming also that the RXCP seal leakage is not extensive).

If the cooldown is not successful, AFW continues for 8 hours, and power is restored at any time before X=9 hours, core uncovery can be averted (assuming also that the RCP seal leakage is not extensive).

In the above cases, operator restoration of RCS inventory by initiation of SI is required to prevent core damage. Also, there is still the possibility that the core has uncovered even if power is restored within the above time limits (see CCV node description below). If power is not recovered within the above time limits, core damage is assumed.

CORE COVERED (CCV)

This top event addresses core uncovery due to an RXCP seal LOCA after power is restored. The event tree failure path is the probability of core uncovery resulting from RXCP seal leakage. If core uncovery has not occurred, there is still a small amount of RXCP seal leakage (about 21 gpm per pump), so some RCS makeup is still required. Therefore, for the event tree success paths for CCV, there is a small LOCA and RCS makeup is required.

OPERATOR ACTION - RESTORE RCS INVENTORY (ORI)

When power is restored, the Emergency Operating Procedure, ECA-0.2, Loss of All AC Power Recovery With SI Required, instructs the operator to restore the safeguard systems. SI to restore RCS inventory is required and decay heat removal must be established or maintained. The operator actions and systems required depend on the postulated accident progression at the time that power is restored. To keep the analysis for ORI manageable, it is sufficient to model the basic actions and systems used in the recovery for a small LOCA.

If core uncovery does not occur (success of fault tree CCV) and RXCP seal leakage is assessed as a small LOCA, high pressure SI with 1 out of 2 SI pumps is required to mitigate the event. The operator actions needed to restore the safeguard systems also include operation of at least one AFW pump with injection to at least one SG (otherwise bleed and feed recovery is used; at least one pressurizer PORV is opened for the bleed and feed contingency actions). Based on prior success for AF2, a secondary heat sink is still available (or can be quickly restored) at various times during the accident (ACX), so the success criterion is consistent with that required for small LOCA or bleed and feed recovery.

HIGH PRESSURE RECIRCULATION (HR1)

High pressure recirculation (HR1) provides two necessary functions. The first function is to provide a source of water from the containment sump to ensure that the core remains covered with water after SI depletes the water in the RWST. The second function is to provide long term post-accident cooldown and decay heat removal after the SI phase. If SI (ORI) is successful, the high pressure SI pumps taking suction from the RWST eventually deplete its inventory. The time of the injection phase depends on the number of pumps injecting.

The same success criterion used for small LOCA will be assumed here for SBO. This criterion is 1 of 2 RHR pumps supplying flow to the suction of 1 of 2 SI pumps, injecting flow to 1 of 2 cold legs. CCW and SWS support conditions are also necessary for recirculation.

FIGURE 3.1.2/3-11

STATION BLACKOUT EVENT TREE

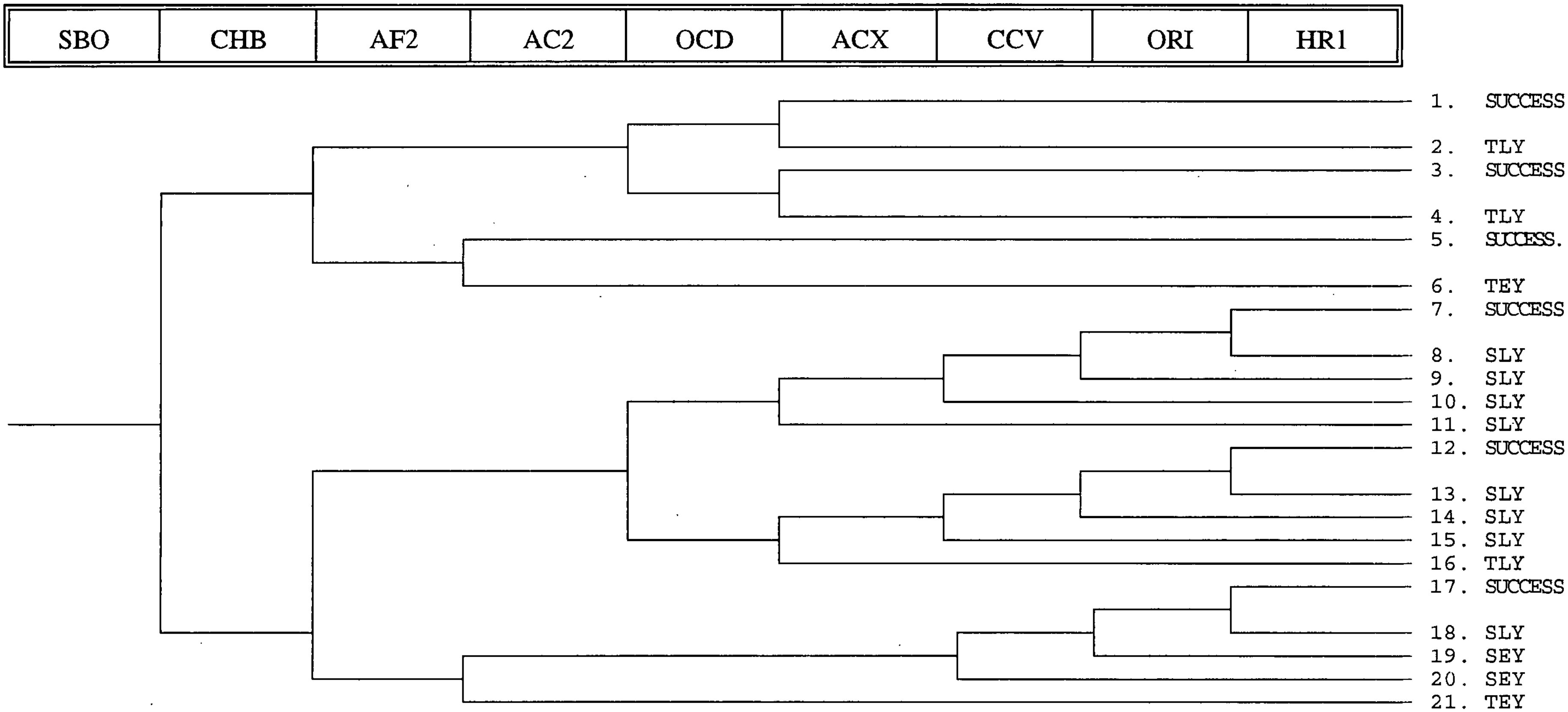


TABLE 3.1.2/3-11

SUCCESS CRITERIA FOR STATION BLACKOUT

Top Event Description	System Success Criteria	Necessary Operator Actions	Mission Time (hrs)
CHB - CHARGING PUMP OPERATION	1 of 2 charging pumps supplying flow for seal injection.	Strip all loads from bus 52, start TSC diesel, align to bus 46, align bus 46 to bus 52, start 1 of 2 charging pumps.	Run for 24 hours.
AF2 - TURBINE DRIVEN AUXILIARY FEEDWATER PUMP	Turbine driven AFW pump supplying at least 200 gpm to at least 1 of 2 SGs with associated valves and CST available.	Confirm operation of system.	Run for 8 hours.
AC2 - POWER RESTORED IN 2 HOURS	Power Restored to 1 of 2 vital buses within 2 hours.	As appropriate.	
OCD - OPERATOR ACTION-RCS COOLDOWN	1 of 2 SG PORVs available for local operation, at least 1 of 2 SGs supplied by auxiliary feedwater flow.	Locally depressurize SGs using SG PORVs to 300 psig at maximum rate while maintaining 4% narrow range SG level. Cooldown performed per ECA-0.0.	Potentially 24 hours (steam relief needed for decay heat removal).

TABLE 3.1.2/3-11

SUCCESS CRITERIA FOR STATION BLACKOUT (Continued)

Top Event Description	System Success Criteria	Necessary Operator Actions	Mission Time (hrs)
ACX - POWER RESTORED IN X HOURS			
(OCD is successful)	Power restored to 1 of 2 vital buses within 11 hours.	As appropriate.	None
(OCD is unsuccessful)	Power restored to 1 of 2 vital buses within 9 hours.	As appropriate.	None
CCV - CORE COVERED	No equipment success criteria. Success of this node is that the core is covered at the time power is restored.	None	None

TABLE 3.1.2/3-11

SUCCESS CRITERIA FOR STATION BLACKOUT (Continued)

Top Event Description	System Success Criteria	Necessary Operator Actions	Mission Time (hrs)
ORI - OPERATOR ACTION - RESTORE RCS INVENTORY	Restore safeguards systems and initiate high pressure SI flow to RCS cold legs. Success requires one of the following combinations: 1. HI2 plus AF0, or 2. HI2 plus OB1 (bleed and feed needed if there is no feedwater)	Restore safeguards pumps and cooling, start SI pumps, operate AFW pumps or manually open pressurizer PORVs and block valves as required.	24
HR1 - HIGH PRESSURE RECIRC-ULATION	1 of 2 RHR/SI trains delivering flow from the containment sump to 1 of 2 RCS cold legs, pump valve on operable recirc train open.	Manually align high pressure containment sump recirculation on low RWST level (may include re-start of RHR pump), align CCW to residual Hx, confirm operation of system.	20.5

3.1.2/3.13 ATWS Without Main Feedwater

INITIATING EVENT - ATWS WITHOUT MAIN FEEDWATER (AWS)

This initiator covers all transients that generate a reactor trip signal by the reactor protection system (RPS) and for which the automatic reactor trip signal does not result in reactor trip. It is conservatively assumed that MFW is not available following the initiator and AWS.

AUTOMATIC REACTOR TRIP FAILURE MODE (AFM)

Failure of the RPS logic, reactor trip breakers, or control rods to move will cause the failure of the RPS to trip the reactor. If the reactor is not tripped, plant procedures instruct the operator to manually trip the reactor. The manual trip function operates through a shunt coil to the trip breakers. If the reactor trip unavailability is not related to the trip breakers or control rods, then manual scram through the shunt coil is possible. Reliability of the RPS has been assessed by the Westinghouse Owners Group(26) and accepted by the NRC. Failure of both the undervoltage coil and the shunt coil are included in the unavailability of the automatic reactor trip signal.

Top event AFM denotes the means of failure of the automatic reactor trip. The "success" path is failure of the automatic trip signal (i.e., RPS) logic. The "failure" path is failure of the control rods to more. The "intermediate" path for AFM is failure of the reactor trip breakers.

MANUAL REACTOR TRIP (MRT)

Top event MRT models the manual trip of the reactor by the operators. If the failure of the automatic reactor trip function is due to failure of the RPS logic, the operator may be successful at manually tripping the reactor using 1 of 2 pushbuttons. Success for this event tree node means that an ATWS has not occurred.

OPERATOR ACTIONS TO DEENERGIZE RDMG SETS (ORT)

Following an AWS, the operators are instructed by procedure to first manually trip the reactor. The failure of this action is accounted for in the MRT event tree node. If a manual trip is unsuccessful the operators are instructed to manually insert the control rods, manually open the supply breakers to the buses supplying the control rod drive MG sets, locally trip the reactor trip breakers, locally open the MG set supply breakers, and verify turbine trip and AFW pumps running. Due to the length of time required to achieve reactor trip from manually inserting the control rods, this method of tripping the reactor is considered in top event LTS.

Success for this event tree node requires the reactor to be tripped by the operators prior to steam generator dryout (within 2 minutes for an AWS at full power). This is accomplished manually by opening the supply breakers for buses 33 and 43 (which deenergizes the MG sets). The reactor can be tripped locally within 2 minutes by either opening 1 of 2 of the reactor trip breakers or by opening both of the MG set supply breakers.

Success of ORT will preclude the need for AMSAC to function to mitigate the transient.

ATWS MITIGATING SYSTEM ACTUATION CIRCUITRY (AMS)

If the reactor fails to scram, AMS provides two functions: turbine trip and auxiliary feedwater flow actuation. Tripping the turbine early in an AWS loss of feedwater event causes a rapid reduction in steam flow out of the steam generator and a resultant rapid increase in steam pressure to the steam line safety valve set pressure. A turbine trip extends SG inventory and results in an increase in core coolant temperature. The increase in coolant temperature causes a decrease in core power early in the transient before SG tubes begin to uncover. Later, as steam generator tubes begin to uncover, the rate of increase in RCS temperature (and peak RCS pressure) is lower because it started at a lower core power level. AMSAC is actuated at 13% narrow range SG level on 3/4 level channels. AFW actuation and turbine trip occur 25 seconds after the AMSAC actuation signal. Normally the RPS would actuate turbine trip and AFW flow before the conditions that cause AMSAC actuation are reached. However, if a common mode failure in the RPS were to fail to initiate auxiliary feedwater flow or turbine trip in addition to prohibiting a reactor trip, then AMSAC is an alternative method of providing AFW flow and turbine trip.

As stated in the AFM node description, the RPS failures are dominated by reactor trip breaker failures rather than actuation logic, so that RPS failures do not significantly effect actuation of AFW or turbine trip. However, in this evaluation, no credit is given for automatic or manual turbine trip actuation of AFW by the low SG level signal, or for manual actuation of AFW. For this study, it is assumed that if AMS fails when required to mitigate the transient, core melt will occur.

AUXILIARY FEEDWATER (AFG)

This event tree node is entered if operator actions to trip the reactor fail, but AMS is successful in starting AFW and tripping the turbine.

Success of this node is two out of three AFW pumps supplying at least 400 GPM to one out of two SGs.

PRESSURE RELIEF (PPR)

This event tree node addresses the probability that pressurizer pressure relief capacity is adequate to prevent a peak RCS pressure in excess of 3200 psig. 3200 psig is the maximum RCS pressure limit for Westinghouse plants corresponding to the ASME Boiler and Pressure Vessel Code Level C service limit stress criterion.[27] Early core damage is assumed to occur if this pressure limit is exceeded.

An unfavorable exposure time (UET) is defined as the time during the fuel cycle life when the reactivity feedback is not sufficient to limit the RCS pressure for ATWS to less than 3200 psig for a given plant configuration (power level, manual rod insertion, auxiliary feedwater flow, and pressurizer PORV availability).[28] Although the PORVs may be blocked for part of the cycle life, the pressurizer safety valves are assumed to be available through the cycle life. Success for this top event require both safety valves to be operable.

To determine the success criterion of top event PPR for Kewaunee, an evaluation of the analysis described in Appendix B of Reference 3 was performed. From this evaluation, a success criterion of 2 safeties and 1 PORV available for RCS pressure relief is bounding for all but the first 40 days of the fuel cycle if top event AFG is successful (i.e., if 2 out of 3 AFW pumps are delivering at least 400 gpm to the SGs). Thus, this success criterion bounds about 90% of the days in the 12 month fuel cycle of Kewaunee. It should also be pointed out that the pressure requirements in the Reference 28 analysis are based on a "worst case" initiating event, i.e., an ATWS with loss of load and concurrent loss of main feedwater. In view of this, the success criterion is judged to be adequate for the purpose of event tree modeling.

LONG TERM SHUTDOWN (LTS)

If automatic trip, manual trip, or manual/local opening of the breakers for the rod drive MG sets does not shut down the reactor early in the transient and the peak RCS pressure does not exceed the stress criterion within the first few minutes of the transient, then alternate means to achieve subcriticality and maintain the shutdown condition are available. The emergency operating procedure FR-S.1 instructs the operator to begin manual rod insertion. Boration of the RCS is also initiated with the charging pumps in emergency boration via the boric acid tank.

If there is no mechanical failure associated with the control rods, the operators are able to manually insert the control rod banks to achieve long term shutdown. Successful manual rod insertion is adequate to ensure long term shutdown at the hot zero power condition. Analysis shows that reactor shutdown can be achieved within 20 minutes by manual insertion of the control rods.[29]

If the control rods can not be inserted, emergency boration can be used to achieve long term shutdown. The limiting boration time is estimated by assuming the RCS boron concentration must be increased from full power beginning-of-life equilibrium xenon conditions to one percent, xenon free shutdown conditions with two boric acid transfer pumps taking suction from the boric acid tank and feeding directly to the suction of two charging pumps, the one percent shutdown boron concentration is achieved within 15 minutes.[29]

If long term shutdown fails, it is assumed core damage occurs.

FIGURE 3.1.2/3-12

ATWS EVENT TREE

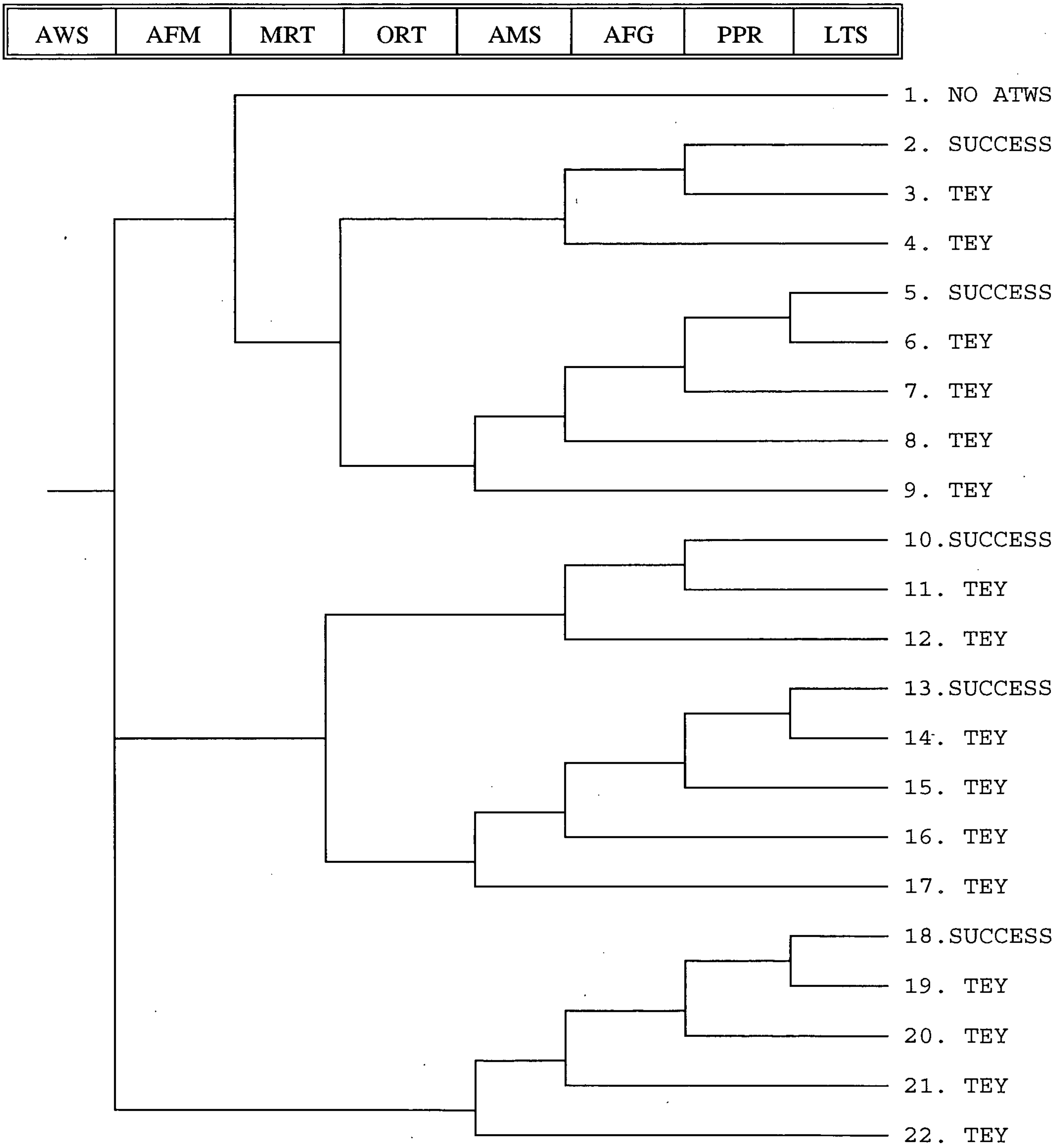


TABLE 3.1.2/3-12

SUCCESS CRITERIA FOR ATWS WITHOUT MAIN FEEDWATER (AWS) EVENT TREE

Top Event Description	System Success Criteria	Necessary Operator Actions	Mission Time (hrs)
AFM - AUTOMATIC REACTOR TRIP FAILURE MODE	Failure of RPS logic	None	None
MRT - MANUAL REACTOR TRIP	Operator action to manually trip the reactor.	Manually Trip the Reactor.	None
ORT - OPERATOR ACTIONS TO TRIP REACTOR	2 of 2 480V bus supply breakers opened within 2 minutes or locally trip the reactor within 2 minutes.	Manually open the supply breakers to deenergize the rod drive MG sets or locally trip the reactor (per FR-S.1).	None
AFG - AUXILIARY FEEDWATER	2 of 3 AFW pumps supplying at least 400 GPM to at least 1 of 2 SGs.	Verify AFW pumps running. Control SG levels and provide steam relief.	Run for 24 hours

TABLE 3.1.2/3-12

SUCCESS CRITERIA FOR ATWS WITHOUT MAIN FEEDWATER (AWS) EVENT TREE (Continued)

Top Event Description		System Success Criteria	Necessary Operator Actions	Mission Time (hrs)
PPR -	PRIMARY PRESSURE RELIEF	1 of 2 pressurizer PORVs and 2 of 2 pressurizer safety valves open.	None	None
LTS -	LONG TERM SHUTDOWN	Either of two conditions:		
		1) Manual insertion of control rods to achieve reactor shutdown within 20 minutes.	1) Manually insert control rods,	1) None
		2) Emergency boration to 1611 ppm at 80 gpm in 15 minutes using 2 charging pumps and 2 BA transfer pumps.	2) Align BAT and BA transfer pumps, operate charging pumps in emergency boration mode.	2) Run for 15 minutes
AMS -	ATWS MITIGATING CIRCUITRY (AMSAC)	Signal to start AFW and trip the turbine.	Confirm operation of system.	None.

3.1.2/3.14 Loss of Service Water System

INITIATING EVENT - LOSS OF SERVICE WATER SYSTEM (SWS)

This initiator covers a loss of the SWS which causes loss of ECCS injection, recirculation, instrument air, residual heat exchangers, and RXCPs.

CHARGING PUMP OPERATION (CHS)

With loss of CCW cooling to the RXCP thermal barrier, the operator must maintain a minimum amount of charging flow to supply RXCP seal injection. One charging pump will provide adequate RXCP seal cooling and thereby prevent a small LOCA due to seal degradation following loss of all seal cooling. Continued post-trip operation of the charging pump plus operator training ensure that seal injection is maintained with little or no interruption following reactor trip. In addition, Step 4 of EOP ES-0.1, Reactor Trip Response, explicitly directs the operator to verify or establish charging flow. It is highly probable that Step 4 of EOP ES-0.1 is reached within 10 minutes following reactor trip since the EOP E-0 to EOP ES-0.1 transition occurs very quickly (via Step 4 of EOP E-0). Based on the expected RXCP seal response to the loss of all cooling described in Section 10.1 of WCAP-10541, Rev. 2[(24)], a normal seal flow requirement of 3 to 5 gpm per pump (less than 10 gpm total) is expected if seal injection is restored by 10 minutes, i.e., prior to the transient heatup phase. Even if seal injection is delayed until about 30 minutes, the seal leakage rate is expected to be less than 21 gpm per pump or 42 gpm total. This is still within the capacity of one of the 60 gpm positive-displacement charging pumps.

Based on the above description, success for the CHS top event requires either

1. continued operation of 1 of 3 charging pumps for seal injection, or

2. operator action (based on training or the EOPs) to start a charging pump for seal injection within 30 minutes following reactor trip.

AUXILIARY FEEDWATER (AF6)

Auxiliary feedwater (AF6) is required to remove decay heat for this event. AFW is actuated on lo-lo steam generator level following the reactor trip or by manual actuation. SW backup to the AFW System is not available for this event, but sufficient inventory is provided by the CSTs and reactor makeup storage tank to remove decay heat for more than 24 hours. A sufficient amount of the condensate flow entering each AFW pump is diverted through its lube oil exchanger to cool the pump. SW or CCW are not required to keep the pump cool. SW is required, however, for room cooling for the A AFW pump room. Therefore, only motor driven AFW pump B and the turbine driven AFW pump are considered available in AF6.

It is assumed that secondary cooling is required for the entire event. Success of this node is 1 of 2 AFW pumps supplying at least 200 gpm to at least 1 of 2 steam generators for the entire event. If AFW fails, core melt occurs because ECCS is unavailable to remove the decay heat.

FIGURE 3.1.2/3-13

LOSS OF SERVICE WATER SYSTEM EVENT TREE

SWS	CHS	AF6

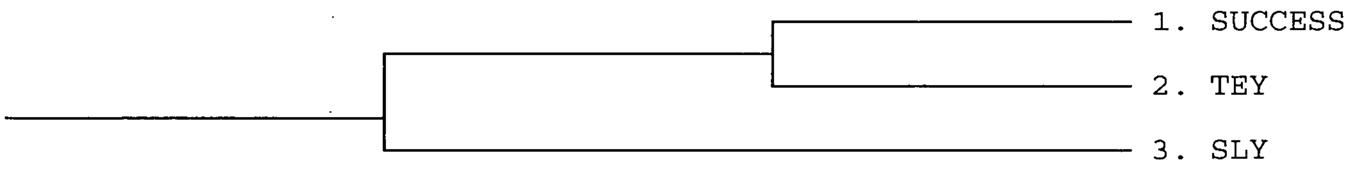


TABLE 3.1.2/3-13

SUCCESS CRITERIA FOR LOSS OF SERVICE WATER

Top Event Description	System Success Criteria	Necessary Operator Actions	Mission Time (hrs)
AF6 - AUXILIARY FEEDWATER	1 of 2 AFW pumps delivering at least 200 GPM to at least 1 of 2 steam generators.	If AFW not automatically initiated by event, manually initiate AFW. Confirm operation of system.	24
CHS - CHARGING PUMP OPERATION	1 of 3 charging pumps continues to operate after initiating event or 1 of 3 charging pumps started within 30 minutes after reactor trip for RXCP seal injection.	If a charging pump continues to operate after initiating event, verify RXCP seal injection. Manually start at least 1 charging pump, if none operating, within 30 minutes after reactor trip and establish RXCP seal injection.	24

3.1.2/3.15 Loss of Component Cooling Water System

INITIATING EVENT - LOSS OF COMPONENT COOLING WATER SYSTEM (CCS)

This initiator covers a loss of the CCS which causes loss of SI pumps and RHR pumps, residual heat exchangers, RXCPs, and containment spray pumps.

CHARGING PUMP OPERATION (CHG)

With loss of CCW cooling to the RXCP thermal barrier, the operator must maintain a minimum amount of charging flow to supply RXCP seal injection. One charging pump provides adequate RXCP seal cooling and thereby prevents a small LOCA due to seal degradation following loss of all seal cooling. Continued post-trip operation of the charging pump plus operator training most likely ensures that seal injection is maintained with little or no interruption following reactor trip. In addition, step 4 of EOP ES-0.1, Reactor Trip Response, explicitly directs the operator to verify or establish charging flow. It is highly probable that Step 4 of EOP ES-0.1 is reached within 10 minutes following reactor trip since the EOP E-0 to EOP ES-0.1 transition occurs very quickly (via Step 4 of EOP E-0). Based on the expected RXCP seal response to the loss of all cooling described in Section 10.1 of WCAP-10541, Rev. 2[(24)], a normal seal flow requirement of 3 to 5 gpm per pump (less than 10 gpm total) is expected if seal injection is restored by 10 minutes, i.e., prior to the transient heatup phase. Even if seal injection is delayed until about 30 minutes, the seal leakage rate is expected to be less than 21 gpm per pump or 42 gpm total. This is still within the capacity of one of the 60 gpm positive-displacement charging pumps.

Based on the above description, success for the CHG top event requires either

1. continued operation of 1 of 3 charging pumps for seal injection, or

2. operator action (based on training or the EOPs) to start a charging pump for seal injection within 30 minutes following reactor trip.

AUXILIARY FEEDWATER (AF3)

Following reactor trip, AFW flow is initiated on either a lo-lo SG level signal or by manual actuation. The mission time for this node is 24 hours. Success of AF3 is 1 of 3 AFW pumps supplying at least 200 GPM to 1 of 2 SGs for the 24 hour mission time.

OPERATOR ACTION - ESTABLISH MAIN FEEDWATER (OM2)

This top event models the availability of the MFW system to remove decay heat via the SG if AF3 fails. Success of MF2 requires one source of main feedwater delivering to at least one SG. The following paragraphs describe the additional actions needed to ensure availability of this source of normal feedwater.

In ES-0.1, Reactor Trip Response, the operator is directed to verify or control the average RCS temperature near 547°F and pressurizer pressure near 2235 psig. Based on these conditions, the main feed regulating valves would be expected to shut (based on TAVE <554°F). The operator also places the main feed regulating valve controllers in manual and verifies valve closure. Since pressure is maintained above 1815 psig, it is not likely that SI actuation occurs; MFW should therefore still be available for secondary cooling if needed. ES-0.1 therefore directs the operator to verify 200 gpm total feedwater flow using either MFW or AFW. ES-0.1 further directs the operator to maintain SG narrow range levels between 4 and 50%, so it is not likely that MFW is stopped due to high SG level.

If MFW is needed after feedwater isolation because AFW fails, it is most likely restored using FR-H.1, Response to Loss of Secondary Heat Sink. This procedure is entered if the secondary inventory in both SGs is low and the total feedwater flow is less than 200 gpm. To recover feedwater, the operator first attempts to establish MFW to at least one SG. This requires operation of a condensate pump and a MFW pump. If SI actuates (not expected for this event) it is necessary to reset the SI signal and the feed regulating bypass valve lockout from SI.

If MFW is unavailable, the operators are instructed to depressurize the RCS in order to block SI and depressurize a SG in order to provide secondary coolant flow from the condensate system. However, due to the complexity of the operator actions required to establish flow to a SG from the condensate system alone, it would take a significant amount of time to establish this flow and it is not included in the event tree modeling.

Success for this event tree node is 1 of 2 MFW trains delivering a flow of at least 200 gpm to at least one SG.

It is assumed that if AFW and MFW are unavailable, early core melt will occur. This is due to the unavailability of high pressure SI for bleed and feed.

FIGURE 3.1.2/3-14

LOSS OF COMPONENT COOLING WATER SYSTEM EVENT TREE

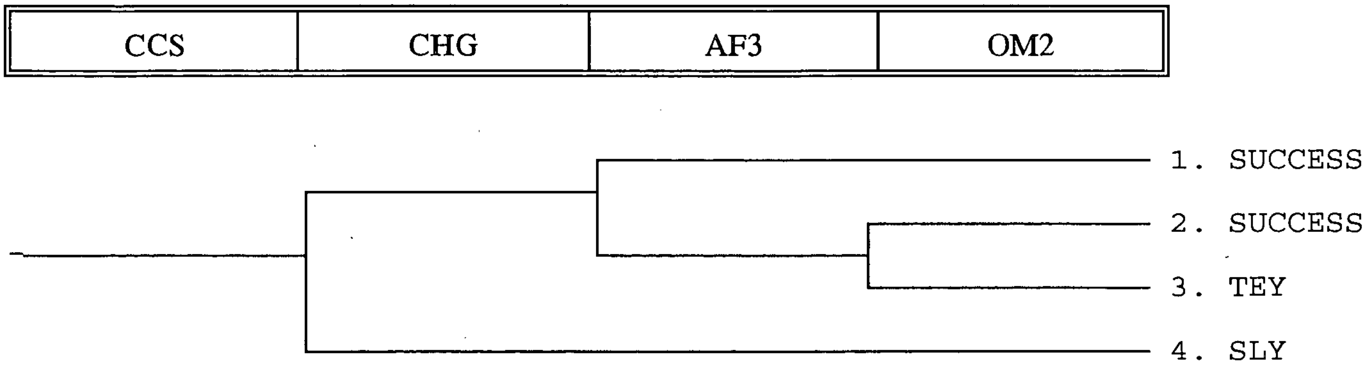


TABLE 3.1.2/3-14

SUCCESS CRITERIA FOR LOSS OF COMPONENT COOLING

Top Event Description	System Success Criteria	Necessary Operator Actions	Mission Time (hrs)
AF3 - AUXILIARY FEEDWATER	1 of 3 AFW pumps delivering at least 200 GPM to at least 1 of 2 steam generators	If AFW not automatically initiated by event, manually initiate AFW. Confirm operation of system.	24
OM2 - OPERATOR ACTION - ESTABLISH MAIN FEEDWATER	1 of 2 MFW trains delivering at least 200 GPM to at least 1 of 2 steam generators.	If AFW is automatically initiated by event but is not available, manually align and initiate MFW. Confirm operation of system.	24
CHG - CHARGING PUMP OPERATION	1 of 3 charging pumps continues to operate after initiating event <u>or</u> 1 of 3 charging pumps started within 30 minutes after reactor trip for RXCP seal injection.	If a charging pump continues to operate after initiating event, verify RXCP seal injection. Manually start at least 1 charging pump, if none operating, within 30 minutes after reactor trip and establish RXCP seal injection.	24

3.1.2/3.16 Loss of 125V DC Bus

INITIATING EVENT - LOSS OF 125V DC BUS (TDC)

This initiator includes all transients that begin with the loss of a 125V DC bus.

AUXILIARY FEEDWATER (AF4)

Following reactor trip, auxiliary feedwater (AF4) flow is initiated either automatically or manually/locally to provide secondary cooling. It is assumed that the turbine driven AFW pump is not available for this event and the A motor driven AFW pump must be started by locally operating its supply breaker using operating procedure E-EDC-38A.[30] The other motor driven AFW pump is started automatically on low SG levels or by manual actuation. Success of AF4 is 1 of 2 motor driven AFW pumps supplying at least 200 gpm to 1 of 2 steam generators. Success for AF4 also assumes adequate steam relieving capability. This is achieved by operation of steam dump to condenser, if available, or by operation of one of the relief valves (one PORV and five safety valves) for each active steam generator.

OPERATOR ACTION - ESTABLISH MAIN FEEDWATER (OM4)

This top event models the availability of the MFW system to remove decay heat via the steam generator if AF4 fails. Success of OM4 requires one train of MFW delivering to 1 of 2 SGs. The following paragraphs describe the actions needed to ensure availability of this source of feedwater.

In ES-0.1, Reactor Trip Response, the operator is directed to verify or control the average RCS temperature near 547°F and pressurizer pressure near 2235 psig. Based on these conditions, the main feed regulating valves are expected to shut (based on Tavg <554°F). The operator also places the main feed regulating valve controllers in manual and verifies valve closure. Since pressure is maintained above 1815 psig, it is not likely that SI actuation occurs. Thus, MFW is still available for secondary cooling if needed. ES-0.1 therefore directs the operator to verify 200 gpm total feedwater flow using either MFW or AFW. ES-0.1 further directs the operator to maintain SG narrow range levels between 4 and 50%, so it is not likely that MFW is stopped due to high SG level.

If MFW is needed after feedwater isolation because AFW fails, it is most likely restored using FR-H.1, Response to Loss of Secondary Heat Sink. This procedure is entered if the secondary inventory in both SGs is low and the total feedwater flow is less than 200 gpm. To recover feedwater, the operator first attempts to establish MFW to at least one SG. This requires operation of a condensate pump and a MFW pump which would still be running. If SI has been actuated (not expected for this event) it is also necessary to reset the SI signal and the feed regulating bypass valve lockout from SI.

If MFW is unavailable, the operators are instructed to depressurize the RCS in order to block SI and depressurize a SG in order to provide secondary coolant flow from the condensate system. However, due to the complexity of the operator actions required to establish flow to a SG from the condensate system alone, it would take a significant amount of time to establish this flow and it is not included in the event tree modeling.

Success for this event tree node is 1 of 2 MFW trains delivering a flow of at least 200 gpm to at least one SG.

OPERATOR ACTION - BLEED AND FEED (OB3)

If secondary cooling with AFW and the MFW is unavailable, the operators are instructed to initiate primary system bleed and feed. Emergency operating procedure FR-H.1, Response to Loss of Secondary Cooling, instructs the operators to initiate bleed and feed if secondary cooling is lost and wide range steam generator level in either steam generator drops below 15% or pressurizer pressure increases above 2335 psig. The operators start at least one high pressure SI pump and establish a RCS bleed path by opening the available pressurizer PORV. The A SI pump has to be started locally, while the B SI pump can be started manually. According to FR-H.1 instructions, it is likely that bleed and feed cooling would be established by 30 minutes. SG secondary dryout would be expected at approximately one hour.

Success of OB3 is 1 of 2 high pressure SI trains delivering flow to 1 of 2 RCS cold legs with the available pressurizer PORV open. Bleed and feed initiation prior to SG dryout with this success criterion is expected to result in effective decay heat removal. For simplicity, it is assumed that bleed and feed initiated by 30 minutes using one high pressure SI pump and one pressurizer PORV results in success.

It is assumed that failure of this node results in early core melt due to loss of all secondary cooling.

HIGH PRESSURE RECIRCULATION (HR2)

If OB3 is successful, long term cooling is addressed. Long term cooling is provided by sump recirculation. If RCS pressure remains above 140 psig, a low pressure SI train is lined up to take suction from the containment sump and discharge to the suction of the high pressure SI pumps via the residual heat exchangers. This lineup is referred as a SI/RHR train in the Emergency Operating Procedures. Since control power for the A RHR pump is lost when DC bus BRA-104 is lost, the A RHR pump must be started locally for recirculation operation, while the remaining RHR pump can be started manually.

Success of this node requires at least 1 of 2 SI/RHR train to provide flow to 1 of 2 RCS cold legs.

It is assumed that failure of this mode results in late core melt.

FIGURE 3.1.2/3-15

LOSS OF 125V DC BUS

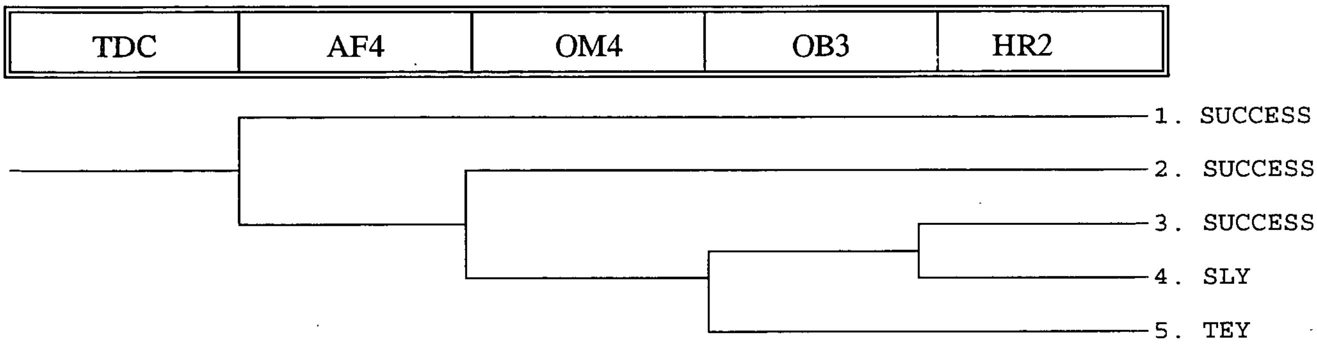


TABLE 3.1.2/3-15

SUCCESS CRITERIA FOR LOSS OF 125V DC BUS

Top Event Description		System Success Criteria	Necessary Operator Actions	Mission Time (hrs)
AF4 -	AUXILIARY FEEDWATER	1 of 2 motor driven AFW pumps supplying at least 200 GPM to at least 1 of 2 steam generators.	Locally start A motor driven AFW pump from its supply breaker per E-EDC-38A.	24
OM4 -	OPERATOR ACTION - ESTABLISH MAIN FEEDWATER	1 of 2 MFW trains delivering at least 200 GPM to at least 1 of 2 steam generators.	If AFW is automatically initiated by event, but is not available, manually lineup and initiate MFW.	24
OB3 -	OPERATOR ACTION - BLEED AND FEED	1 of 2 high pressure SI trains delivering flow to 1 of 2 RCS cold legs, 1 of 1 pressurizer PORVs open (bleed and feed initiated prior to secondary dryout - assume 30 minutes).	Manually open only available PORV and its associated block valve, start SI pumps (A SI pump must be started locally from it supply breaker, while the remaining SI pump can be started manually).	Run for 24 hours.
HR2 -	HIGH PRESSURE RECIRCULATION	1 of 2 SI/RHR train delivering flow from the containment sump to 1 of 2 RCS cold legs, sump valve on operable recirc. train open.	Manually align high pressure containment sump recirculation on low RWST level (may include re-start of RHR pump), align CCW to RHR Hx, confirm operation of system. A RHR pump and A SI pump must be locally started from their supply breakers.	20.5 hours

3.1.2/3.17 Loss of Station and Instrument Air System

INITIATING EVENT - LOSS OF INSTRUMENT AIR (INA)

This initiator includes all transients caused by a loss of instrument air.

AUXILIARY FEEDWATER (AF5)

Auxiliary feedwater is required to remove decay heat following a reactor trip. AFW is actuated on lo-lo SG level. Operator action is required according to procedure E-AS-01 to locally operate AFW pump discharge valves in order to control SG levels.

It is assumed that secondary cooling is required for the entire event. Success of AF5 is 1 of 3 AFW pumps supplying at least 200 gpm to 1 of 2 steam generators for the entire event. Success for AF5 also assumes adequate steam relieving capability. Because the loss of instrument air disables the steam dump valves and secondary PORVs, steam relief is provided by the SG safeties or by local operation of the SG PORVs.

OPERATOR ACTION - ESTABLISH MAIN FEEDWATER (OM3)

If AF5 fails, main feedwater (OM3) is used for secondary cooling. Emergency operating procedure FR-H.1, Response to Loss of Secondary Heat Sink, is entered via the critical safety function trees. FR-H.1 instructs the operators to attempt to restore AFW and to line up MFW to provide secondary cooling. For this event, this requires local operation of components, including FW bypass valves.

Success of OM3 is 1 of 2 MFW trains delivering at least 200 gpm flow to 1 of 2 steam generators for the entire event.

It is assumed that a loss of main feedwater following a loss of auxiliary feedwater results in early core melt.

FIGURE 3.1.2/3-16

LOSS OF STATION AND INSTRUMENT AIR EVENT TREE

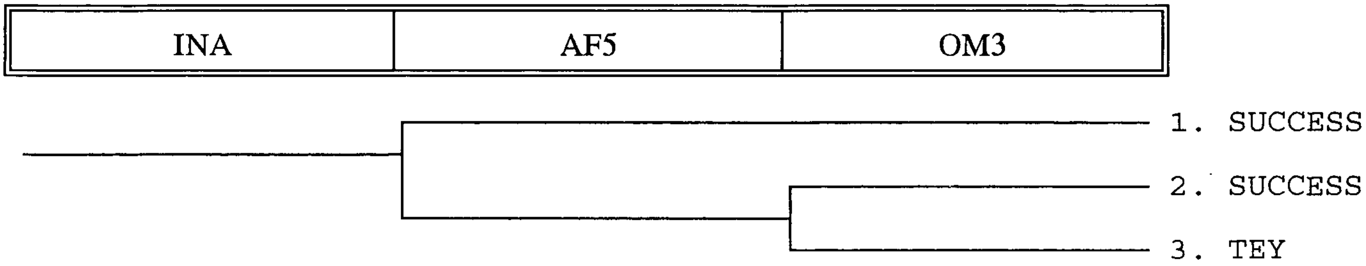


TABLE 3.1.2/3-16

SUCCESS CRITERIA FOR LOSS OF STATION AND INSTRUMENT AIR

Top Event Description	System Success Criteria	Necessary Operator Actions	Mission Time (hrs)
AF5 - AUXILIARY FEEDWATER	1 of 3 AFW pumps delivering at least 200 GPM to at least 1 of 2 steam generators.	Local operation of AFW pump discharge valves to control SG levels. Confirm operation of AFW system.	24
OM3 - OPERATION ACTION - ESTABLISH MAIN FEEDWATER	1 of 2 MFW trains delivering at least 200 GPM to at least 1 of 2 steam generators.	If AFW is automatically initiated by event but is not available, manually align and initiate MFW. Locally control FW flow from FW bypass valves.	24

3.1.2/3.18 Additional Information Regarding Bases for Top Event Success Criteria

All of the top event success criteria are presented in Tables 3.1.2/3-1 through 3.1.2/3-16. For most of the top events, the descriptions in Sections 3.1.2/3.2 through 3.1.2/3.17 provide sufficient information explaining the bases for these success criteria. Additional details are provided in this section for those top events requiring further explanation. The top events are arranged alphabetically.

ACC - Accumulator Injection

For large LOCA event, the accumulators are needed early in the accident to refill the reactor vessel and add water to the core. According to Section 14.3 of the USAR[14], injection from one accumulator into the intact loop, followed by low-head injection from one RHR pump, is adequate for core cooling. The accumulator parameters assumed in the ECCS analyses are approximately 1250 ft^3 water volume per accumulator and 700 psig minimum gas pressure.

For a medium or small LOCA (less than 6" diameter), the accumulators are not required if high pressure SI (HPI) is available. If HPI fails, however, operator action to cool down and depressurize the RCS to cause low pressure injection from the accumulators and RHR pumps is assumed. To keep the event tree analysis simple yet conservative, success with accumulator injection can be achieved assuming one accumulator injects into one intact cold leg (i.e., the same as large LOCA). For a small or medium LOCA, however, it is not necessary to assume that the accumulator flow to the broken loop spills (the break is much smaller than the 10" accumulator line), so the success criterion for ACC could be further relaxed to model accumulator injection to any 1 of 2 loops.

An analysis specifically modeling injection from only 1 accumulator for the above small LOCA inadequate core cooling (ICC) transient is not readily available. However, Technical Training Session 20 of the Westinghouse Owners Group Loss of Reactor or Secondary Coolant Training Program[31] presents a TREAT analysis for a 4-loop plant similar to Wolf Creek that can be used to evaluate the success criteria for Kewaunee. During the post-LOCA cooldown and depressurization for this small LOCA ICC scenario, a significant increase in Reactor Coolant System (RCS) inventory occurs during the time period when the four accumulators inject (RVLIS wide range level increases from 50% to 80%). A total water volume of 1960 ft^3 is injected prior to accumulator isolation. In the WOG analysis the accumulators are isolated after the hot leg temperatures decreased to less than 400°F. This action is taken to prevent nitrogen injection. In the Kewaunee procedures (ES-1.2, FR-C.1, and FR-C.2), accumulator isolation occurs after the RCS pressure decreases to 210 psig or less. When the Kewaunee accumulators are isolated, almost all of the 1250 ft^3 water volume is injected. Since the RCS component volumes for Kewaunee are about one-half of those for the 4-loop plant, a similar response in RCS inventory levels could be achieved with one accumulator for the 2-loop plant as is achieved above for the 4-loop plant. Based on this evaluation, one instead of two accumulators is sufficient for the small LOCA accumulator success criterion.

AC2 - Power Restored in 2 Hours

Following a station blackout, if the turbine-driven auxiliary feedwater (AFW) pump is not available (i.e., no feedwater is being delivered to the SGs), analyses performed for Kewaunee show that a successful recovery can be made if power is restored in 2 hours or less. These analyses are discussed in detail in section 3.1.2/3.12. With AC power restored in 2 hours or less, bleed and feed cooling of the RCS prevents core damage. Success for this top event is restoration of AC power to 1 of 2 vital buses in 2 hours or less.

ACX - Power Restored in X Hours

This top event models the probability that AC power is restored in X hours following a station blackout, where X depends on the success or failure of the RCS cooldown. As discussed in top event OCD, a successful RCS cooldown to obtain accumulator injection and lower RCS pressure and temperature (which reduces any reactor coolant pump seal leakage) delays core uncovery until 2 hours after SG dryout. SG dryout occurs around one hour following termination of auxiliary feedwater (see top event AFX below). Thus, core uncovery does not occur until at least one hour after termination of auxiliary feedwater if the RCS cooldown is successful and until 3 hours after termination of auxiliary feedwater if the RCS cooldown is successful. It is conservatively assumed that if power is not restored prior to initiation of core uncovery, a successful recovery cannot be made.

As discussed in detail in Section 3.1.2/3.12, the condensate supply is sufficient to remove decay heat for more than 24 hours, and DC power from the safeguards batteries for the turbine-driven AFW pump is available for 8 hours. Although this pump could be controlled manually, instrumentation to monitor the pump, primary and secondary parameters is lost after DC power is lost. It is therefore conservatively assumed that auxiliary feedwater continues only until DC power is lost at 8 hours. This means that, to avoid core uncovery and subsequent core damage, AC power must be restored within 11 hours if the RCS cooldown is successful and within 9 hours if the RCS cooldown is unsuccessful.

AFG - Auxiliary Feedwater

This top event addresses the availability of the AFW for the AWS event. AFW must remove heat from the RCS to prevent a loss of heat sink. The amount of AFW flow required for a successful recovery at Kewaunee is based on the amount assumed in the supporting analyses described in WCAP-8830[32] and the superseding ATWS submittal[33]. The analyses for the 2-loop plants (with Model 44 or Model 51 steam generators) assume 400 gpm for the case of at least half of the total AFW flow condition. These analyses are therefore applicable to Kewaunee with 2 of the 3 AFW pumps delivering at least 400 gpm to at least 1 of 2 steam generators.

AFM - Automatic Reactor Trip Failure Mode

This node describes the failure mechanism leading to the ATWS event. Both failure of the RPS logic and failure of the reactor trip breakers as well as the rods failing to insert are modeled as the causes of the failure of the automatic reactor trip function to shut down the reactor. For this top event, a "success" path does not exist due to the nature of the event.

The AFM top event discussion in Section 3.1.2/3.13 provides additional information.

AFX - Auxiliary Feedwater

With the exception of LLO and VEF, all the event trees rely on AFW and secondary side steam relief for cooldown or decay heat removal. For all events in which there is a reactor trip, a minimum feedwater flow of 200 gpm to one steam generator is the limiting success criterion. This minimum feed flow requirement is consistent with the loss of normal feedwater transient and other events analyzed in Section 14.1 of the USAR.(14) It is also the same minimum feed flow requirement used extensively in the emergency procedures. Top events AF0, AF1, AF3, AF4, AF5 and AF6 assume the minimum feedwater flow criterion for success. Differences in these top event success criteria reflect differences specific to the initiating event, i.e., differences in the number of AFW pumps available (e.g., 1 of 2 versus 1 of 3 AFW pumps), the number of steam generators available (e.g., 1 of 2 intact SGs versus 1 of 1 intact SG), and manual versus manual plus local actions required.

The USAR transients are typically analyzed assuming a one minute delay in feedwater flow from the start of the incident. However, because of the initial mass of the secondary water, a longer time delay would be acceptable before AFW cooling or an alternate decay heat removal mechanism must be provided. Based on the minimum AFW flow (200 gpm) and the capability for bleed and feed cooling, the time at the onset of secondary dryout is an acceptable value to use for this delay. These times have been evaluated for two different SG inventory conditions at the time of reactor trip (nominal water level in both SGs and low-low level of 5% narrow range in both SGs). Based on decay heat estimates comparable to ANS-5.1-1979 plus 2-sigma uncertainty and water inventories of 100,000 lbm per SG at nominal conditions and 70,000 lbm per SG at lo-lo level (5% narrow range), the secondary dryout times are estimated to be about 60 minutes (nominal water volume) and 35-minutes (water volume corresponding to lo-lo level). The decay heat rates used are expected to be conservatively high by about 5 to 10%. Furthermore, the initial heatup of the SG water to the relief valve set-pressure has been ignored. This initial heatup (approximately 40°F) is a 6% conservatism that would be eliminated in a more realistic calculation or simulation with TREAT. One possible non-conservatism, however, is the heat addition due to the reactor coolant pumps (RXCPs). If operating, the RXCPs increase the integrated residual heat by about 15% for both cases. Considering these sensitivities, the above dryout times are reasonable estimates if the RXCPs are running and conservatively low by about 5 to 10 minutes if the RXCPs are tripped.

Based on this, it is reasonable to allow for a delay of up to 30 minutes before AFW (or alternate cooling) must be established. If AFW is initiated at 30 minutes at the minimum flow from the motor-driven AFW pump, as assumed in the loss of normal feedwater analysis (200 gpm), the secondary would be able to remove approximately 32 MWt. This is adequate to assure removal of the assumed conservative decay heat (1.9% or 31 MWt).

AF2 - Turbine-Driven Auxiliary Feedwater Pump

The AFW System is required to remove decay heat from the RCS. For the station blackout, only the turbine-driven AFW pump is available. The turbine-driven auxiliary feedwater pump starts automatically upon loss of offsite power (i.e., loss of power to the MFW electrical buses). This pump also starts on low-low SG levels or by manual actuation.

Success of the turbine-driven AFW pump requires that the pump supplies flow (at least the minimum feed flow requirement of 200 gpm) to at least one steam generator. The analytical basis for this requirement is the same as that provided previously for the AFX top events.

AMS - ATWS Mitigating System Actuation Circuitry (AMSAC)

The function of AMSAC is described in the AWS event tree write-up of Section 3.1.2/3.13. For Kewaunee, AMSAC is armed at all power levels.[14]

CCV - Core Covered

For the station blackout event, this top event addresses core uncovery due to the RXCP seal LOCA after the safeguards systems are restored. The event tree failure path is the probability of core uncovery resulting from RXCP seal leakage. If core uncovery is postulated, early core damage is assumed. If core uncovery does not occur, there is still a small amount of RXCP seal leakage (about 21 gpm per pump), so some RCS makeup is still required. Therefore, for the event tree success paths for CCV, there is a small LOCA and RCS makeup is required.

Westinghouse has developed a probabilistic RXCP seal LOCA model that can be used for the calculations necessary to quantify CCV. This model is similar to the one presented in WCAP-10541 Rev. 2[24] except that some additional conservatisms have been incorporated to address NRC concerns related to the "binding" and "popping" modes of failure. A description of the Westinghouse RXCP seal LOCA model can be found in Reference 34.

CHG/CHS - Charging Pump Operation

For the CCW, LSP, TRA, TRS and SWS events, is assumed that RXCP seal cooling is not available due to loss of CCW to the RXCP thermal barrier. Therefore, it is necessary to operate one charging pump to provide for seal injection for seal cooling The CHG/CHS top events descriptions from Sections 3.1.2/3.14 or 3.1.2/3.15, along the WCAP-10541, Rev. 2 provide adequate justification for this top event.

EC3/EC4 - Operator Action to Cooldown and Depressurize the RCS per ECA-3.1/3.2

To recover from a SG tube rupture accident, the ECA-3.1 and possibly ECA-3.2 recovery procedures may need to be followed for several different reasons:

1. SG overfill occurs and a secondary side relief valve on the ruptured SG sticks open (OS1 and SSV fail),

2. The ruptured SG can not be isolated from the intact SG used for cooldown (ISO fails),

3. There is no feedwater available to the intact SG so the ruptured SG must be used for cooldown, or

4. High pressure injection is unavailable (HI1 fails) resulting in a loss of RCS subcooling (Step 17 of E-3).

The EC3/EC4 top event description provided in Section 3.1.2/3.5 provides an adequate basis for the EC3/EC4 success criteria.

ES1 - Operator Action-Cooldown and Depressurize RCS for Charging Flow

This top event is featured in the Small LOCA event tree of Figure 3.1.2/3-3. The actions modeled are similar to those used in the EC3 top event described above for SGR recovery contingencies. The ES1 top event description provided in Section 3.1.2/3.4 summarizes and provides a basis for the ES1 success criterion.

HI0, HI1, HI2, HI3, HI4 - High Pressure Injection

High pressure safety injection (SI) for Medium LOCA, top event HI0, is automatically actuated on an SI signal upon receipt of a low pressurizer pressure signal or high containment pressure signal. The design basis[14] success criterion of 1 of 2 SI pumps injecting to the intact loop is assumed for HI0. A mission time of 3.5 hours, identical to that for SLO, is selected for simplicity.

The success criterion for high pressure injection for small LOCA (break sizes less than 2" diameter), top event HI2, is essentially the same as that assumed for the MLO (HI0) except that it is not necessary to assume that flow to the broken loop spills (the size of the break is much smaller than the size of the 10" injection line). Therefore, the limiting success criterion is relaxed to model 1 of 2 SI pumps injecting to 1 of 2 loops. Also, with the smaller break, credit is taken for the operator manually starting the SI pumps if the SI signal fails to automatically start them.

High pressure injection for SGR, top event HI1, is modeled but not required for recovery. The HI1 description in Section 3.1.2/3.5 provides additional details on high pressure injection for the SGR event.

High pressure injection for secondary breaks, top event HI3, requires one SI pump to deliver sufficient boric acid from the boric acid tank (BAT) to the RCS to ensure core shutdown. Success is at least one of two SI pumps injecting the contents of one BAT into at least one of two cold legs.

High pressure injection for interfacing system LOCA events, top event HI4, has the same requirements as HI0 with the exception of a longer mission time as described in section 3.1.2/3.7.

For all of the above events with the exception of medium LOCA, interfacing systems LOCA and also for the transients events, high pressure injection is also be required for bleed and feed cooling if AFW fails. The success criteria for OB1, OB2, OB3, OB4 and OB5 provided later in this section describe the basis for bleed and feed cooling.

HR0, HR1, HR2 - High Pressure Recirculation

If the primary system pressure is high following a medium or small LOCA, long term cooling is established by pumping sump water via the RHR pumps to the suction of the SI pumps. This requires operator action to reposition valves to properly align the SI pumps to the RHR discharge lines. The RHR pump suction valves (SI-300A, SI-300B) and the SI pump suction isolation valves (SI-5A, SI-5B) must be closed by the operator in order to transfer from the injection phase to the recirculation phase.

At switchover time for medium LOCA, the decay heat is low enough that only one of two SI pumps is required. The limiting success criterion is therefore one of two RHR pumps supplying flow to the suction of one of two SI pumps, injecting to the intact cold leg.

Consistent with the licensing basis, injection to 1 of 1 intact loop is conservatively assumed for Medium LOCA. For Small LOCA, spill is neglected and the limiting success criterion is relaxed to assume delivery to 1 of 2 loops.

High pressure recirculation is also necessary for a number of transient events if feedwater is not available and bleed and feed recovery is necessary. The open pressurizer PORV essentially creates a small LOCA, so the success criterion for high pressure recirculation for long term bleed and feed is assumed to be the same as small LOCA. Top event HR2 is used for the loss of 125V DC bus event in which it is assumed that one pressurizer PORV is inoperable due to the initiating event. The success criterion for HR2 is identical with HR1 except that one RHR pump and one SI pump must be started locally from their supply breakers.

ISO - SG Isolation by MSIV Closure

In the normal EOP E-3 SGR recovery, the ruptured SG must be isolated by the operator from the intact SG by closure of an MSIV. Other paths to and from the ruptured SG also require isolation (e.g., blowdown, steam supply to the turbine-driven AFW pump, etc.). Isolation of these paths, however, is not as crucial to the recovery as main steam isolation. It is preferable to close the MSIV for the ruptured SG since this gives the operator the option of using steam dump to condenser, if available, for the subsequent cooldown using the intact SG. Should the MSIV for the ruptured SG fail to close, the MSIV for the intact SG could be closed and the corresponding atmospheric relief valve (or PORV) could be used for the cooldown. The initial cooldown is limited to about 500°F and only the intact SG with feedwater would be used. Therefore, success for the ISO function is determined by the ability to close at least one MSIV on either SG. It should again be noted that it is preferable that only the MSIV for the ruptured SG be closed. For a design basis SGR[14], it is assumed that the operator must identify the ruptured SG and perform the ISO isolation function by 15 minutes for ISO to be successful. This will allow adequate time for completion of subsequent actions (see OS1) by about 30 minutes.

IS1 - Main Steam Isolation

To allow for an arbitrary break location for the two secondary break events, success of main steam isolation requires closure of both MSIVs. This assumption ensures that the broken SG is isolated, whether it is the A or the B generator.

LI1 and LI2 - Low Pressure Injection

The RHR pumps function as low pressure ECCS pumps to provide low pressure SI from the RWST directly to the reactor vessel. The RHR pumps aid in the filling of the reactor vessel and supply water to maintain reactor vessel water level and complete the core reflooding process. For the large LOCA (top event LI1), one of the two RHR pumps with its associated valves and piping is sufficient to meet the requirements for core cooling. This is consistent with the assumptions made for the large LOCA ECCS analysis.

To justify elimination of the SI pumps as a requirement for the injection phase of the large LOCA, it is noted that for design basis COBRA/TRAC analysis of the RHR flow from 1 pump injecting at 0 psig RCS pressure is 1860 gpm. The flow assumed from 2 SI pumps (one line spilling) is only 413 gpm at 0 psig. (Note: failure of one RHR pump is typically considered as the limiting single failure for large LOCA ECCS analysis since assumptions leading to minimum containment pressure are generally conservative). The conservatisms in the design analysis (e.g., high decay heat, derated RHR flows, minimum containment back-pressure to maximize break flow, etc.) are expected to overshadow the 20 to 25% increase in the injection flow caused by addition of the SI pumps. Thus, for better estimate event tree modeling, only one RHR pump is required for large LOCA. The expected mission time for LI1 for large LOCA is around one hour.

For medium and small LOCA, low pressure safety injection (top event LI2) is addressed if high pressure injection (HI0 and HI2 respectively) fails. With high pressure SI failure, accumulator injection (ACC) and operator action to cool down and depressurize the RCS (OP1 and OP2) are also assumed necessary. Since high pressure SI is assumed failed for these applications, the description presented above for large LOCA applies for LI2 for small and medium LOCA. However, for a large LOCA, the automatic initiation of low pressure injection is required. For a small or medium LOCA, the operator manually initiates low pressure injection following the RCS cooldown and depressurization. The time to switchover to low pressure recirculation is considerably longer since the break flows are smaller. For the small LOCA, the time to switchover is several hours. For simplicity, the LI2 mission time assumed for both medium and small LOCA is 1.0 hour.

LR1/LR2 - Low Pressure Recirculation

The LR1 top event is used in the large LOCA, medium LOCA, and small LOCA event trees. The LR2 top event is used in the medium LOCA and small LOCA event trees. The success criterion is first discussed for large LOCA.

The low-head recirculation function is to remove decay heat from the core and sensible heat from the containment sump water. The low-head recirculation function is provided by the RHR system.

An adequate volume of water must be delivered during the injection phase to assure that sufficient water is available within the containment to meet the NPSH requirements of one SI pump and one RHR pump during the recirculation mode. This required amount of water determines the lo-lo level setpoint for the RWST.

The shortest time to the lo-lo level switchover setpoint is approximately one hour for a large LOCA (all safeguards pumps operating). The RHR system is switched from the injection mode to the recirculation mode upon reaching the lo-lo level setpoint in the RWST. The operator must perform manual actions during this switchover. For example, the RHR pump suction isolation valves (SI-300A, SI-300B) and the SI pump suction isolation valves (SI-5A, SI-5B) must be closed by the operator in order to transfer from the injection phase to the recirculation phase.

For the medium and small LOCA, low pressure recirculation (LR1) is addressed if high pressure injection (HI0 and HI2, respectively) fails and if high pressure recirculation (HR0 and HR1 respectively) fails. With high pressure SI failure, accumulator injection (ACC), low-head injection (LI2), and operator action to cooldown and depressurize RCS the (OP1 and OP2) are also assumed necessary. Since high pressure SI is assumed failed for these applications, the description presented above for the large LOCA applies for LR1 for small and medium LOCA. The times to switchover are considerably longer since the break flows are smaller. For small LOCA, the time to switchover is several hours or longer.

For all three events, the flow requirement for LR1 is assumed to be the same as that required for LI1 and LI2, i.e., 1 out of 2 RHR pumps recirculating flow into the reactor vessel.

LTS - Long Term Shutdown

If automatic trip, manual trip, or manual/local opening of the breakers for the rod drive MG sets does not shut down the reactor early in the transient and the peak RCS pressure has not exceeded the stress criterion within the first few minutes of the transient, then alternate means to achieve subcriticality and maintain the shutdown condition are available. The emergency operating procedure FR-S.1 instructs the operator to begin manual rod insertion. Boration of the RCS is also initiated with high pressure charging pumps with emergency boration via the boric acid tank.

The LTS top event description in Section 3.1.2/3.13 summarizes the bases for the two alternate success criteria for long term shutdown.

MRT - Manual Reactor Trip

This top event models the manual trip of the reactor by the operators. If the automatic reactor trip function fails due to a failure of the RPS logic, the operator can still successfully trip the reactor manually. By this method, the reactor is tripped prior to the earliest time at which SG dryout would occur (within 2 minutes for an ATWS followed by a loss of feedwater at full power). The success criterion is operator action to trip the reactor within 2 minutes.

OB1, OB2, OB3, OB4, OB5 - Operator Action - Bleed and Feed

Decay heat removal can be accomplished by bleed and feed if the MFW and AFW Systems fail. As explained below, success requires the operator to recognize the need for action, start or verify operation of at least one of the high pressure SI pumps, and open at least one of the pressurizer PORVs (and the associated block valve, if necessary). To ensure success (prevent core damage) it is sufficient to assume injection into 1 of 2 loops by the time of secondary dryout, i.e., a limiting time of about 30 minutes according to the AFX description provided earlier.

An analysis[35] was performed using the thermal hydraulic code TREAT-PC that shows that one pressurizer PORV and one SI pump adequately removes decay heat if bleed and feed is initiated when SG wide range level in one of the SGs decreases below 10% following a loss of all feedwater transient. The wide range level in an SG does not decrease to less than 10% until 2080 seconds (or approximately 35 minutes) into the transient. This is later than the success criterion for OB1, OB2, OB3, OB4 and OB5 of initiation of bleed and feed prior to 30 minutes (SG dryout) in order to attain a successful recovery. Thus, the analysis not only confirms that a successful recovery using bleed and feed cooling of the RCS can be made with only one pressurizer PORV and one SI pump, but also that the 30 minute criterion for initiation of bleed and feed (based on a conservative analysis to determine SG dryout time) is conservative.

The differences in the top events reflect differences in the equipment initially available. For OB1, manual opening of 1 of 2 PORVs is assumed; high pressure SI (HI1, HI2 and HI3) is first checked since the events in which OB1 is featured are ones in which SI actuation is expected. In OB2 for TRA and TRS, manual opening of 1 of 2 PORVs and manual starting of 1 of 2 SI pumps is modeled. For OB3, manual opening of 1 of 1 PORV and manual or local starting of 1 of 2 SI pumps is modeled since the initiating event (loss of a 125V DC bus) causes the loss of one of the PORVs and one SI pump must be started locally. For OB4, manual opening of 1 of 2 PORVs and verification of 1 of 2 SI pumps running with the suction aligned to the RWST is modeled. OB5 is the same as OB2 except that only the safeguards air compressors (A, B, and C) are available. OB5 applies to the loss of offsite power event.

<u>OCD - Operator Action - RCS Cooldown</u>

As described in the accident progression for station blackout, operator action to cool down the RCS prolongs the time to core uncovery if AC power is not restored. By following the procedures (ECA-0.0), the operator is instructed to depressurize the intact steam generators to 300 psig by manually or locally dumping steam at the maximum rate using the steam generator PORVs. By establishing AFW to and using the associated SG PORVs for one or both SGs, it is possible to cool the RCS to 410°F within one hour. By reducing the temperature to 410°F or less, the RXCP seal leak rate is significantly reduced. In addition, for the more highly voided "problem" scenarios, the cooldown to 410°F results in an RCS depressurization that causes most of the contents of the accumulators to be injected. Success for OCD is based on operation of the SG PORV on at least one of two SGs with AFW. Analyses on plants similar to Kewaunee have shown that if the RCS cooldown is successful, core uncovery is delayed for 2 hours following SG dryout.[24]

OMX - Operator Action - Establish Main Feedwater

If AFW is not available, the operator can manually lineup and initiate MFW. To restore MFW the operator resets SI (if necessary), resets feedwater isolation, restarts the MFW pumps, and opens the MOV discharge valves and bypass regulating valves. All of these actions are performed manually from the control room at Kewaunee.

The success criteria for top events OM0, OM1, OM2, OM3 and OM4 require 1 of 2 MFW trains to deliver at least 200 gpm prior to SG dryout at 30 minutes. The bases for the flow and time requirements were previously discussed in top event AFX. OM4 is found in the TDC accident sequence where one vital DC bus is assumed to have failed.

Differences in the success criteria for top events OM0, OM1, OM2, OM3 and OM4 reflect differences in the number of steam generators available (e.g., 1 of 2 intact SGs versus 1 of 1 intact SG), and manual versus manual plus local (for loss of station and instrument air) actions required.

OP1 - Operator Action - Cooldown and Depressurize the RCS

If high pressure injection fails, the core can still be successfully recovered if the operator performs a cooldown and depressurization to allow injection from the accumulators and low pressure SI system (RHR pumps). This cooldown and depressurization is performed with the Emergency Operating Procedure ES-1.2, Post LOCA Cooldown and Depressurization. As a backup to ES-1.2, the degraded and inadequate core cooling procedures (FR-C.2 and FR-C.1, respectively) also direct the operator to depressurize the secondary to allow low pressure injection. The secondary depressurization can be accomplished by dumping steam to the condenser, if available, or by dumping steam through the atmospheric relief valves and using at least one SG supplied with feedwater.

The time at which the OP1 actions must be taken is estimated based the NOTRUMP small LOCA analysis. The success criterion for OP2 (similar to OP1 except for a Small LOCA) considers the more limiting action time for the 2" break for the small LOCA. The timing for OP1 is based on the expected required action time for a break somewhere in the range of 2" to 4", i.e., for a break size for which successful accumulator injection would occur without secondary depressurization. The 3" break depressurizes the RCS such that accumulator injection occurred at 686 seconds (11.4 minutes), which helped recover level to above the top of the core by 868 seconds (14.5 minutes). This accident analysis case is, of course, successful without the secondary depressurization. However, had the break been slightly smaller or high pressure SI not been available, the OP1 action would have been important for restoring core level. In view of the conservatisms in the analysis (i.e., high decay heat), it is reasonable to extend the operator action time for initiation of the OP1 cooldown until 15 minutes.

OP2 - Operator Action To Cooldown and Depressurize the RCS for ACC and LI2

Again, if high pressure injection fails, a successful recovery for small LOCA is still possible if the operator performs a cooldown and depressurization to allow injection from the accumulators and low pressure SI system (RHR pumps). This cooldown and depressurization is performed in the Emergency Operating Procedure ES-1.2, Post LOCA Cooldown and Depressurization. As a backup to ES-1.2, the degraded and inadequate core cooling procedures (FR-C.2 and FR-C.1, respectively) direct the operator to depressurize the secondary at an earlier time or at a faster rate if the core exit temperatures become excessive. The secondary depressurization can be accomplished by dumping steam to the condenser, if available, or by dumping steam through the SG PORVs to the atmosphere and using at least one steam generator supplied with feedwater.

The time at which the OP2 actions must be taken is estimated based on the expected depletion for a limiting 2" break. A TREAT analysis for an inadequate core cooling transient for the 3-loop Farley plant (see Appendix F of WCAP-11370,[36]) can be used to evaluate the limiting action time for Kewaunee. For this 2" cold leg break inadequate core cooling transient for the 3-loop plant, a prolonged core uncovery starts to occur at 2548 seconds (42.5 minutes). By 2766 seconds (46.1 minutes), the core exit temperatures reach 1200°F. A secondary depressurization initiated at 2800 sec (46.7 minutes) results in accumulator injection and rapid recovery of level in the core. Since the component volumes of the 3-loop plant are about 50% larger than those of Kewaunee, the depletion times and allowed OP2 action times are also about 50% longer. Thus, the secondary depressurization initiated at 46.7 minutes for the 3-loop plant has to be initiated at around 30 minutes for Kewaunee for a similar response. For breaks smaller than 2" diameter, the action times for OP2 are longer than 30 minutes.

ORI - Operator Action - Restore RCS Inventory

When power is restored following a station blackout event, the emergency procedures (ECA-0.0 and ECA-0.2 or ECA-0.1) instruct the operator to restore the safeguard systems. Safety injection to restore RCS inventory is required and decay heat removal must be established (or maintained). As explained in the ORI top event description of section 3.1.2/3.12, it is sufficient to model recovery from a small LOCA for the post-station blackout period. Therefore, consistent with the small LOCA, high pressure SI (HI2) is required to increase or maintain RCS inventory and AFW (AF0) is necessary for decay heat removal. Bleed and feed (OB1 + HI2) is used for decay heat removal if AFW fails.

ORT - Operator Actions to Trip Reactor

If top event AFM and top event MRT are unsuccessful (i.e., both automatic and manual trip of the reactor fail), the operator attempts to trip the reactor by either manually opening 2 of 2 480V bus supply breakers (which deenergizes the rod drive MG sets) within 2 minutes or locally tripping the reactor within 2 minutes. The methods of locally tripping the reactor are discussed in detail in Section 3.1.2/3.13. The time criterion of 2 minutes is based on the earliest time at which the SGs would dry out for an AWS at full power followed by a loss of feedwater.

OSP - Power Available

Following a loss of offsite power, sufficient AC emergency power to prevent core damage is provided by a 4.16 kV safeguards bus. Success for top event OSP is that at least one of the two 4.16 kV safeguards buses has power being supplied by an EDG. Further information on this top event is given in Section 3.1.2/3.11.

OS1 - Operator Action to Cooldown and Depressurize the RCS and Terminate SI Before the Ruptured SG Fills

Success of this action requires the operator to successfully complete the actions in EOP E-3 to stabilize RCS pressure less than the ruptured SG pressure (or the lowest SG relief valve setpoint) before the ruptured SG fills due to the addition of AFW and break flow. This normally requires three different high level operator actions: initial cooldown using the intact SG, RCS depressurization, and SI termination. For a design basis SGR as described in Section 14.2.4 of the USAR[14], these actions should be completed within about 30 minutes. Additional discussion on the OS1 top event is provided in Section 3.1.2/3.5.

The 30 minute time allotted to cool down and depressurize the RCS and terminate SI to prevent SG overfill is based on the following conservative analysis. A nominal initial SG fluid mass of 100,000 lbm is assumed. Combined with this initial fluid mass is the 120,000 lbm of reactor coolant discharged to the secondary through the ruptured tube 30 minutes following the tube rupture (assumed in the USAR analysis), and a conservatively high estimate of AFW addition of 37,500 lbm (i.e., 300 gpm for 15 minutes). Assuming saturation at the safety valve pressure setpoint of 1100 psia, the fluid density is 45.6 lbm/ft^3. The resulting volume of fluid in the ruptured SG after the 30 minutes of the SGTR transient is 5647 ft^3. The secondary volume of a Model 51 SG is 5706 ft^3.[37] Thus, it takes longer than 30 minutes to overfill the ruptured SG. To be conservative, it is assumed that the operator must cool down and depressurize the RCS and terminate SI within 30 minutes for OS1 to be successful.

It should be noted that a longer time to overfill is estimated assuming the SG fluid density is weighted more to the density of water in the CST (i.e., 62 lbm/ft^3). Also, in a more realistic scenario, the operator takes action to reduce AFW flow to the ruptured SG and to cool down and depressurize the RCS during the initial 30 minutes. If spaced out over the initial 30 minutes, these actions would significantly reduce the ruptured SG secondary inventory.

OS2 - Operator Action to Cooldown and Depressurize the RCS and Terminate SI After the Ruptured SG Fills

This event models the same actions as OS1. It is assumed, however, that the ruptured SG fills prior to the completion of these actions. The 1982 Ginna event is an example of this case.[38] For this SGR event, the ruptured SG filled and one of the safety valves briefly opened, possibly several times. Upon SI termination, the safety valve did re-seat and the recovery proceeded normally.

To simplify the evaluation of the OS2 top event, the actions described for OS1 are assumed to be completed before 60 minutes for success of OS2. This is a comparatively long time (compared to the time frame assumed for OS1 success). It is still short in comparison to the potential time to core uncovery in the event a secondary side safety valve were to stick open and eventually deplete the RWST (this time is on the order of 10 hours).

PPR - Primary Pressure Relief

This top event addresses the probability that pressurizer pressure relief capacity is adequate to limit the RCS pressure to less than 3200 psig. This is addressed only for loss of MFW AWS events; other conditions do not lead to a peak RCS pressure greater than 3200 psig.

The PPR description provided in Section 3.1.2/3.13 is expected to be a conservative basis for primary pressure relief.

SSV - Integrity Maintained or Restored in Ruptured SG

If one of the secondary relief valves sticks opens following overfill of the ruptured SG, the SGR recovery strategy becomes somewhat more complicated (the operator would transition to ECA-3.1 and possibly to ECA-3.2). Success is defined as all valves being closed to maintain or restore secondary integrity after the actions defined for OS1/OS2 (EOP-3 recovery) are complete. Failure of any one of the relief valves could result in secondary leakage large enough to require transition to the ECA-3.1 recovery procedure.

3.1.2/3.19 Scalar Development

This section describes the top level events that are not fault trees but instead are scalars. Scalars are used as top level events in five accident sequences. The scalars and how they were determined are provided below:

A. Steam Line Break (SLB) Scalars

A unique feature of the SLB initiating event is an excessive cooldown leading to return to criticality. The cooldown is more severe at low powers due to the greater stored energy in the steam generators. Reference 25 confirms that at 10% power, a return to criticality cannot occur even if both steam generators blow down and insert the maximum positive reactivity expected during a fuel cycle. A scalar representing the relative length of time that the reactor is operating below 10% power is therefore present in the SLB event tree.

It is assumed, since the turbine generator is generally placed on line at 8-12% power, that the amount of time with the generator on line is a close approximation of the amount of time above 10% power. The time at cold shutdown is subtracted off as well because an SLB at cold shutdown cannot cause core damage. The following statistics are as of February 29, 1992.

$$\text{(Fraction of time above 10\% power)} = \frac{\text{(Total hours of plant life - Total hours on-line - Total hours at cold shutdown)}}{\text{(Total hours of plant life - Total hours at cold shutdown)}}$$

$$\text{PWR-LOW} = \frac{155234 - 131313 - 18166}{155234 - 18166}$$

PWR-LOW = 4.20E-2

B. Station Blackout (SBO) Scalars

There are six scalars for the SBO event, all of which are calculated in Reference 43. These scalars represent the probability of offsite power restoration within 2, 9 and 11 hours and probability that the core will be uncovered due to reactor coolant pump seal LOCA after power restoration. Following is a summary of those scalars:

Time	Event	Scalar	Probability
2 hours	power non-restoration	AC2-FAIL	2.65E-1
2 hours	core uncovery	CCV-2	2.83E-2
9 hours	power non-restoration	ACX-9	4.10E-2
9 hours	core uncovery	CCV-9	7.618E-2
11 hours	power non-restoration	ACX-11	2.00E-2
11 hours	core uncovery	CCV-11	7.069E-2

C. ATWS Without Main Feedwater Scalars

An ATWS can be caused by three things, Failure of Reactor Protection System (RPS) circuitry, failure of the reactor trip breakers to open or mechanical failure of more than one rod control cluster assemblies (RCCAs). This distinction is important because if the RPS circuitry fails, a manual reactor trip (MRT) can still trip the reactor. On the other hand, if the failure is due to the trip breakers failing, MRT cannot trip the reactor.

A fault tree AFM was developed to assess the probability of each mode. The total failure probability for AFM is 2.34E-5. The probability of failure of reactor trip breakers is 1.11E-5.

The probability that an ATWS is due to the reactor trip breakers is therefore 1.11E-5/2.34E-5 = 4.74E-1. The probability that an ATWS is due to mechanical failure of more than one RCCAs is 7.7E-2. The probability that an ATWS is due to an RPS circuitry failure is (1-4.74E-1) - (7.7E-2) = 4.46E-1. Following is a summary of AWS scalars:

Mode	Probability
AFM-SIGNAL	4.46E-1
AFM-BREAKER	4.74E-1
AFM-RODS	7.70E-2

D. Interfacing Systems LOCA (ISL) Scalars

There are two scalars for the ISL event, both are calculated below. These scalars represent the probability of an RHR system pipe break and the failure of the RHR pumps seals.

The calculation to determine the probability for a RHR pipe break is as follows:

System:	RHR
Diameter:	10"
Type:	A312 Type 316 Schedule 40S
ID:	10.02"
t:	0.365"
P_{rcs}:	2250 psia

$$\text{Hoop Stress} = \frac{(2250)\ (10.02+0.365)}{(2)\ (0.365)}$$

$$= 31.95 \text{ ksi}$$

Using figure 4 on page F-12 of NUREG/CR-5102[19], the probability of pipe failure is approximately 5E-03.

The calculation for the RHR pumps seals is as follows:

$1 - P_{PIPE} = P_{SEALS}$
$1 - (5E\text{-}3) = 9.95E\text{-}1$

The following is a summary of those scalars:

Event	Scalar	Probability
Pipe Break	BR-PIPE	5E-3
Seal Failure	BR-SEAL	9.95E-1

E. Steam Generator For Tube Rupture (SGR) Scalars

There are two scalars for the SGR event, both are calculated below. These scalars represent the probability for success and failure for the closure of five steam generator safeties and one power operated relief valve.

The calculation of the probability of all six valves successfully closing is as follows:

$P_{SUC} = (5E\text{-}1)^6 = 1.6E\text{-}2$ 5E-1/valve

The calculation of any one valve failing to close is as follows:

$P_{FAIL} = 1 - (P_{SUC}) = 9.84E\text{-}1$

The following is a summary of those scalars:

Event	Scalar	Probability
1 Valve Fails	SSV-FAIL	9.84E-1
0 Valves Fail	SSV-SUC	1.6E-2

3.1.4 Support System Event Trees

The Kewaunee PRA uses fault tree linking to quantify accident sequences. Therefore, support system event trees were not 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 Interfaces

Plant damage states are defined as part of the Level 1 PRA and function as a link between the core damage accident sequences and the containment response and source term analysis that are part of the Level 2 PRA. The internal accident sequences modeled describe multiple scenarios that result in degraded core conditions of some sort. These scenarios are grouped or categorized in a manner such that the containment analysis can be performed. For a given initiating event, several event tree paths may be identical in terms of final plant conditions. This results from the fact that different combinations of success or failure of frontline systems often yield the same general plant condition. Hence, such identical outcomes representing the same final plant condition are combined.

In addition, all event tree accident sequence outcomes are grouped into several plant conditions or plant damage states on the basis of their similarity with respect to the following:

- Initiating event type (large LOCA, small LOCA, transient, containment bypass)
- Injection phase faults (core damage occurs early) or later recirculation phase faults (core damage occurs late)
- Containment safeguards operability (low pressure containment recirculation, containment spray, containment fan coolers and containment isolation).

Therefore, the output of the accident sequence modeling after categorization are the plant damage states and associated probabilities and are compactly represented in a matrix form. The matrix elements provide the conditional probability of each initiating event that could lead to one of the plant damage states.

For the plant damage state analysis, the WLINK Code System (WCS) was used. WCS uses the fault tree linking method to calculate the frequencies of the various plant damage states and to identify fault sequences (cutsets) in terms of component failure, operator actions, and other failures. WCS is used to quantify the core melt frequency for each initiating category and to obtain component level cutsets. Once the dominant core melt sequences are identified, they are further broken down to determine the frequency of core melt coupled with:

1) Success or failure of containment safeguard systems discussed below.
2) Timing of core melt.
3) RCS Pressure at time of core melt.

There are four systems used after a core melt to mitigate the off-site dose consequences. These systems are:

1) Low pressure recirculation
2) Containment air cooling

3) Containment spray
4) Containment isolation

3.1.5.1 Plant Damage States Definitions

In order to facilitate Level 2 analysis, each dominant core melt sequence was reported in a table with the following information provided with it:

1. Initiating event type

2. Event tree nodes modeled in fault tree linking

3. Status of low pressure recirculation system (LPR)
   Available or Failed (A or F)

4. Status of containment air cooling system (FCH)
   Available or Failed (A or F)

5. Status of containment sprays (ICS)
   Available or Failed (A or F)

6. Status of containment isolation (CI)
   Isolated (Available), Failed isolation, Bypassed (A or F or B)

7. Estimated timing of core melt
   Early core damage (within 0-4 hours)
   Late core damage (within 4-24 hours)
   (E or L)

8. RCS pressure at core damage
   High RCS pressure (400 psia or above) or Low RCS pressure (less than 400 psia).
   (HP or LP).

9. Frequency of Event Sequence

To capture the failure probabilities and mechanisms of the containment safeguards systems, the system fault trees for these systems are linked to the dominant core melt sequences identified by the quantification of the event trees. Containment isolation and low pressure recirculation are also included in this linking process. Thus, for each dominant core melt sequence, there may be up to 16 plant damage states, related to the various states of low pressure recirculation, containment coolers, sprays, and containment isolation.

3.1.5.2 Containment Safeguards Systems

The following defines the low pressure recirculation system, containment air cooling system (fan coolers), containment sprays, and containment isolation functions and their success criteria. Refer to section 4.1.1.2 for greater detail on systems associated with containment safeguards.

Low Pressure Recirculation System

The low pressure recirculation system provides cooling of debris after vessel failure by delivering water to the reactor upper plenum. This water then spills out of the failed vessel onto the debris in the reactor cavity. As it cools the debris, it boils off and gets cooled by the containment spray or the fan coolers. It then condenses and falls to the containment basement where it enters the recirculation sump and goes to the suction of the RHR pumps where it completes the loop.

Success Criterion

The success criterion for the low pressure recirculation system is one of two trains to operate for 24 hours.

Operator Actions: Low pressure injection actuation is either by the automatic safety injection signal or by the control room operator. The switchover to recirculation is implemented by the control room operator upon RWST low-low level alarm.

Containment Air Cooling System

The containment air cooling system provides cooling by recirculation of the containment air across air-to-water cooling coils. It consists of four fan coil units with 2 parallel units in each of 2 redundant SW trains. Water supply to the coils is from the SW system. Three fan coil units are normally operated. The fans normally operate with the cooling water flow to the coils at full flow. The fans are automatically started and the SW flow to the coils is automatically increased to full flow upon receipt of an SI signal.

Success Criterion

The success criterion for the containment air cooling system is two of four containment coolers operating at full flow for 24 hours.

This basis for this success criterion is provided in the USAR.[14]

Operator Actions: None needed.

Containment Spray System

The internal containment spray (ICS) system contains two separate trains of equal capacity, each independently capable of meeting the design bases. Each train includes a containment spray pump, spray header and nozzles, a common spray additive tank, containment recirculation sump screens, valves and necessary piping, instrumentation and control.

Success Criteria

The success criteria are as follows:

At least one containment spray pump delivering full flow from the RWST to the ring header and nozzles, with sodium hydroxide added to the injected water during the injection phase.

At least one containment spray pump delivering full flow from the containment sump to the ring header and nozzles after the switchover to the recirculation is implemented by the operator to take suction from the containment sump when the RWST low-low level alarm is received.

Both of these criteria must be satisfied for ICS to be considered available.

The total mission time for both the injection and recirculation phases of containment spray operation is 24 hours.

Operator Actions: Containment spray actuation is either by the automatic containment pressure signal, or by the control room operator. The switchover to recirculation is implemented by the control room operator upon RWST low-low level alarm.

Containment Isolation System

Containment isolation, as used in this report, refers to the closure of containment penetrations to limit the release of radioactive fluids following an accident. The failure of lines penetrating containment that result in the loss of reactor coolant inventory from the containment, which can potentially lead to a severe accident are considered under containment bypass failures, as opposed to containment isolation failures.

Success Criteria

IDCOR studies[39] have shown that significant fission product releases to the environment will occur only under the following conditions:

1. The line penetrating containment is greater than 2 inches in diameter,

2. The line penetrating containment directly communicates with either the containment atmosphere or the reactor coolant system during any severe accident scenario, and

3. The line-penetrating containment is not part of a 'closed' system outside of containment, capable of withstanding severe accident conditions.

These three conclusions of the IDCOR studies become a screening criterion for the identification of those containment penetrations requiring further evaluation to determine the potential for failure of containment isolation.

The containment fluid system and mechanical penetrations are categorized into three groups:

1. Those that are administratively controlled and required to be closed during operation (Category A), and

2. Those that can be open during normal operation and are required to change position upon receipt of an isolation signal (Category B), and

3. Those that are normally open following a containment isolation signal (Category C).

For the penetrations that fit into the above categories, the success criteria for containment isolation require that:

1. Those penetrations in Category A be in the closed position at the time of the accident (i.e. they should be closed during normal operation),

2. Those penetrations in Category B have moved to the closed position, if they were open, upon receipt of an isolation signal, and

3. Those penetrations in Category C are sealed from the atmosphere by properly functioning check valves and/or motor operated valves if the system is unavailable or not operating.

Operator Actions: Containment isolation is either automatic or by the control room operator. Operator actions are determined by those automatic actions which did not occur.

Scalars

Due to the large number of sequences involved in the plant damage state calculation, fault trees are linked together prior to the sequence calculation. In the Transients With Main Feedwater (TRA) calculation, for example, the fault trees AF3, OM2 and OB2 would be multiplied together 16 times. In this calculation, they were multiplied together once and given the name FB1. Similarly, each combination of containment safeguards has its own fault tree. IFC, for example, is the failure of ICS, FCH, and CI.

The one area in which the core melt sequences and plant damage state sequences do not coincide is the station blackout. In the station blackout, there are scalars called AC2-FAIL, ACX-9 and ACX-11 that represent the probability of restoring power in 2, 9, and 11 hours respectively. If these nodes fail, core melt occurs. Level 2 analyses have shown, however, that if power is restored within 24 hours and containment safeguards systems are available, containment failure does not occur. Therefore, for the purposes of this plant damage state calculation, AC2-FAIL, ACX-9 and ACX11 have been revised to be AC2-24, AC9-24 and AC11-24 representing the probability of regaining power in the windows of 2-24, 9-24 and 11-24 hours respectively. Another scalar, ACX-24, represents the probability of not regaining power in a 24 hour period. The failure of ACX-24 always results in containment failure. The above scalars are all based on Reference 24 and have the following values:

AC2-24 = 2.55E-01
AC9-24 = 3.10E-02
AC11-24 = 1.00E-02
ACX-24 = 1.00E-02

3.1.5.3 List of Plant Damage States

Table 3.1.5-1 provides the List of Plant Damage States for the Kewaunee PRA.

TABLE 3.1.5-1

LIST OF PLANT DAMAGE STATES

PDS#	EVENT SEQUENCE	LP RECIRC	FAN COOLERS	SPRAYS	CONT. ISOL.	TIME	RCS PRESS	FRE-QUENCY
LLO-1	SYS-ACC	A	A	A	A	E	L	1.00E-07
LLO-2	SYS-LI1	F	A	F	A	E	L	4.57E-07
LLO-3	SYS-LI1	F	F	F	A	E	L	6.52E-10
LLO-4	SYS-LR1	F	A	A	A	E	L	4.90E-09
LLO-5	SYS-LR1	F	A	F	A	E	L	1.35E-06
LLO-6	SYS-LR1	F	F	F	A	E	L	3.16E-08
MLO-1	SYS-HI0 SYS-OP1	A	A	A	A	E	H	1.13E-08
MLO-2	SYS-HI0 SYS-OP1	F	F	F	A	E	H	1.99E-09
MLO-3	SYS-HI0 SYS-LI2	F	A	F	A	E	L	5.35E-07
MLO-4	SYS-H10 SYS-LI2	F	A	F	F	E	L	7.12E-10
MLO-5	SYS-HI0 SYS-LI2	F	F	F	A	E	L	2.63E-09
MLO-6	SYS-HI0 SYS-LR1	F	A	F	A	L	L	1.22E-08
MLO-7	SYS-HI0 SYS-LR1	F	F	F	A	L	L	1.47E-07
MLO-8	SYS-HR0 SYS-LR2	F	A	A	A	L	L	1.83E-09
MLO-9	SYS-HR0 SYS-LR2	F	A	F	A	L	L	7.42E-06
MLO-10	SYS-HR0 SYS-LR2	F	A	F	F	L	L	1.57E-09
MLO-11	SYS-HR0 SYS-LR2	F	F	F	A	L	L	2.97E-09
SLO-1	SYS-HI2 SYS-OP2	A	A	A	A	E	H	1.20E-08

TABLE 3.1.5-1

LIST OF PLANT DAMAGE STATES

PDS#	EVENT SEQUENCE	LP RECIRC	FAN COOLERS	SPRAYS	CONT. ISOL.	TIME	RCS PRESS	FRE-QUENCY
SLO-2	SYS-HI2 SYS-OP2	F	F	F	A	E	H	4.72E-09
SLO-3	SYS-HI2 SYS-LI2	F	A	F	A	E	L	1.17E-06
SLO-4	SYS-HI2 SYS-LI2	F	A	F	F	E	L	1.69E-09
SLO-5	SYS-HI2 SYS-LI2	F	F	F	A	E	L	5.85E-09
SLO-6	SYS-HI2 SYS-LR1	F	A	F	A	L	L	4.40E-08
SLO-7	SYS-HI2 SYS-LR1	F	F	F	A	L	L	3.18E-07
SLO-8	SYS-ES1 SYS-HR1 SYS-LR2	F	A	A	A	L	L	2.09E-09
SLO-9	SYS-ES1 SYS-HR1 SYS-LR2	F	A	F	A	L	L	1.21E-05
SLO-10	SYS-ES1 SYS-HR1 SYS-LR2	F	A	F	F	L	L	3.52E-09
SLO-11	SYS-ES1 SYS-HR1 SYS-LR2	F	F	F	A	L	L	5.85E-09
SGR-1	SYS-AF1 SYS-OM1	F*	N/A	F*	B	L	H	6.02E-07
SGR-2	SYS-ISO SYS-EC3	F*	N/A	F*	B	L	H	1.19E-07
SGR-3	SYS-HI1 SYS-OS1 SYS-SSV	F*	N/A	F*	B	L	H	1.93E-07
SGR-4	SYS-OS1 SYS-SSV SYS-EC4	F*	N/A	F*	B	L	H	4.31E-06

TABLE 3.1.5-1

LIST OF PLANT DAMAGE STATES

PDS#	EVENT SEQUENCE	LP RECIRC	FAN COOLERS	SPRAYS	CONT. ISOL.	TIME	RCS PRESS	FRE-QUENCY
SGR-5	SYS-OS1 SYS-OS2	F*	N/A	F*	B	L	H	5.96E-08
VEF-1		A	A	A	A	E	H	3.00E-07
VEF-2		A	A	F	A	E	H	1.45E-09
VEF-3		A	F	A	A	E	H	1.79E-10
VEF-4		F	A	A	A	E	H	1.26E-10
VEF-5		F	A	F	A	E	H	6.37E-10
ISL-1	PIPE-BREAK	N/A	N/A	N/A	B	E	L	7.40E-09
TRA-1	SYS-AF3 SYS-OM2 SYS-OB2	A	A	A	A	E	H	2.02E-06
TRA-2	SYS-AF3 SYS-OM2 SYS-OB2	A	A	A	F	E	H	1.78E-10
TRA-3	SYS-AF3 SYS-OM2 SYS-OB2	A	A	F	A	E	H	1.12E-08
TRA-4	SYS-AF3 SYS-OM2 SYS-OB2	A	F	A	A	E	H	1.03E-09
TRA-5	SYS-AF3 SYS-OM2 SYS-OB2	F	A	A	A	E	H	7.31E-10
TRA-6	SYS-AF3 SYS-OM2 SYS-OB2	F	A	F	A	E	H	4.59E-09
TRA-7	SYS-AF3 SYS-OM2 SYS-OB2	F	F	F	A	E	H	6.74E-07
TRA-8	SYS-AF3 SYS-OM2 SYS-HR1	A	A	A	A	L	L	4.40E-09

TABLE 3.1.5-1

LIST OF PLANT DAMAGE STATES

PDS#	EVENT SEQUENCE	LP RECIRC	FAN COOLERS	SPRAYS	CONT. ISOL.	TIME	RCS PRESS	FRE-QUENCY
TRA-9	SYS-AF3 SYS-OM2 SYS-HR1	F	A	F	A	L	L	9.23E-09
TRA-10	SYS-AF3 SYS-OM2 SYS-HR1	F	F	F	A	L	L	2.37E-10
TRA-11	SYS-CHG SYS-CCW	F	A	F	A	L	H	3.21E-08
TRA-12	SYS-CHG SYS-CCW	F	F	F	A	L	H	2.35E-08
TRS-1	SYS-AF3 SYS-OB2	A	A	A	A	E	H	2.20E-07
TRS-2	SYS-AF3 SYS-OB2	A	A	F	A	E	H	2.30E-10
TRS-3	SYS-AF3 SYS-OB2	F	A	A	A	E	H	4.51E-10
TRS-4	SYS-AF3 SYS-OB2	F	A	F	A	E	H	6.86E-09
TRS-5	SYS-AF3 SYS-OB2	F	F	F	A	E	H	3.06E-08
TRS-6	SYS-AF3 SYS-HR1	A	A	A	A	L	L	7.62E-08
TRS-7	SYS-AF3 SYS-HR1	A	A	F	A	L	L	1.05E-08
TRS-8	SYS-AF3 SYS-HR1	F	A	F	A	L	L	1.14E-07
SLB-1	PWR-LOW SYS-IS1 SYS-HI3	A	A	A	A	E	H	2.85E-08
SLB-2	PWR-LOW SYS-IS1 SYS-HI3	F	A	A	A	E	H	4.31E-10

TABLE 3.1.5-1

LIST OF PLANT DAMAGE STATES

PDS#	EVENT SEQUENCE	LP RECIRC	FAN COOLERS	SPRAYS	CONT. ISOL.	TIME	RCS PRESS	FRE-QUENCY
SLB-3	PWR-LOW SYS-IS1 SYS-HI3	F	A	F	A	E	H	1.78E-08
SLB-4	PWR-LOW SYS-IS1 SYS-HI3	F	F	F	A	E	H	1.76E-09
SLB-5	SYS-HI3 SYS-AF1 SYS-OM1	F	A	F	A	E	H	6.79E-10
SLB-6	SYS-HI3 SYS-AF1 SYS-OM1	F	F	F	A	E	H	2.39E-08
SLB-7	SYS-AF1 SYS-OM1 SYS-OB4	A	A	A	A	E	H	2.71E-08
LSP-1	SYS-AF3 SYS-OB5	A	A	A	A	E	H	1.83E-06
LSP-2	SYS-AF3 SYS-OB5	A	A	F	A	E	H	8.66E-08
LSP-3	SYS-AF3 SYS-OB5	A	F	A	A	E	H	3.87E-09
LSP-4	SYS-AF3 SYS-OB5	F	A	A	A	E	H	4.27E-09
LSP-5	SYS-AF3 SYS-OB5	F	A	F	A	E	H	2.67E-07
LSP-6	SYS-AF3 SYS-OB5	F	F	F	A	E	H	6.54E-08
LSP-7	SYS-AF3 SYS-HR1	A	A	A	A	L	L	3.49E-07
LSP-8	SYS-AF3 SYS-HR1	A	A	F	A	L	L	3.22E-08
LSP-9	SYS-AF3 SYS-HR1	F	A	F	A	L	L	6.55E-07

TABLE 3.1.5-1

LIST OF PLANT DAMAGE STATES

PDS#	EVENT SEQUENCE	LP RECIRC	FAN COOLERS	SPRAYS	CONT. ISOL.	TIME	RCS PRESS	FRE-QUENCY
LSP-10	SYS-CHG SYS-CCW	F	A	F	A	L	H	1.23E-06
LSP-11	SYS-CHG SYS-CCW	F	F	F	A	L	H	4.39E-09
SBO-1	SYS-AF2 AC2-24	A	A	A	A	E	H	1.25E-05
SBO-2	SYS-AF2 AC2-24	A	A	A	F	E	H	4.39E-09
SBO-3	SYS-AF2 AC2-24	A	A	F	A	E	H	9.33E-08
SBO-4	SYS-AF2 AC2-24	A	F	A	A	E	H	8.01E-09
SBO-5	SYS-AF2 AC2-24	F	A	A	A	E	H	5.99E-09
SBO-6	SYS-AF2 AC2-24	F	A	F	A	E	H	3.44E-08
SBO-7	SYS-AF2 AC2-24	F	F	F	A	E	H	6.60E-10
SBO-8	SYS-CHB SYS-AF2 CCV-2	A	A	A	A	E	H	1.79E-07
SBO-9	SYS-CHB SYS-ORI	A	A	A	A	L	H	1.53E-07
SBO-10	SYS-CHB SYS-ORI	F	A	A	A	L	H	5.06E-10
SBO-11	SYS-CHB SYS-ORI	F	A	F	A	L	H	1.02E-08
SBO-12	SYS-CHB SYS-ORI	F	F	F	A	L	H	2.94E-09
SBO-13	SYS-CHB SYS-HR1	A	A	A	A	L	H	1.01E-07
SBO-14	SYS-CHB SYS-HR1	A	A	F	A	L	H	1.56E-08

TABLE 3.1.5-1

LIST OF PLANT DAMAGE STATES

PDS#	EVENT SEQUENCE	LP RECIRC	FAN COOLERS	SPRAYS	CONT. ISOL.	TIME	RCS PRESS	FRE-QUENCY
SBO-15	SYS-CHB SYS-HR1	F	A	F	A	L	H	1.60E-07
SBO-16	SYS-OCD AC9-24	A	A	A	A	L	H	4.91E-08
SBO-17	SYS-CHB SYS-OCD CCV-9	A	A	A	A	L	H	1.51E-08
SBO-18	AC11-24	A	A	A	A	L	H	4.35E-06
SBO-19	AC11-24	A	A	A	F	L	H	1.80E-09
SBO-20	AC11-24	A	A	F	A	L	H	3.39E-08
SBO-21	AC11-24	A	F	A	A	L	H	2.86E-09
SBO-22	AC1I-24	F	A	A	A	L	H	2.08E-09
SBO-23	AC1I-24	F	A	F	A	L	H	1.26E-08
SBO-24	AC1I-24	F	F	F	A	L	H	2.71E-10
SBO-25	SYS-CHB CCV-11	A	A	A	A	L	H	4.01E-06
SBO-26	SYS-CHB CCV-11	A	A	A	F	L	H	8.47E-10
SBO-27	SYS-CHB CCV-11	A	A	F	A	L	H	2.38E-08
SBO-28	SYS-CHB CCV-11	A	F	A	A	L	H	1.90E-09
SBO-29	SYS-CHB CCV-11	F	A	A	A	L	H	1.38E-09
SBO-30	ACX-24	F*	F*	F*	A	L	H	4.35E-06
SBO-31	ACX-24	F*	F*	F*	F	L	H	1.80E-09
AWS-1	AFM-RODS SYS-PPR	A	A	A	A	E	H	7.10E-09
AWS-2	AFM-BREAKER SYS-AFG	A	A	A	A	E	H	1.24E-08

TABLE 3.1.5-1

LIST OF PLANT DAMAGE STATES

PDS#	EVENT SEQUENCE	LP RECIRC	FAN COOLERS	SPRAYS	CONT. ISOL.	TIME	RCS PRESS	FRE-QUENCY
AWS-3	AFM-BREAKER SYS-PPR	A	A	A	A	E	H	4.40E-08
SWS-1	SYS-AF6	F*	F*	F*	A	E	H	4.21E-07
CCS-1	SYS-CHG	F*	A	F*	A	L	H	1.27E-08
CCS-2	SYS-AF3 SYS-OM2	F*	A	F*	A	E	H	1.56E-08
TDC-1	SYS-AF4 SYS-OM4 SYS-OB3	A	A	A	A	E	H	1.18E-07
TDC-2	SYS-AF4 SYS-OM4 SYS-OB3	A	A	F	A	E	H	4.53E-09
TDC-3	SYS-AF4 SYS-OM4 SYS-OB3	A	F	A	A	E	H	2.56E-10
TDC-4	SYS-AF4 SYS-OM4 SYS-OB3	F	A	F	A	E	H	2.01E-09
TDC-5	SYS-AF4 SYS-OM4 SYS-OB3	F	F	F	A	E	H	9.28E-08
INA-1	SYS-AF5 SYS-OM3	A	A	A	A	E	H	7.41E-10
INA-2	SYS-AF5 SYS-OM3	F	F	F	A	E	H	2.08E-06
INA-3	SYS-AF5 SYS-OM3	F	F	F	F	E	H	6.79E-10

F* Containment safeguards function unavailable due to the nature of the core melt sequence.

3.2 System Analysis

There are two categories of systems modeled in the Kewaunee PRA: front line systems and support systems. Frontline systems for the Level 1 portion of the PRA are those that are used to maintain reactivity control, reactor coolant system inventory, and reactor coolant system heat removal capability. Frontline systems for the Level 2 portion of the PRA are those that are used for maintaining containment integrity, containment heat removal capability, and containment pressure control. Support systems are those that are necessary for the successful operation of the frontline systems. Support systems directly support the frontline systems, other support systems, or a combination of both frontline and support systems.

Individual system notebooks were developed for each of the following systems:

auxiliary feedwater
component cooling
containment air cooling
containment isolation
electric power (4160/480/120V AC, 120V DC)
high pressure safety injection
internal containment spray
low pressure safety injection
main feedwater (including condensate)
reactor protection
service water

Another notebook was developed to include descriptions for all of the miscellaneous system fault trees, many of which are fault trees that include operator actions and the equipment associated with those actions. The systems included in the miscellaneous systems notebook are listed in Table 3.2-1. The system notebooks contain the system function, description, necessary support systems, instrumentation and control, test and maintenance, references to applicable technical specifications, normal and accident operation, success criteria, operating experience, possible accident initiators arising from the system, operator interface, reactor protection, engineered safety features actuation interface, fault tree assumptions, dependency and common cause failures, and the system fault trees. System and fault tree descriptions for each of the frontline and support systems are described in section 3.2.1. The drawings supplied with these descriptions are provided for information only as they do not reflect recent design modifications made to the plant. The fault tree development and methodology are discussed in more detail in section 3.2.2. System dependencies are shown in section 3.2.3.

TABLE 3.2-1

MISCELLANEOUS NODE DESCRIPTION TABLE

SYSTEM/OPERATOR ACTION	FAULT TREE
FAILURE OF AUXILIARY BUILDING BASEMENT COOLING	ABBC
NO FLOW FROM 1 OF 2 CHARGING PUMPS	CHB
NO FLOW FROM 1 OF 3 CHARGING PUMPS	CHG
NO FLOW FROM 1 OF 3 CHARGING PUMPS	CHS
FAILURE TO COOLDOWN AND DEPRESSURIZE RCS	EC3
FAILURE TO COOLDOWN AND DEPRESSURIZE RCS	EC4
FAILURE TO COOLDOWN AND DEPRESSURIZE RCS	ES1
FAILURE OF ISOLATION AFTER SLB EVENT	IS1
FAILURE TO ISOLATE 1 OF 2 STEAM GENERATORS	ISO
FAILURE TO MAINTAIN LONG TERM SHUTDOWN	LTS
MANUAL REACTOR TRIP FAILURE	MRT
FAILURE TO ESTABLISH BLEED AND FEED	OB1
FAILURE TO ESTABLISH BLEED AND FEED	OB2
FAILURE TO ESTABLISH BLEED AND FEED	OB3
FAILURE TO ESTABLISH BLEED AND FEED	OB4
FAILURE TO ESTABLISH BLEED AND FEED	OB5
FAILURE TO COOLDOWN RCS	OCD
FAILURE TO COOLDOWN AND DEPRESSURIZE RCS	OP1
FAILURE TO COOLDOWN AND DEPRESSURIZE RCS	OP2

TABLE 3.2-1

MISCELLANEOUS NODE DESCRIPTION TABLE

SYSTEM/OPERATOR ACTION	FAULT TREE
FAILURE TO RESTORE RCS INVENTORY	ORI
FAILURE TO DEENERGIZE 480V BUSSES 33 AND 43	ORT
FAILURE TO COOL DOWN AND DEPRESSURIZE RCS	OS1
FAILURE TO COOLDOWN AND DEPRESSURIZE RCS AND STOP SI	OS2
FAILURE TO TERMINATE DEPRESSURIZATION	OSD
FAILURE OF ON-SITE POWER	OSP
FAILURE TO THROTTLE SI FLOW	OSR
FAILURE TO ISOLATE RHR PUMPS	OIP
FAILURE TO DEPRESSURIZE RCS	PPR
NO RESIDUAL HEAT REMOVAL FLOW FOR RCS COOLDOWN	RHR
RHR PUMP RELIEF VALVES FAIL TO RECLOSE	RVC
INTEGRITY NOT MAINTAINED OR RESTORED IN RUPTURED STEAM GENERATOR	SSV
LOSS OF INSTRUMENT AIR	IAS
LOSS OF INSTRUMENT AIR FOR LSP	IASP
LOSS OF INSTRUMENT AIR TERMINATION FOR LSP	IASPT
LOSS OF INSTRUMENT AIR TERMINATION	IAST
LOSS OF INSTRUMENT AIR TERMINATION	IASTA
LOSS OF INSTRUMENT AIR TERMINATION	IASTB

TABLE 3.2-1

MISCELLANEOUS NODE DESCRIPTION TABLE

SYSTEM/OPERATOR ACTION	FAULT TREE
LOSS OF INSTRUMENT AIR FOR TDC	IASD

3.2.1 System Descriptions

3.2.1.1 Auxiliary Feedwater System

Function

The auxiliary feedwater (AFW) system is an engineered safeguard system designed to supply high pressure feedwater to the steam generators (SGs) following an interruption of the main feedwater (FW) system supply. Periods when the AFW system may be required for the removal of residual heat from the core include startup, safety injection, failure of the FW system and for long term decay heat removal. AFW system operation prevents the release of reactor coolant through the pressurizer safety valves and protects the core by maintaining a heat sink for the removal of residual heat from the core by heat transfer in the SGs.

The AFW system is in a standby condition during normal plant operation. The system is aligned to provide heat removal capability in the event main feedwater (MFW) system supply is interrupted. The motor driven (MD) AFW pumps automatically start on lo-lo SG level in either SG, an SI, a blackout, or an opening of both of the MFW pump breakers. The turbine driven (TD) AFW pump automatically starts on lo-lo SG level in both SGs or an undervoltage on buses 1 and 2. The pumps are capable of automatically starting and delivering full flow within one minute after the signal for pump actuation. The system is also manually started to use during normal plant heatup, cooldown, hot standby and hot shutdown operation.

The AFW pumps normally take a suction from the Condensate Storage Tanks (CSTs). The CSTs contain a minimum volume of 30,000 gallons for use by the AFW System. The minimum volume is based on having sufficient water for 90 minutes at hot shutdown with a suitable margin to prevent a loss of net positive suction head prior to switching AFW pump suction to the SW system. There is a Technical Specification change in process to increase the minimum CST volume to a 4 hour supply for coping with a station blackout event.

Description

The AFW system delivers feedwater from the CSTs, or the service water (SW) system, to the MFW piping at a location near the SG inlet. The system consists of the AFW pumps, associated valves, piping and control systems. A detailed system piping and instrumentation drawing (P&ID) is shown in Figure 3.2-1 and Figure 3.2-2. The system consists of one steam TDAFW pump capable of delivering AFW to either or both SGs and two MDAFW pumps, one for each SG, which are interconnected on the discharge side by a common header. Each MDAFW pump is capable of supplying feedwater to both SGs via normally open motor operated valves (MOV) AFW-10A/10B. The two MDAFW pumps are energized from separate safeguards buses. Motive force for the TDAFW pump is provided from the main steam system via normally open MOVs MS-100A/100B, check valves MS-101A/101B and normally closed MOV MS-102, which receives an open signal on TDAFW pump actuation.

The normal feedwater supply to the AFW pumps is a common header from the two cross connected 75,000 gallon CSTs, through locked open isolation valve MU-300 and check valves MU-301, MU-311A/311B/311C. The backup water supply to the pumps is provided by the SW system. The MDAFW pumps are aligned to their respective SW header via MOVs SW-601A/601B, while the TDAFW pump is aligned to both SW headers via SW-502 and check valves SW-501A/501B. The SW system MOVs are normally closed.

Operator action is required to switch AFW pump suction from the CSTs to the SW system. When CST levels reach 4%, as indicated in the control room, the operators are instructed to open MOVs SW-601A/601B/502 to align SW to the AFW pumps. This action must be performed over a short time span to preclude the loss of AFW pump net positive suction head. The SW pumps deliver an unlimited makeup source to the AFW pumps from Lake Michigan.

<u>Fault Trees</u>

Eight fault trees were developed for the AFW system.

The success of the AFW system is based on its ability to cool the reactor coolant system (RCS), via the SGs, to approximately 300 - 350°F. Thereafter, the residual heat removal (RHR) system is capable of providing the necessary heat sink.

The AFW system provides for the following during abnormal conditions:

1. prevents thermal cycling of the SG tube sheet upon loss of the main FW pump;

2. removes residual heat from the RCS until the temperature drops below 300-350°F and RHR system is capable of providing the necessary heat sink,

3. maintains a head of water in the SG following a loss of coolant accident (LOCA).

The second and third of these three functions are safeguards. The feedwater flow rate required for removing residual heat is approximately 200 gpm. The eight fault trees are:

AF0 - This fault tree was developed for medium break LOCA, small break LOCA and interfacing system LOCA events. The basis for this fault tree is that both SGs, feed lines and main steam lines are intact. The initial suction source for the AFW pumps are the CSTs. When this water supply is depleted, the SW system is aligned to the pump suction by operator action. The defined mission time is 24 hours. The success criterion for AF0 is 200 gpm flow to 1 of 2 SGs from 1 of 3 AFW pumps.

AF1 - The basis for this fault tree is that one of two SGs is ruptured or faulted and applies to steam generator tube rupture and steam line break events. Under these conditions, the operators are directed by emergency operating procedures to isolate the effected SG. The suction sources are the CSTs and the SW system. The mission time is also 24 hours. The success criteria for

AF1 is 200 gpm flow to the intact SG from 1 of 2 AFW pumps. For the two events it is assumed that steam generator A is intact and steam generator B is faulted. It is also assumed that the AFW pump B is unavailable due to SG isolation procedures.

AF2 - This fault tree is for a station blackout event. Under this condition, only the TDAFW pump and the CSTs would be available. The defined mission time is 8 hours and is based upon the volume of water available in the condensate system and the need to cool down to avoid a RXCP seal LOCA. The success criterion for AF2 is 200 gpm flow to 1 of 2 SGs from 1 of 1 AFW pumps.

AF3 - This fault tree is for the following events: Transients with MFW, transients without MFW, loss of offsite power and loss of CCW. The mission time is 24 hours. The success criterion for AF3 is 200 gpm flow to 1 of 2 SGs from 1 of 3 AFW pumps.

AF4 - This fault tree is for the loss of DC bus event where DC bus BRA-104 is assumed to have failed. The mission time is 24 hours. Loss of the DC bus BRA-104 prevents the operation of AFW pumps A and C. AFW pump A can be started by local operator action. The success criterion for AF4 is 200 gpm flow to 1 of 2 SGs from 1 of 2 AFW pumps.

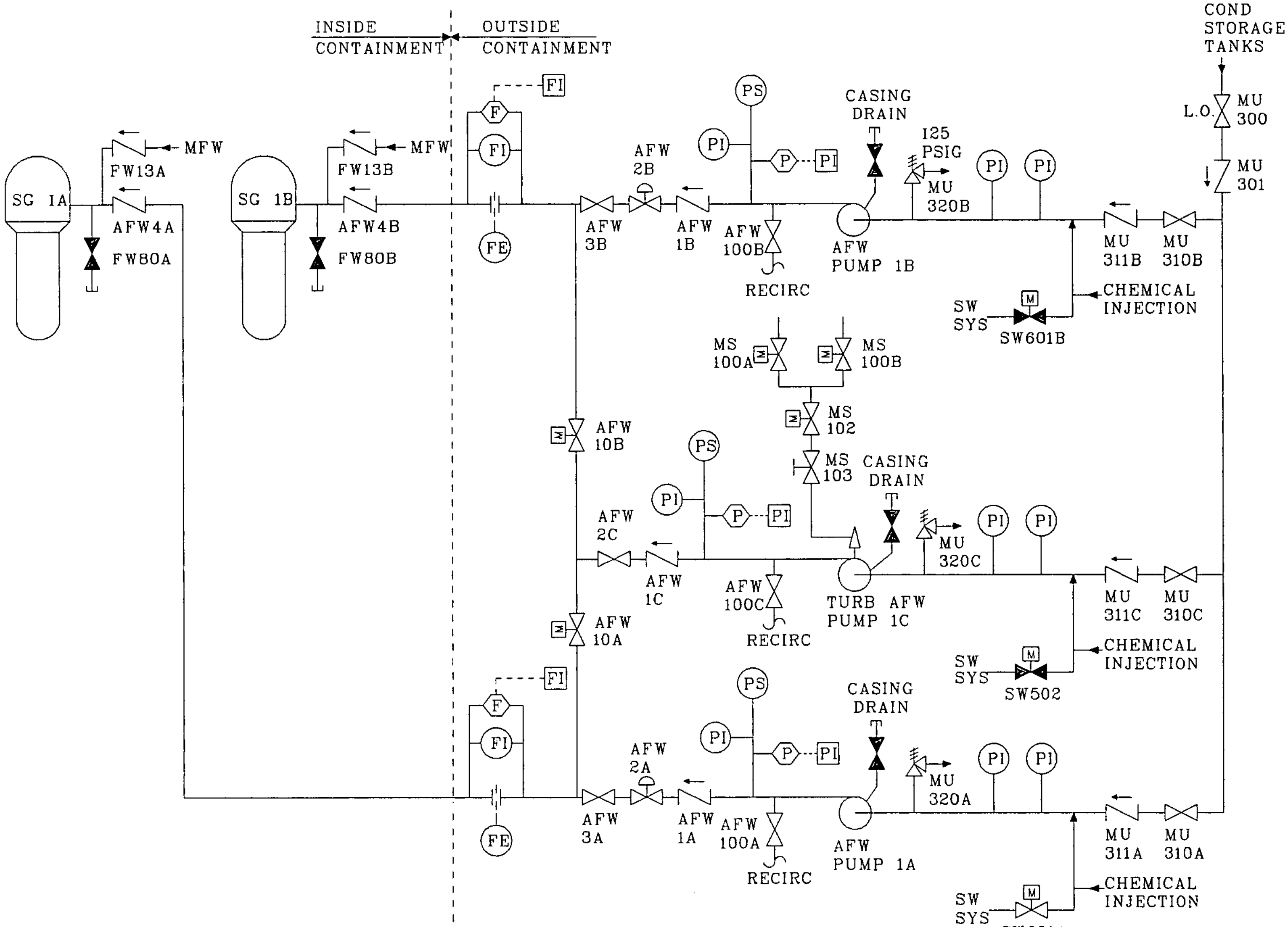
AF5 - This fault tree is for the loss of instrument air event. The mission time is 24 hours. The success criteria for AF5 is 200 gpm flow to 1 of 2 SGs from 1 of 3 AFW pumps.

AF6 - This fault tree is for the loss of SW system event. The mission time is 24 hours. It is assumed that the SW system is unavailable and that the CSTs are the only water source. The success criterion for AF6 is 200 gpm flow to 1 of 2 SGs from 1 of 3 AFW pumps.

AFG - This fault tree is for the ATWS event. The mission time is 24 hours. The success criteria for AFG is 400 gpm flow to 1 of 2 SGs from 2 of 3 AFW pumps.

FIGURE 3.2-1

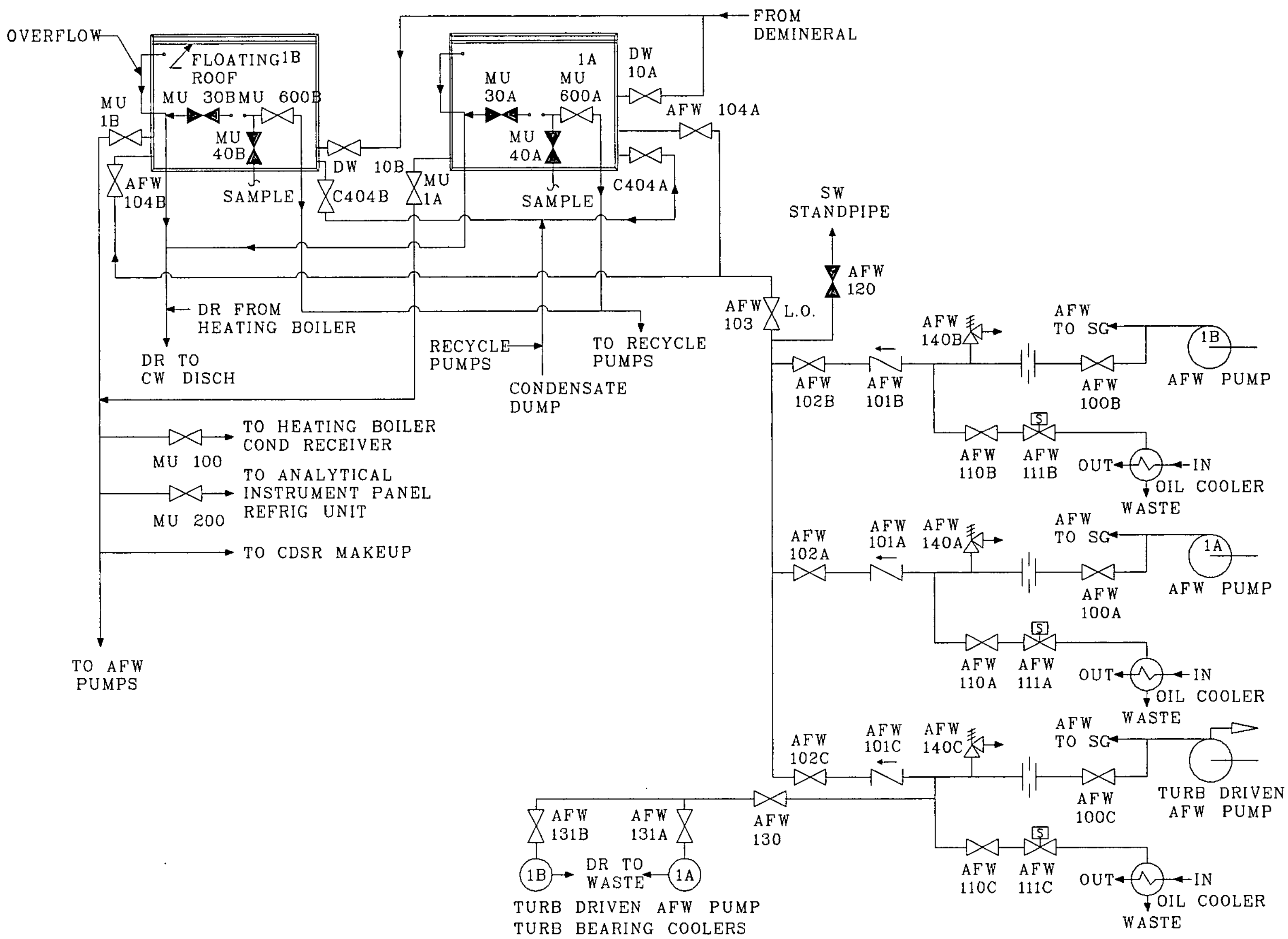
AUXILIARY FEEDWATER SYSTEM



P2-13
OPERM-205
REV. AH

172

FIGURE 3.2-2
AUXILIARY FEEDWATER RECIRCULATION



P2-14
OPERM-204
REV. GP

3.2.1.2 Component Cooling Water

Function

The component cooling water system (CCW) is designed to remove sensible heat from the reactor coolant system (RCS) via the residual heat removal system (RHR) during plant cooldown and shutdown. It also provides cooling of water taken from the containment sump during the recirculation phase of emergency core cooling. The CCW system cools the RCS letdown and reactor coolant pump (RXCP) leakoff flows to the chemical and volume control system (CVCS). The CCW system provides cooling for the RXCPs, low head safety injection pumps (RHR), high head safety injection (SI) pumps and the internal containment spray (ICS) pumps. The CCW system serves as an intermediate loop between systems processing or containing radioactive fluids and the service water system. The CCW system thereby minimizes the chance of contaminating SW with leakage from radioactive system coolers.

Description

The CCW system is a closed loop system with two CCW pumps in parallel supplying a common header which splits into two parallel CCW heat exchangers. The outlets of the CCW heat exchangers combine into a single supply header. The supply header provides cooling water to the serviced components via branch lines. The returns from the various serviced components connect into a single common return header which ends at the combined suction of the CCW pumps. The CCW surge tank connects to the CCW return header. The CCW surge tank supplies a volume to accommodate expansion and contraction in the system during temperature transients. The CCW pumps provide the necessary discharge head to deliver the required CCW flow throughout the system. The normal operating configuration is shown on Figures 3.2-3 and 3.2-4.

CCW pump A and the CCW heat exchangers are provided with general area cooling via the auxiliary building fan coil units A and B. CCW pump B is located in a separate room that has its own fan coil unit to meet the requirements of 10CFR50, Appendix R.

In July 1984 Westinghouse Electric Corporation informed WPSC and the NRC of a safety hazard associated with the CCW system. This involved the potential for overpressurizing the CCW system due to system inleakage or thermal expansion coincident with the closure of the CCW surge tank vent valve. A design change (DCR 1560) was subsequently implemented which removed the surge tank relief valve and installed a hard pipe connection from the tank to the waste holdup tank.

During normal plant operation one CCW pump is in operation with the second pump in standby. The operating pump provides approximately 2000 gpm for the operating loads on the system. This would include the boric acid evaporator and continuous supply to the SI, RHR and ICS pumps. The operating and standby pump status is shifted regularly to equalize operating time.

During normal plant operation valves CC-6A and CC-6B are open with CCW flow through both heat exchangers. Service water outlet valves from the CCW heat exchangers (SW-1300A and SW-1300B) are normally closed. CCW loop temperature control is accomplished by modulating an air operated valves (SW-1306A and B) in a bypass line around the outlet isolation valves described above.

Fault Trees

In the Kewaunee PRA models, CCW is both a front line and a support system. CCW is modeled as a front line system in the transients with and without main feedwater events (TRA and TRS), and the loss of offsite power event (LSP). In all three cases, CCW is modeled as a method of maintaining the reactor coolant pump seals in the event that charging (seal injection) is lost. There are two fault trees for CCW as a support system CCW and CCWP with the fault tree CCW also used as the front line system model.

CCW - CCW is the fault tree used as the front line system in TRA, TRS and LSP. It is also used as a support system for numerous other trees except for those associated with the LSP event. The success criterion for CCW is 1 of 2 CCW trains delivering flow to the CCW supply header, and the mission time is 24 hours. Each CCW train consists of one CCW pump and one CCW heat exchanger and all the valves and piping required to function during accident conditions. The CCW pump seals are self cooled; therefore, the unavailability of the cooling to the pump seals is not considered separate from the pump unavailability. The need for area cooling for CCW pump B is explicitly modeled because the pump is located in a small Appendix R room with a dedicated fan coil unit (FCU). It is assumed the pump fails if the FCU fails.

CCWP - CCWP is used as a support system in the LSP event. It is similar to CCW except that no offsite power supplies are available. The success criterion and mission time for CCWP are the same as for CCW.

FIGURE 3.2-3
COMPONENT COOLING

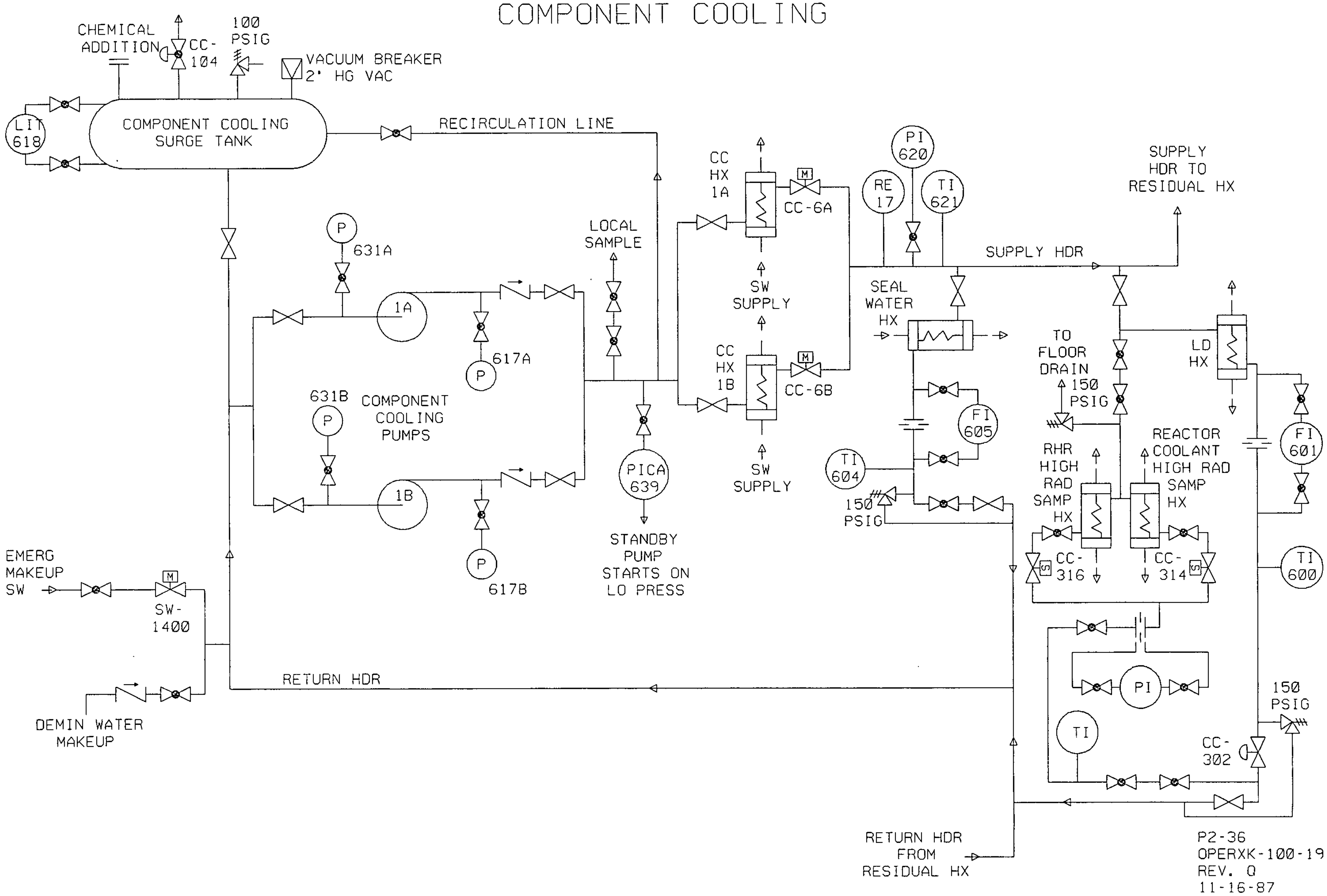
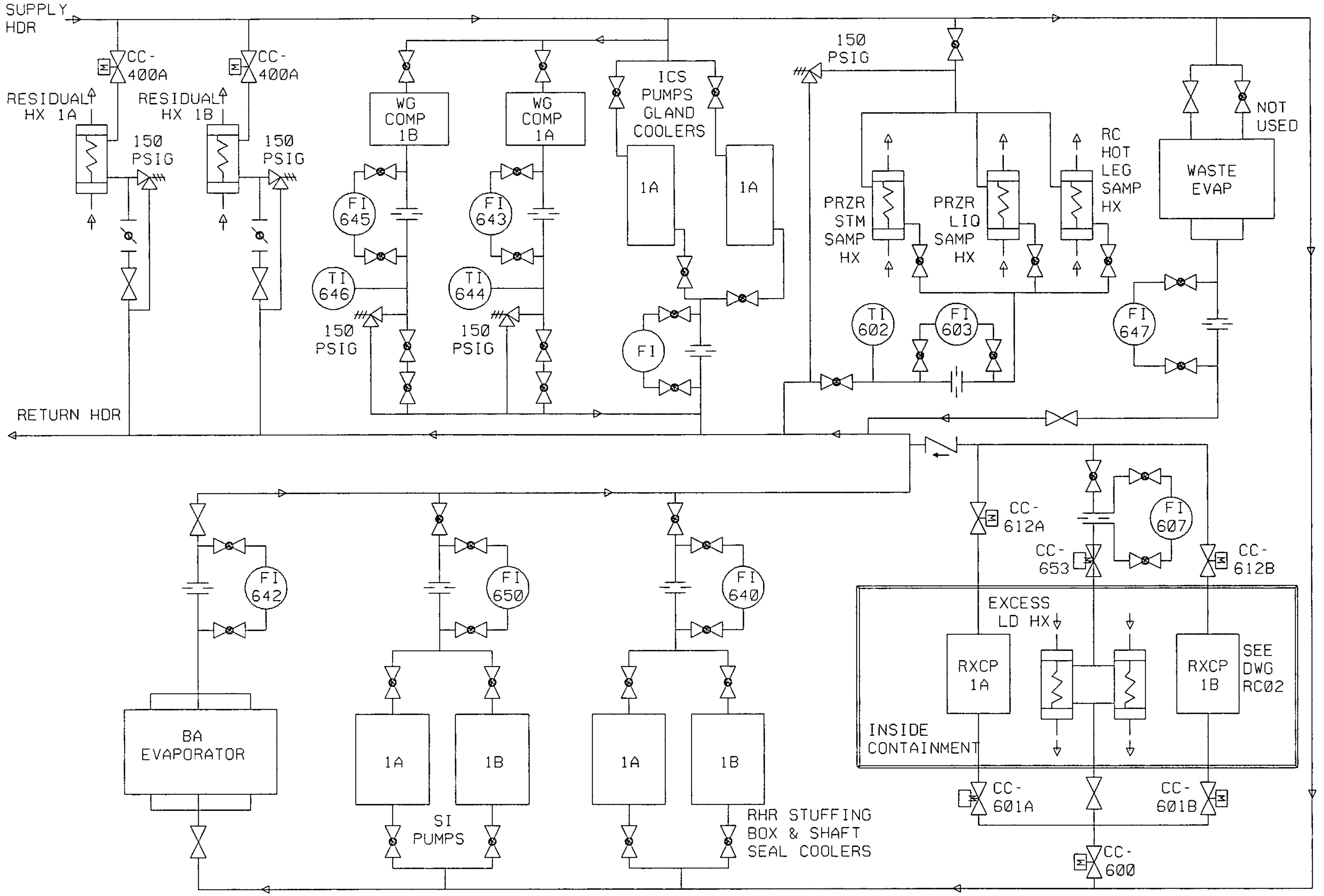


FIGURE 3.2-4 COMPONENT COOLING



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176

3.2.1.3 Containment Air Cooling

Function

The containment air cooling (CAC) system is a subsystem of the reactor building ventilation (RBV) system. The RBV system consists of several subsystems that operate together to cool and circulate containment air during all modes of plant operation; provide containment purge and vent capabilities; provide containment vacuum protection and post-loss-of-coolant accident (LOCA) hydrogen control.

During normal plant operation the CAC system provides the majority of air cooling and circulation in order to maintain containment air temperatures below 120°F. Post-LOCA, the CAC system is designed to remove sufficient heat from the containment vessel, following the initial pressure transient, to keep containment pressure from exceeding the design value of 46 psig at 268°F (100% relative humidity).

Description

The CAC system consists of four fan coil units (FCUs), duct distribution, emergency discharge dampers, associated instrumentation and controls and SW system support cooling. Figure 3.2-5 is a detailed piping and instrument diagram (P&ID) of the CAC system and Figure 3.2-6 is a detailed P&ID of the SW System.

The four FCUs are installed in pairs on opposite sides of the reactor vessel; FCU A and B are powered from 480V safeguards bus 51 and FCU C and D are powered from 480V safeguards bus 61. During power operation all four FCU are normally in operation to cool and circulate the containment air.

Each FCU has an air flow capacity of approximately 44,000 cfm under normal conditions. Air is drawn in from the area immediately around the FCUs, passes through the air side of the SW cooling coils and vane axial fan, and discharged to the distribution duct header. Each pair of FCUs is connected to a common ductwork which branches to direct the cooled air to the reactor coolant pump (RXCP) vaults and lower elevations in containment, and to the ring duct above the refueling floor. The ring duct has large registers which blow cool air over the refueling floor and supply the intake to the reactor vessel gap and neutron detector cooling fans.

During RXCP operation, at least one FCU on each train must be in operation to cool the RXCP motors.

Immediately downstream of each FCU is a passive pressure equalizing damper. This damper opens and allows pressure to equalize if the duct external pressure exceeds the internal pressure. Located a short distance from the pressure equalizing dampers are the emergency discharge dampers RBV-150A/150B/150C/150D. These dampers are normally closed and receive an open signal when containment pressure exceeds approximately 3.85 psig. These dampers open to ensure design basis flow rate through the FCUs in the event the remaining ductwork collapses during a LOCA pressure transient.

The heat sink for the FCUs is the SW system. Each FCU has a separate SW supply and return header inside of containment. On the supply line manual isolation valves SW-900A/900B/900C/900D are located outside of containment and check valves SW-901A/901B/901C/901D are located immediately inside of containment.

Downstream of the FCUs the SW return flows in separate headers through the control rod drive mechanism (CRDM) shroud cooling coils and out through the containment vessel penetrations. Approximately 75 gpm of flow is diverted to each shroud cooling coil by throttling flow diversion valves SW-901A-1/901B-1/901C-1/901D-1. Outside of containment are manual isolation valves SW-902A/902B/902C/902D and motor operated valves (MOVs) SW-903A/903B/903C/903D. Normal operation is with all four MOVs open.

On a safety injection (SI) signal, all four FCUs receive a start signal and the four discharge MOVs receive an open signal. Additionally, the CRDM shroud cooling coil isolation valves receive a close signal and diversion valves receive a full open signal to ensure unrestricted SW flow through the FCUs. Under accident conditions, each FCU has an air flow capacity of 41,000 cfm. Accident conditions that result in a containment pressure of greater than 3.85 psig would have an air flow path through the cooling coils, vane axial fan, distribution duct header and the emergency discharge dampers. When the emergency discharge dampers open, flow through the downstream ductwork is severely curtailed and cooling is lost to the RXCP motors.

Failure of the CRDM shroud cooling isolation and diversion valves to reposition on an SI signal would not adversely effect the CAC system design basis performance. Approximately 900 gpm of SW flow is required through each FCU to achieve the design basis heat removal rate under post-accident conditions. Data collected from SW flow tests have measured in excess of 1300 gpm in each SW discharge line. Therefore, since the flow diversion path through the shroud cooling coils is set at 75 gpm, failure to isolate is not considered in the fault tree model.

The two containment dome ventilation (CDV) fans recirculate and mix the hot air and post-accident H_2 from the top of containment vessel to prevent stratification. Each CDV fan has separate ductwork which allows the fan to pull air from the containment dome and discharge the air to the intakes of the FCUs. During the Kewaunee internal safety system functional inspection of the containment spray system[22], the post-LOCA containment pressure response was reverified using the computer code CONTEMPT. The reanalysis did not take into account the CDV subsystem operation, therefore, this subsystem is not considered in the fault tree model.

Fault Trees

The CAC and internal containment spray system each provide the required containment cooling during accident conditions independent of the other. The success criterion for the CAC system requires two out of four containment fan coil units operating for a defined mission time of 24 hours. This success criterion is applicable to all three fault trees for this system: FCH, FCD, FCHP. This success criterion is conservative based upon the results of evaluations using the MAAP code, which show that one of the four fan coil units is required for success.

FCH - Fault tree FCH is used in plant damage states quantification for all events except steam generator tube rupture, interfacing systems LOCA, loss of DC power, loss of offsite power, loss of service water, and station blackout events in which power is not restored in 48 hours.

FCD - Fault tree FCD is similar to fault tree FCH except this fault tree is for loss of DC power events and assumes the loss of one train of DC power.

FCHP - Fault tree FCHP is also similar to fault tree FCH except this fault tree is for loss of offsite power events; therefore, the failure of the FCUs to start and run once emergency power is restored is modeled.

FIGURE 3.2-5

REACTOR BUILDING VENTILATION SYSTEM

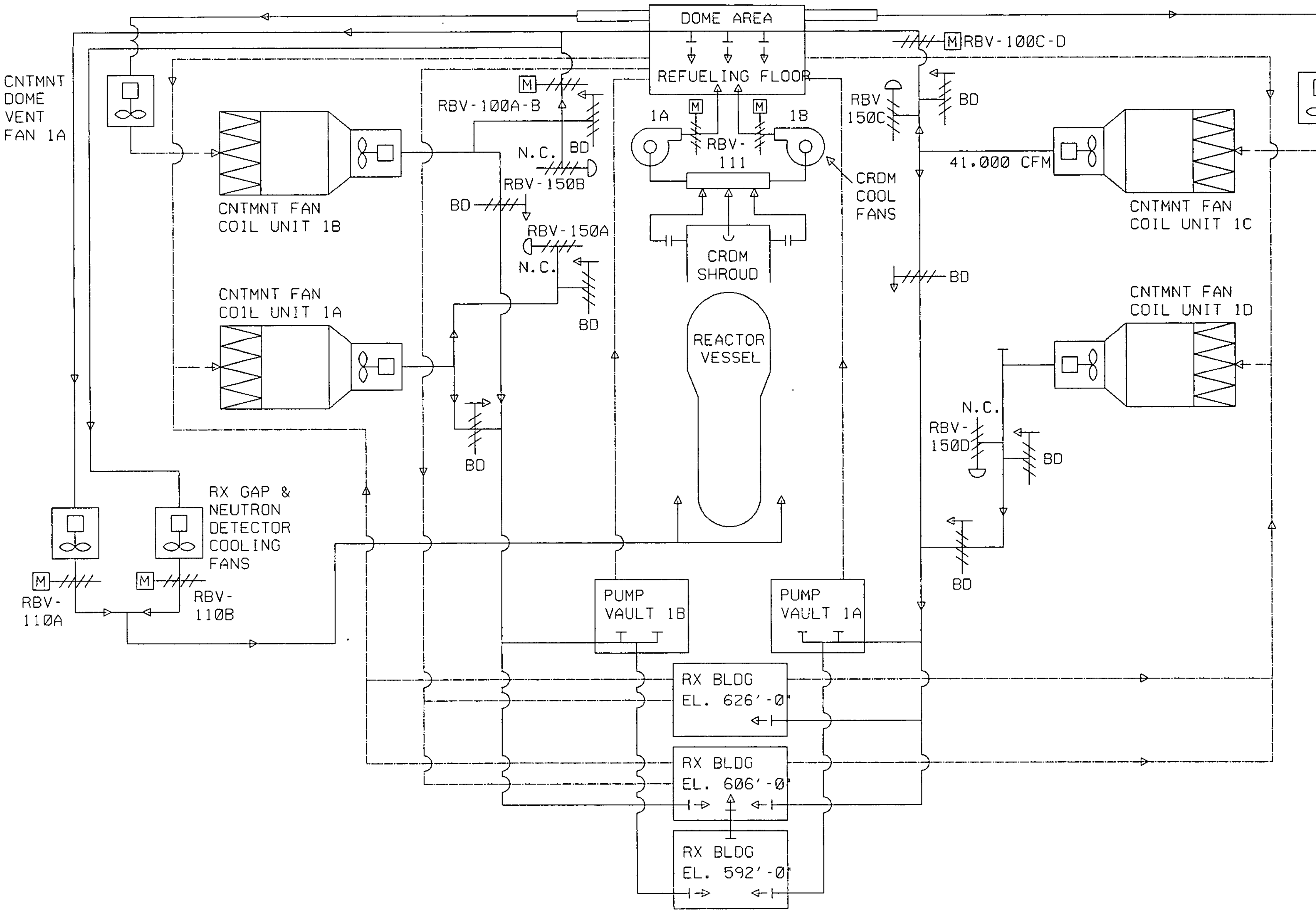

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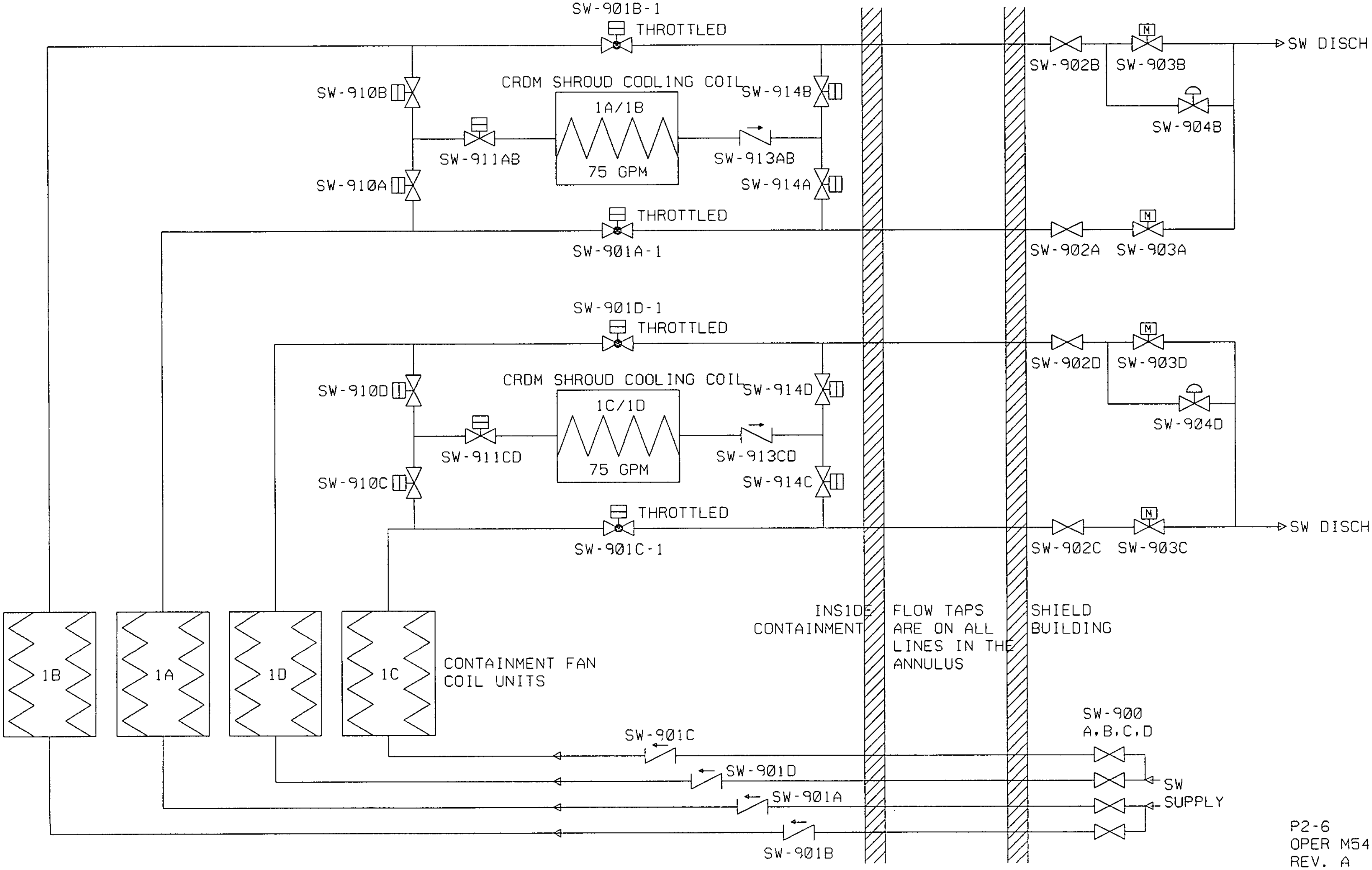
FIGURE 3.2-6

SERVICE WATER SYSTEM



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3.2.1.4 Containment Isolation

Function

The design of the Kewaunee Nuclear Power Plant includes three barriers to prevent the release of fission products to the environment: the fuel rod cladding, the reactor coolant system (RCS), and the containment. The capability of the containment to provide third barrier protection in the event of an accident in which one or both of the other two barriers is not maintained requires that containment integrity be established and maintained to limit the leakage of fission products to a low value.

Any flow of fluids that contain fission products from the containment following an accident is from one of the two sources: leakage from the containment structure and leakage through containment penetrations. The purpose of the study is to determine the probability that containment penetration isolation is not established prior to core damage or maintained following core damage. Refer to section 4.1.1.1 for greater detail on the containment structure.

Description

The containment system is designed to provide protection for the public from the consequences of a design basis loss of coolant accident. This design condition is based on an instantaneous double ended rupture of an RCS cold leg pipe.

The reactor containment vessel is designed for an internal pressure of 46 psig and a temperature of 268°F. The reactor containment vessel is routinely tested. Integrated leak rate tests of all penetrations are required to assure leakage is within the Kewaunee Technical Specifications (TS).

The shield building is a vertical cylinder which surrounds the reactor containment vessel and has a shallow dome roof. The shield building is constructed of reinforced concrete. The walls are 30" thick and the dome is 24" thick. The annulus, or annular space between the reactor containment vessel and the shield building, is approximately 5 feet wide. A slight vacuum is created by the Shield Bldg Ventilation System during accident conditions.

System Components - The Containment Isolation (CI) System consists of the following components and systems:

- Reactor containment vessel
- Piping penetrations
- Air locks
- Shield building penetrations
- Containment isolation valves
- Containment ventilation

Piping Penetrations - All penetrations, except the two vacuum breakers, penetrate the shield building and the reactor containment vessel. Both the shield building and the reactor containment vessel are provided with capped spare penetrations for possible future requirements.

All process lines traverse the boundary between the inside of the reactor containment vessel and the outside of the shield building by means of piping penetration assemblies made up of several elements. Two general types of piping penetration assemblies are provided: those that are not required to accommodate thermal movement (cold), and those that accommodate thermal movement (hot penetrations).

Both hot and cold piping penetration assemblies consist of a containment penetration sleeve and a shield building flexible seal. In the case of a cold penetration, the reactor containment vessel penetration nozzle is an integral part of the process pipe. For hot penetrations, a multiple-flued head becomes an integral part of the process pipe, and is used to attach a guard pipe and expansion joint bellows. The expansion joint bellows is welded to the reactor containment vessel penetration nozzle. The flued head fitting is the only part of the penetration assembly which comes into contact with the shield building at any time.

At the termination of a piping penetration assembly near the shield building, a low pressure leakage barrier is provided in the form of a shield building flexible seal. These devices provide a flexible membrane type closure between the shield building penetration sleeve, which is embedded in the shield building, and the process pipe. In the case of hot penetrations, a circular plate is used rather than a flexible seal. This circular plate serves as both an anchor and a shield building seal.

The double-bellows expansion joints in the hot pipe penetration assemblies and the shield building flexible seals for all pipes are designed to accommodate the maximum combination of vertical, radial, and horizontal differential movements of the reactor containment vessel, the shield building, and the piping.

Air Locks - The equipment hatch and personnel air locks are supported entirely by the reactor containment vessel and are not connected directly or indirectly to any other structure.

The equipment hatch and air locks were fabricated from welded steel and furnished with a double gasketed flange and bolted dished door. Provision is made to pressure test the space between the double gaskets of its flange.

Two personnel air locks are provided. The normal personnel air lock is located on the 649' 6" level, north wall of the containment building, south of the Spent Fuel Pool (SFP). The emergency air lock is located on the containment 626' level, in the auxiliary building southeast wall next to the "B" Main Steam Isolation Valve/Feedwater Regulating Valve Station. Each personnel air lock is a double door welded steel assembly. Quick acting type equalizing valves are provided to equalize pressure in the air lock when entering or leaving the reactor containment vessel. Provisions are made for performing pressure tests of the air locks for periodic leak rate tests.

The two doors in series in each personnel air lock are interlocked to prevent both doors from being opened simultaneously. This interlock ensures that one door is completely closed before the other door can be opened. Remote red/green indicating lights in the control room indicate each doors' operational status. Provision is made to permit bypassing the door interlock system with a special tool to allow both doors to be left open during a plant cold shutdown for maintenance. Each door lock hinge can be adjusted to assist in proper seating.

Local air lock controls and indications include the red/green indicating lights and handwheels to open and close the air lock doors. A lighting and communication system, which is operated from an external emergency source, is provided within each air lock.

Shield Building Penetrations - The shield building penetrations for piping, ducts, and electrical cable are designed to withstand the normal environmental conditions which may prevail during plant operation and also to retain their integrity during and following postulated accidents.

The openings into the shield building, including personnel access openings, equipment access openings, and penetrations for piping, duct, and electrical cable, are designed to provide containment that is as effective as the shield building and consistent with the shield building leak rate.

The shield building is provided with two access openings, one located adjacent to the emergency air lock and the other adjacent to the personnel air lock. Each access opening is provided with double interlocked doors. In addition, a bolted, sealed door is provided at each airlock.

Pipe penetrations through the shield building are sealed with low pressure flexible closures. The seals are of a rubber-impregnated canvas material or equivalent and seal the process line to the embedded sleeve in the shield building.

Containment Isolation Valves - General Design Criteria #55 requires that all penetrations that require closure for the containment function must be protected by redundant valving and associated apparatus (GDC 55).

Isolation valves are provided as required for fluid system lines penetrating the reactor containment vessel to assure the following:

- Leakage through all fluid line penetrations not serving accident-consequence-limiting (non-safeguards) systems is minimized by a double barrier; i.e., two isolation valves in series in the fluid pipe. The double barriers take the form of closed pipe systems, both inside and outside the reactor containment vessel, and various types of isolation valves. The double barrier arrangement provides two reliable low leakage barriers between the

RCS or containment atmosphere and the environment. The failure of any one barrier does not prevent isolation.

- Fluid line penetrations normally serving accident-consequence-limiting (safeguards) systems are isolated by manual action if the automatic system should malfunction.

- No single credible failure or malfunction occurring in any active system component can results in loss of isolation or excessive leakage.

Containment Ventilation - The containment purge & ventilation system consists of a tempered air supply and a filtered exhaust system. The containment vent exhaust discharges to the monitored containment system vent that extends through the shield building annulus to a point approximately 5' above the lower edge of the shield building domed roof. The equipment for the reactor building vent system is located outside the shield building, in the auxiliary building.

The ventilation isolation valves immediately outside the reactor containment vessel are conventional butterfly valves specified to be adequately leak tight with maximum internal pressure inside the reactor containment vessel. The ventilation isolation valves inside the reactor containment vessel are butterfly valves that are leak tight with maximum internal pressure on either side of the valves. This feature permits the space between the two containment purge isolation valves to be pressurized to the maximum internal pressure (46 psig) at any time to ascertain continued leak tightness. The valve shaft seals for all purge isolation valves consist of a double seal with a leak test connection between the seals that is pressurized for shaft seal leak testing. The purge isolation valves fail closed upon loss of actuating power, electric or air.

The vent valves which perform a containment isolation function are designed to withstand the necessary earthquake loadings. Each isolation valve was reviewed during the final design of piping systems to determine the extent of support required. Most of the valves are supported by the piping/vent system of which they are a part.

Vacuum relief valves are provided to protect the reactor containment vessel against excess differential pressures. The vacuum relief valves are sized to assure that the reactor containment vessel is not subjected to an internal vacuum of 0.8 psi or greater below the external pressure. Two valves in series are used in each of the two large containment penetrations which permit air flow from the shield building annulus into the reactor containment vessel. The vacuum relief valves in each line consist of an air operated butterfly valve and a self-actuated, horizontally installed, swing disc check valve. Individual air accumulators are provided at each vacuum breaker isolation butterfly valve to allow one complete cycle of the valve, open and closed, following a loss of instrument air.

Fault Trees

One fault tree was developed to represent the Containment Isolation function. The success criterion is based upon the assumptions that significant fission product releases to the environment occur only under the following conditions:

- The line penetrating contain is greater than 2 inches in diameter,
- The line penetrating containment directly communicates with either the containment atmosphere or the reactor coolant system during any severe accident scenario, and
- The line-penetrating containment is not part of a "closed" system outside of containment, capable of withstanding severe accident conditions.

CI - The CI fault tree was developed to provide the Containment Isolation function for plant damage states quantification. In the CI fault tree the containment fluid system and mechanical penetration have been categorized as described below.

Category A - Those that are administratively controlled and required to be closed during operation,

Category B - Those that can be open during normal operation and are required to change position upon receipt of an isolation signal, and

Category C - Those that are normally open following an engineered safety feature actuation signal.

For penetrations that fit into the above categories, the success criteria for containment isolation are that:

1. Those penetrations in Category A are in the closed position at the time of the accident (i.e. they should be closed during normal operation),
2. Those penetrations in Category B have moved to the closed position, if they were open, upon receipt of an isolation signal, and
3. Those penetrations in Category C are sealed from the atmosphere by properly functioning check valves and/or motor operated valves if the system is unavailable or not operating.

3.2.1.5 Electrical Power

Function

The electric power system (EPS) provides a reliable source of power to all plant systems required during normal or emergency plant operation.

The primary functions of the EPS are to:

- Provide a reliable source of motive power to those components whose operation is necessary for the mitigation of any abnormal event effecting the reactor core, its heat removal systems, or systems that could effect the release of radioactivity to the environment.

- Provide a reliable source of control power for the operation of these systems and for the initiation of safeguards systems actuation signals.

- Provide a reliable source of power to instrumentation necessary for the monitoring of emergency system functions, for the monitoring of key plant parameters, and for inputs to safeguards systems actuation logic matrices.

Independence or isolation of supply to the various redundant engineered safety features (ESF) is maintained so a single bus fault will not result in the total loss of the plant's engineered safeguards systems.

Description

The electrical system for Kewaunee consists of the high voltage AC substation/off-site power distribution system (345kV, 138kV and 13.8kV), the on-site AC distribution systems (4160V, 480V) and the instrumentation and control power distribution system (125VDC, 120VAC). See Figures 3.2-7 through 3.2-20. The on-site distribution systems are divided into vital and non-vital subsystems.

The on-site electrical distribution systems have the ability to receive power from various sources including the output from the main generator via the main auxiliary transformer. During normal plant operation the main generator provides power to the non-vital electrical buses. The vital buses are supplied by two separate off-site sources. During abnormal operating conditions (startup, shutdown, or postulated accident), the non-vital buses are supplied by off-site power via the plant startup transformer (reserve auxiliary). The vital buses continue to be supplied by the off-site power sources described earlier. If off-site power is unavailable, the non-vital AC electrical systems are deenergized. The vital AC buses are supplied by a separate emergency diesel generator dedicated to each bus in order to complete a safe shutdown of the reactor. The on-site electrical system is divided into the following systems:

- Vital 4160VAC Distribution System
- Emergency Diesel Generators
- Vital 480VAC Distribution System
- Vital 125VDC Distribution System
- Vital 120VAC Distribution System
- Non-Vital 4160VAC Distribution System
- Non-Vital 480VAC Distribution System
- TSC Diesel Generator
- Non-Vital 125VDC Distribution System
- Non-Vital 120VAC Distribution System

Vital 4160VAC Distribution System - The vital 4160VAC distribution system is divided into electrical buses 5 and 6. Bus 5 supplies power to ESF train A components. Bus 6 provides power to ESF train B components. Buses 5 and 6 also provide power to two 480VAC vital buses via station service transformers.

The normal source of power to bus 5 is the tertiary auxiliary transformer. The reserve auxiliary and main auxiliary transformers provide backup sources. The normal source of power to bus 6 is the reserve auxiliary transformer. The tertiary auxiliary and main auxiliary transformers provide backup sources. Thus, since the normal source of power for these buses is the 13.8/138/345kV Kewaunee substation, no transfer is required in the event of a turbine-generator trip.

Emergency Diesel Generators - There are two emergency diesel generators at Kewaunee. Diesel generators A and B are normally aligned for standby operation to provide emergency power for 4160V buses 5 and 6, respectively.

Each diesel generator unit is a General Motors Corporation, Electro-Motive Division, Model 999-20 supplied by the Western Engine Company. Each diesel is a 20 cylinder engine model S20E46 and is directly connected to an eight pole air cooled generator model A-20-C1. The continuous rating of each unit is 2600 kW at 0.8 power factor, 4160V three phase, and 60 Hz. Each diesel generator has additional overload ratings of 2860 kW for 2000 hours per year, 2950 kW for seven days per year and 3050 kW for thirty minutes per 24 hours. The normal operating speed of each unit is 900 RPM. The diesel-generator auxiliaries are supplied with power from MCC 52A and MCC 62A located in the respective diesel generator room.

Each diesel generator, as a backup to the normal and standby AC power supplies, is capable of sequentially starting and supplying the power requirements of one complete set of ESF equipment. Each diesel generator receives an automatic starting signal under either of the following conditions:

- Undervoltage on its associated 4160V vital bus

- Safety injection signal from its associated train.

Vital 480VAC Distribution System - The power required for the 480V ESF and other vital plant loads is supplied from four 480V buses fed from 4160V buses 5 and 6. Transformer 51 is fed from 4160V bus 5 through breaker 505 and supplies 480V bus 51. This transformer, bus and breakers, including one bus tie breaker, are assembled as a switchgear unit. In a similar manner, 480V bus 52 is also connected via breaker 505 to 4160V Bus 5.

A redundant 480V system (buses 61 and 62) is supplied by 4160V bus 6 through 4160V breaker 607.

The large 480V ESF motors are connected to buses 51 and 61. Motor Control Centers (MCCs) supplying the smaller loads are fed from buses 52 and 62. MCCs and their locations are listed in Section E, Component Location.

Vital 125VDC Distribution System - The 125VDC Distribution System is divided into two buses (BRA-102, BRB-102) with one battery and battery charger serving each bus. The two batteries (BRA-101 and BRB-101) each consist of 59 cells and is of the lead calcium type. Each battery is rated 125VDC, 1304 ampere hours at the eight hour rate without discharging below minimum cell voltage. The 125VDC supplies plant controls, emergency DC motors, inverters serving emergency lighting and the four 120VAC instrument buses directly through bus breakers by way of distribution cabinets BRA-104 and BRB-104 and the associated fuse panels.

The battery chargers (BRA-108 and BRB-108) supply the normal DC loads as well as maintaining proper charges on the batteries. A third battery charger (BRA/B-108), which is portable, is available to replace either charger should one fail or need to be removed for maintenance. The system is provided with AC input and DC output access with separate AC and DC breakers to enable connecting the spare charger to either DC bus. The battery life to minimum voltage under maximum anticipated load will allow sufficient time to make this changeout.

DC Bus BRA-104 supplies the controls to Diesel Generator A, ESF Buses 5, 51 and 52, as well as one-half of the redundant safety circuits.

DC Bus BRB-104 supplies the controls to Diesel Generator B, ESF Buses 6, 61 and 62, as well as the other half of the redundant safety circuits.

Vital 120VAC Distribution System - The 120VAC Distribution System provides power to the plant instrumentation, control and protection systems. The 120VAC supply is split into several buses. There are four primary independent instrument buses; I (BRA-113), II (BRB-113), III (BRB-114), IV (BRA-114); each fed by an inverter which in turn is fed by normal and alternate AC power sources from 480VAC MCCs and a standby power source from 125VDC Buses BRA-104 and BRB-104. Instrument Buses I (BRA-113) and IV (BRA-114) are associated with train A ESF actuation circuits whereas instrument Buses II (BRB-113) and III (BRB-114) are associated with train B circuits.

The normal source of power to each instrument bus is from its associated inverter. The inverter converts the 480VAC normal supply to a DC voltage. The inverter also modifies the 125VDC standby source. The normal and standby sources are then joined together within the inverter cabinet. The source having the highest amplitude (normally the 480VAC source) is then converted to 120VAC and connected to a synchronization switch within the inverter cabinet. The alternate 120VAC supply is an independent source fed by a 480VAC MCC via an instrument transformer BRA-106/BRB-106 and instrument bus BRA-105/BRB-105. This 120VAC supply is also connected to the internal sync switch as well as a manual bypass switch.

The basic operation of an inverter is such that if the normal source (480VAC) of power is lost, then the standby (125VDC) source will automatically supply the instrument bus via the synchronization switch. If, however, the normal and standby sources are lost for any reason, the synchronization switch would shift to alternate (120VAC) source. The manual bypass switch serves to provide a direct tie to the alternate source as well as bypass the total inverter and synchronization switch to allow maintenance on the inverter cabinet components.

There are also two minimum interruptable instrument buses BRA-105 and BRB-105. Instrument buses BRA-105/BRB-105 provides the alternate power source for instrument buses BRA-113/BRA-I14 and BRB-113/BRB-114 by way of their respective inverters as described previously. These buses also provide power to various vital and non-vital instruments.

The normal power source for BRA-105/BRB-105 is provided by 480V MCC 52C/62C by way of auto transfer switches BRA-107/BRB-107 and transformers BRA-126/BRB-126. Loss of the normal supply causes the auto transfer switch to shift to the alternate supply 480V MCC 52E/62E.

Non-Vital 4160VAC Distribution System - The non-vital 4160V distribution system is divided into four buses. Buses 1 and 2 are connected via bus main breakers to the main auxiliary and reserve auxiliary transformers. These buses supply power to the reactor coolant pumps and the feedwater pumps.

Buses 3 and 4 are also connected via bus main breakers to the main auxiliary and reserve auxiliary transformers. These buses supply power to the normal balance-of-plant auxiliaries, and each bus supplies power to three (4160V-480V) station service transformers. A fourth transformer connected to bus 4 supplies power to the technical support center (TSC) 480V distribution system.

Non-Vital 480V Distribution System - The non-vital 480V distribution system is divided into seven load center or switchgear buses. They are fed from 4160V buses 3 and 4 serve balance-of-plant loads.

Transformers 32 and 42 are connected to 4160V buses 3 and 4, respectively. Transformer 32 feeds 480V bus 32; transformer 42 feeds 480V bus 42. These components including the 480V bus tie are assembled as a conventional, double-ended switchgear unit. In a similar manner, buses 33/43 and 35/45 are connected to 4160V buses 3 and 4. Bus 46, supplying the TSC, is connected to 4160V bus 4 or alternately from the TSC diesel generator.

The various MCCs throughout the plant are then connected to these switchgear buses.

TSC Diesel Generator - The TSC diesel generator (DG) is designed to provide emergency power for the TSC building, security lighting system, and other non-ESF plant systems which are required to operate upon loss of the main generator and loss of offsite power sources.

The TSC DG does not supply power to any ESF equipment. An automatic start occurs on loss of voltage to bus 46. The TSC DG is normally in standby and can start and assume load within 10 seconds of a start signal. The TSC DG has a design rating of 600 kw at a 0.8 power factor (750 KVA). The DG has a continuous rating of 515 kW/643.75 KVA.

There is a plant modification in process which would improve the capability of the EPS. The changes that are being made are in response to the NRC Station Blackout Rule. This design change involves the use of the TSC diesel generator under station blackout conditions. This would provide an alternate power source to the emergency 480V AC bus 52 in the case where offsite and onsite power is lost. This design change is expected to be operational in 1993.

The Non-Vital 125VDC Distribution System - The non-vital 125VDC distribution system is divided into two buses each with one battery and a battery charger, distribution panels and inverters. Components prefixed with BRC and BRD make up the non-safeguard system.

The balance of plant DC power requirements are supplied by two non-safeguard batteries (designated BRC-101 and BRD-101). Each of these batteries consists of 59 cells and are of the lead calcium type, rated at 125VDC, 1680 ampere-hours at the eight hour rate to reach minimum cell voltage. Each battery is connected to a main distribution panel (BRC-102/BRD-102). The main distribution panel connects each battery to a battery charger, sub-distribution panel, bus tie and inverter.

Distribution panels BRC-102 and BRD-102 supply sub-distribution panels BRC-103 and BRD-103, respectively. The BRC-103 and BRD-103 panels, in turn, supply TSC diesel generator control and excitation power, control power for non-ESF buses 1, 2, 3, 4, 32, 33, 35, 42, 43, 45 and 46, other non-safety related equipment sensitive to a loss of AC power and are a standby source for inverters BRC-109, BRD-109, and a proprietary inverter.

Non-Vital 120VAC Distribution System - The non-vital 120VAC distribution system is split into four buses. There are two independent instrument buses, each fed by an inverter which, in turn, is fed from each of the DC buses BRC-103/BRD-103. The independent noninterruptible bus BRD-115 is fed by inverter BRD-109. The other independent bus, BRC-107, is fed by inverter BRC-109.

The other two non-vital instrument buses are BRA-127 and BRB-127. The normal power source for BRA-127/BRB-127 is provided by 480V MCC 52C/62C by way of auto transfer switch BRA-107/BRB-107, transformer BRA-106/BRB-106 and isolation cabinet BRA-126/BRB-126. Loss of the normal supply would cause the auto transfer switch to shift to the alternate supply, 480V or MCC 52E/62E.

Fault Trees

Thirty three fault trees were developed for the electrical power system. Other fault trees were developed from the EPS fault trees and are provided for special use in the quantification process. The electrical power system fault trees are:

BUS1, BUS2, BUS3, BUS4, BUS5 and BUS6 - These models define the logic associated with the unavailability of the six 4160V AC buses during all postulated accidents in which offsite power is available.

BUS5P and BUS6P - These models define the logic associated with the unavailability of the two vital 4160V AC buses during accidents where offsite power is unavailable.

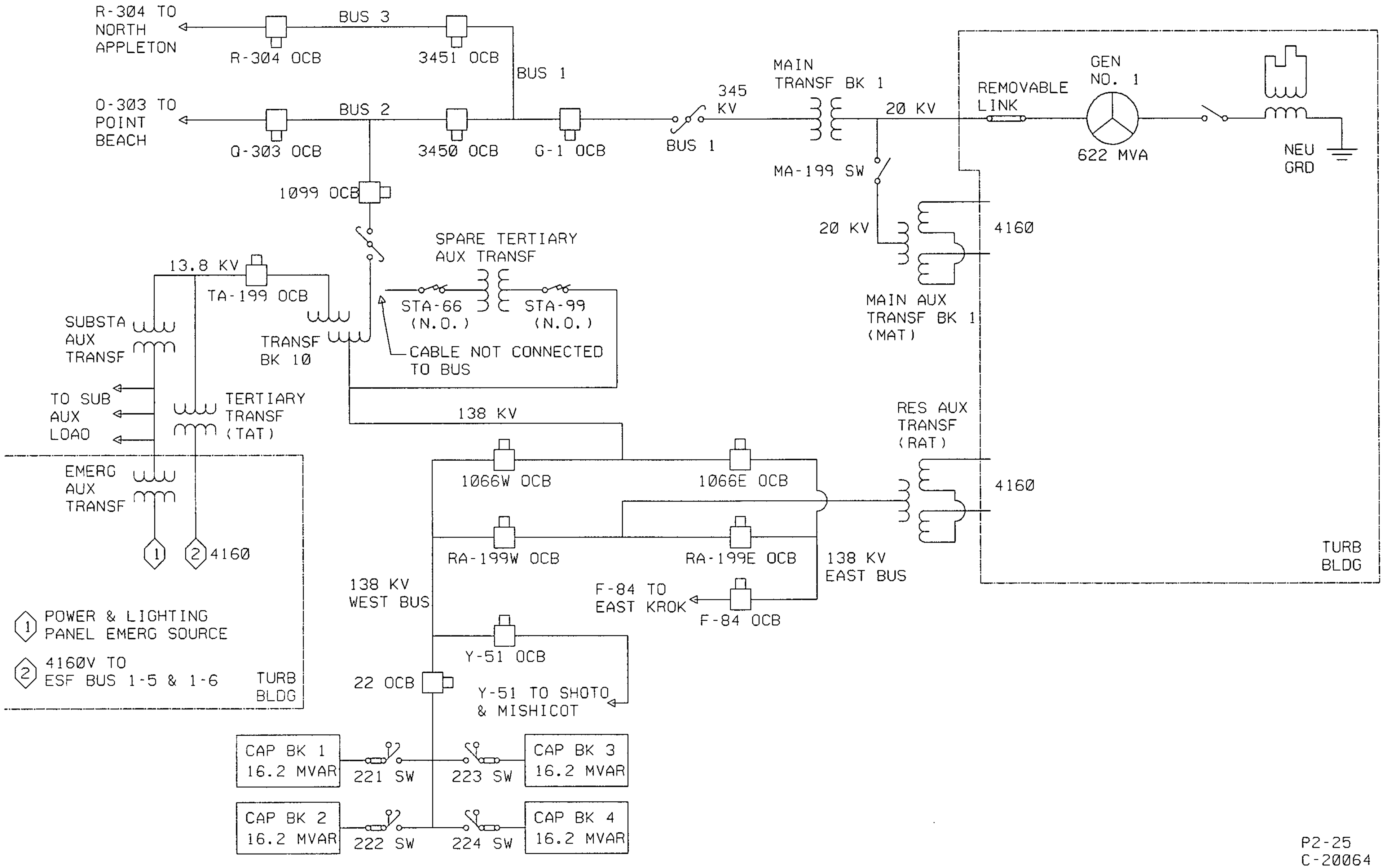
BUS32, BUS42, BUS35, BUS45, BUS46, BUS51, BUS52, BUS61 and BUS 62 - These models define the logic associated with the unavailability of the nine 480V AC buses.

BRC103, BRD103, BRA104, and BRB104 - These models define the logic associated with the unavailability of the four 125V DC buses.

BRA113, BRB113, BRA114, BRB114, BRA105, BRB105, BRA127, BRB127, and BRD115 - These models define the logic associated with the unavailability of the nine 120V AC instrument buses.

DGA, DGB, and TDG - These models define the logic associated with unavailability of the starting and operation of emergency diesel generators and the technical support center diesel generator.

FIGURE 3.2-7
SUBSTATION DISTRIBUTION



P2-25
C-20064
5-9-88

FIGURE 3.2-8
SWITCHYARD ELECTRICAL GENERAL ARRANGEMENT

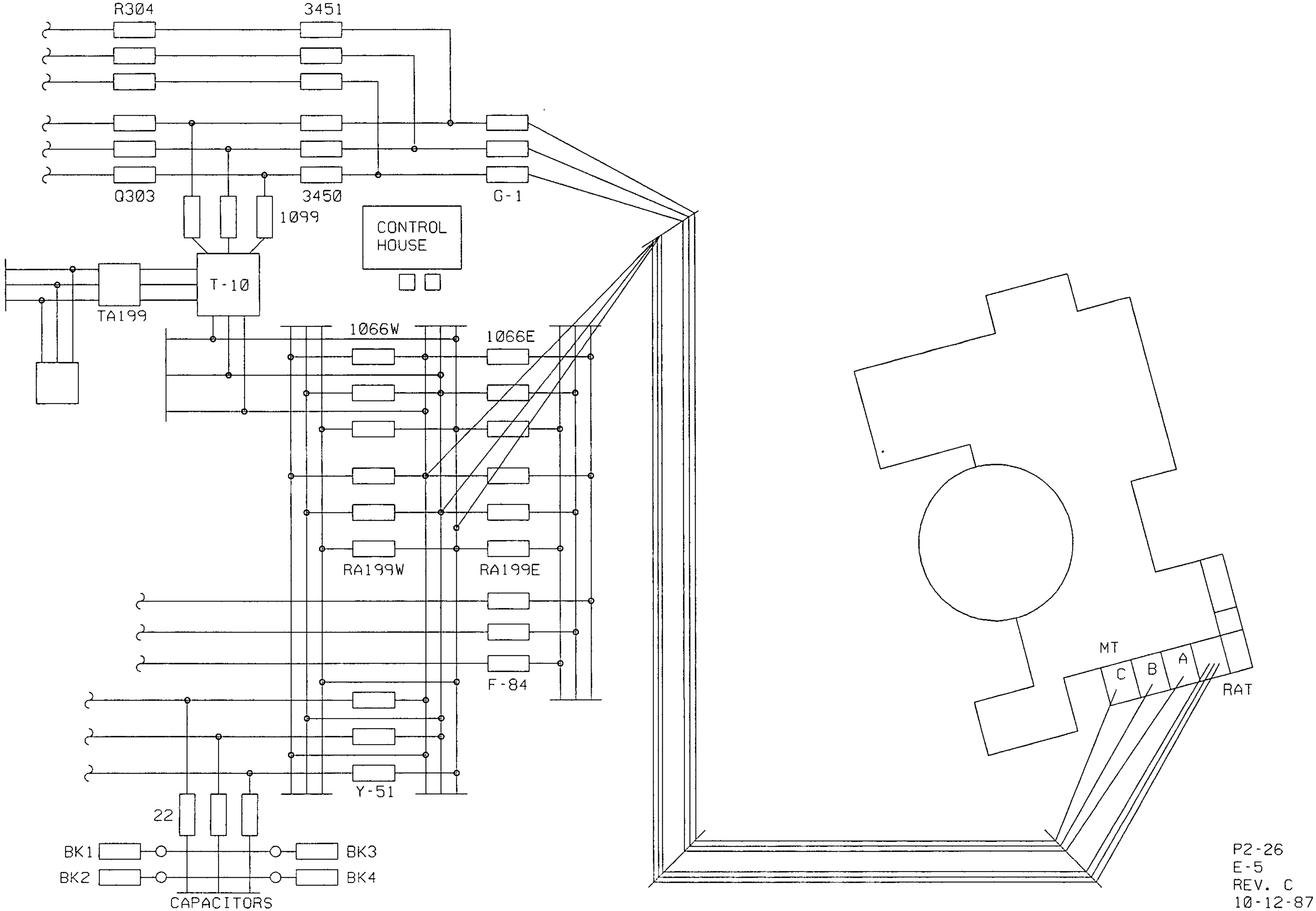
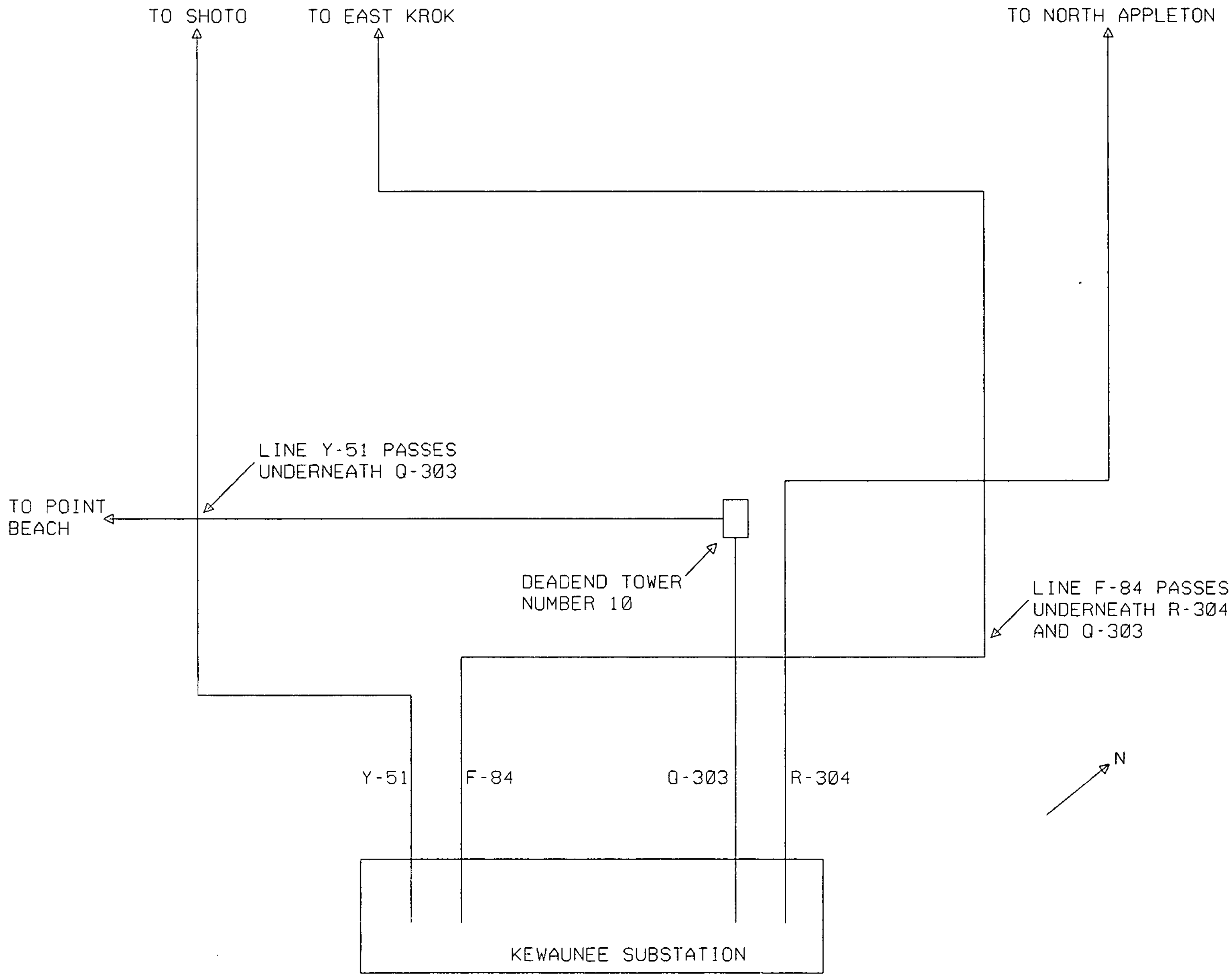


FIGURE 3.2-9

OFFSITE POWER LINES TO SUBSTATION



P2-27
10-12-87

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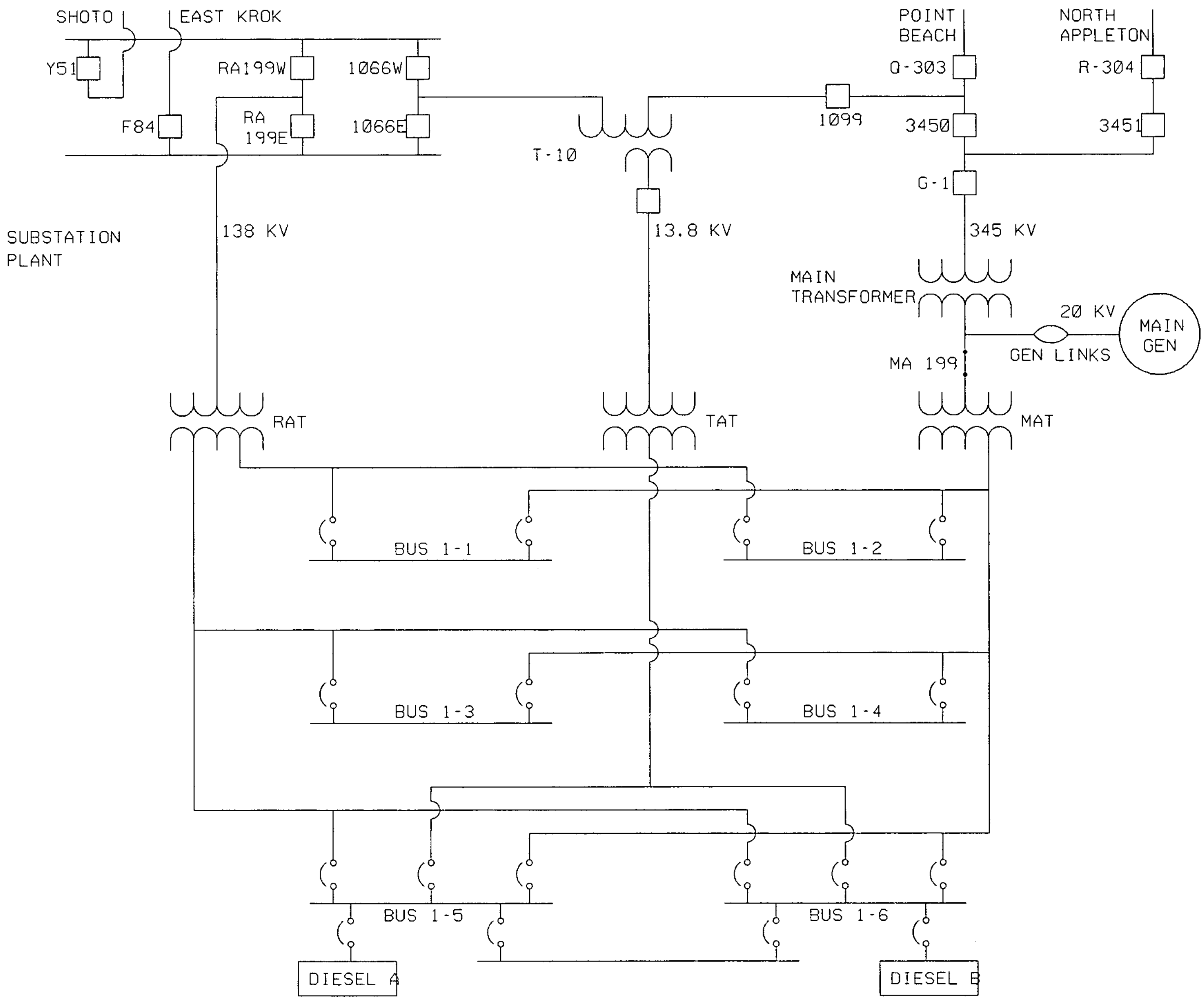
FIGURE 3.2-10

PLANT & SUBSTATION ELECTRICAL DISTRIBUTION



P2-28
10-12-87

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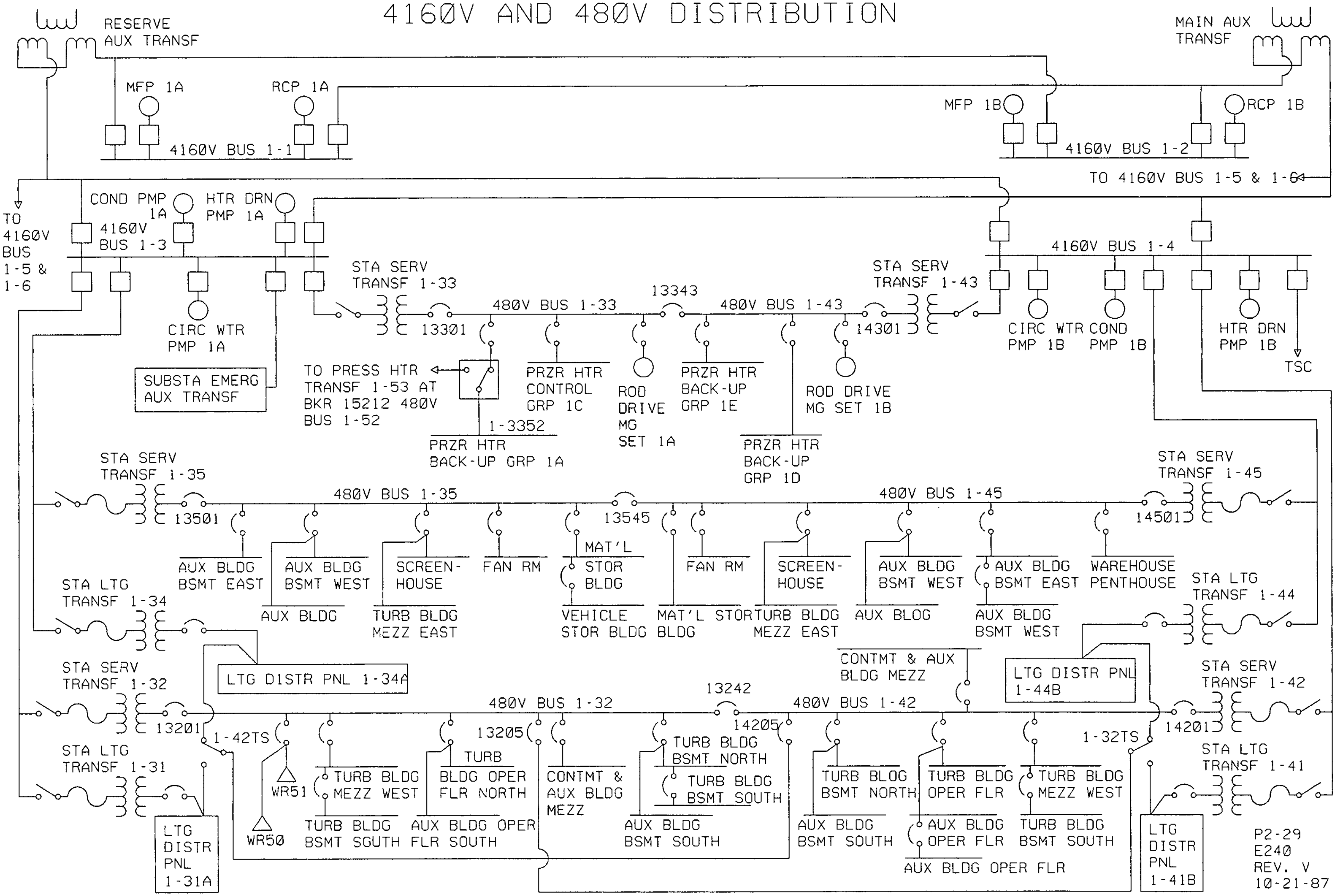
FIGURE 3.2-11

4160V AND 480V DISTRIBUTION



P2-29
E240
REV. V
10-21-87

C:\DGN\FG\M3-11.DGN Sep. 28, 1992 13:46:15

FIGURE 3.2-12

ESF 4160V & 480V DISTRIBUTION

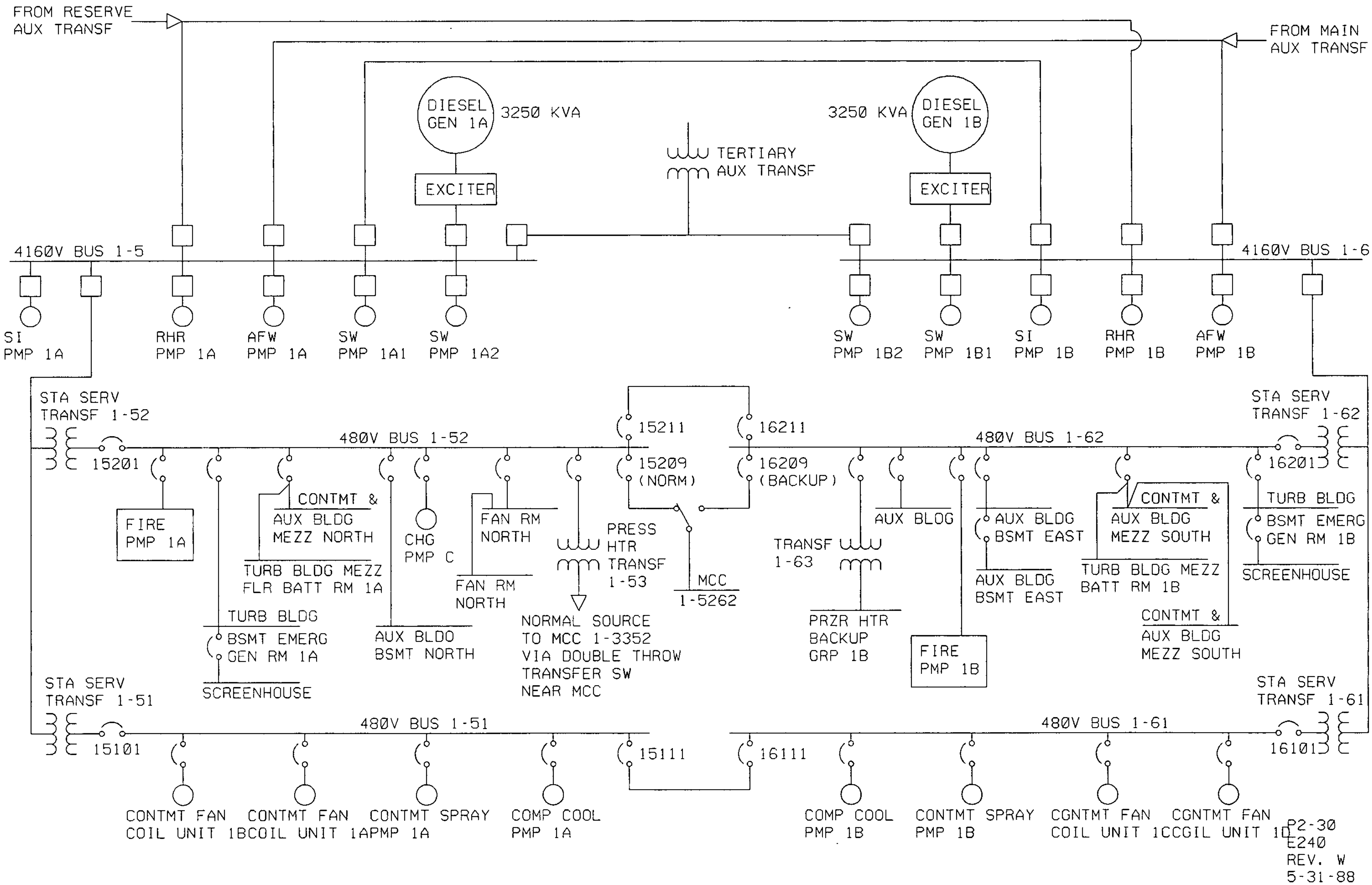
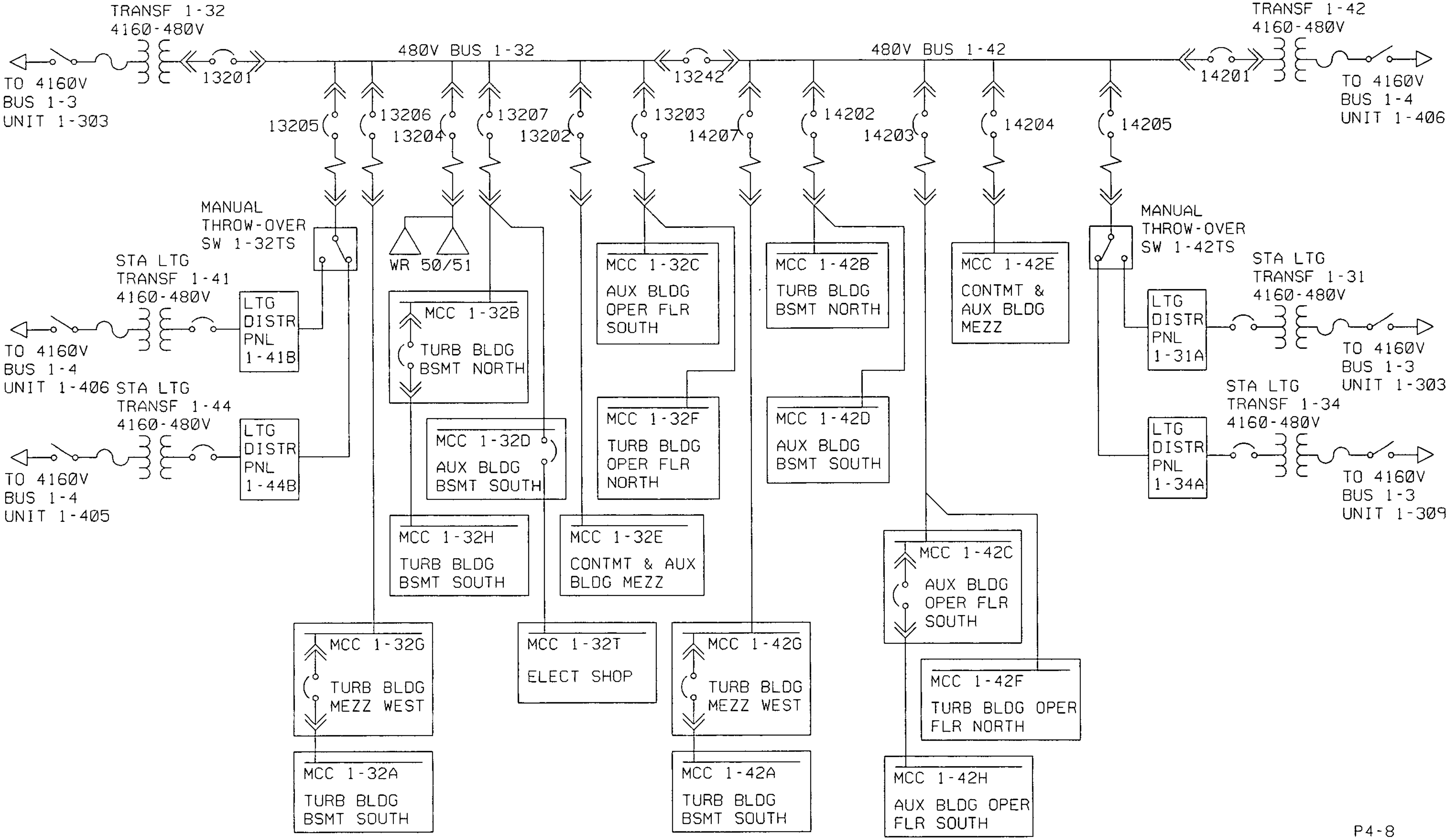


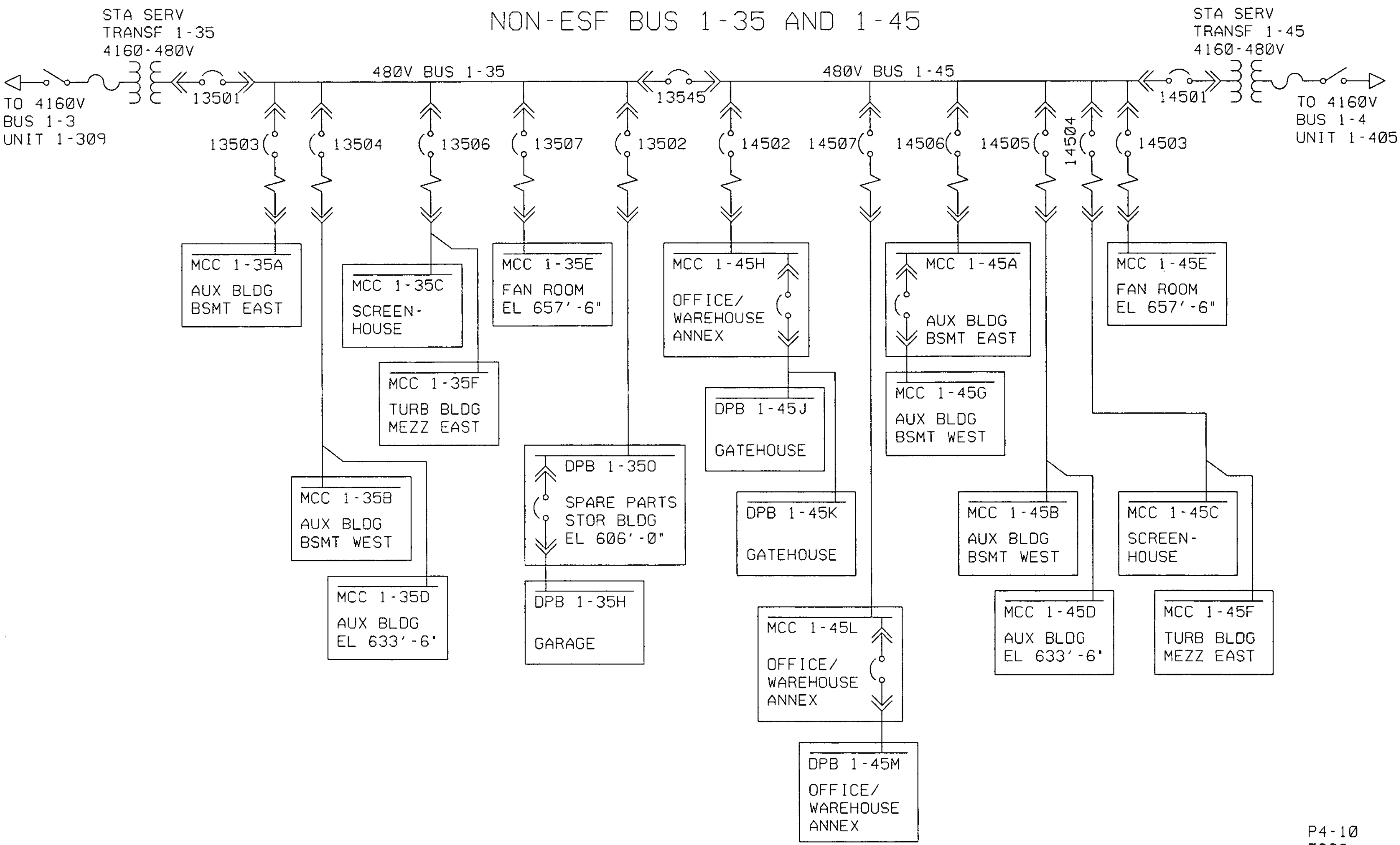
FIGURE 3.2-13
480V SWITCHGEAR
NON-ESF BUS 1-32 AND 1-42



P4-8
E228
REV. AB
6-6-88

FIGURE 3.2-14
480V SWITCHGEAR

NON-ESF BUS 1-35 AND 1-45



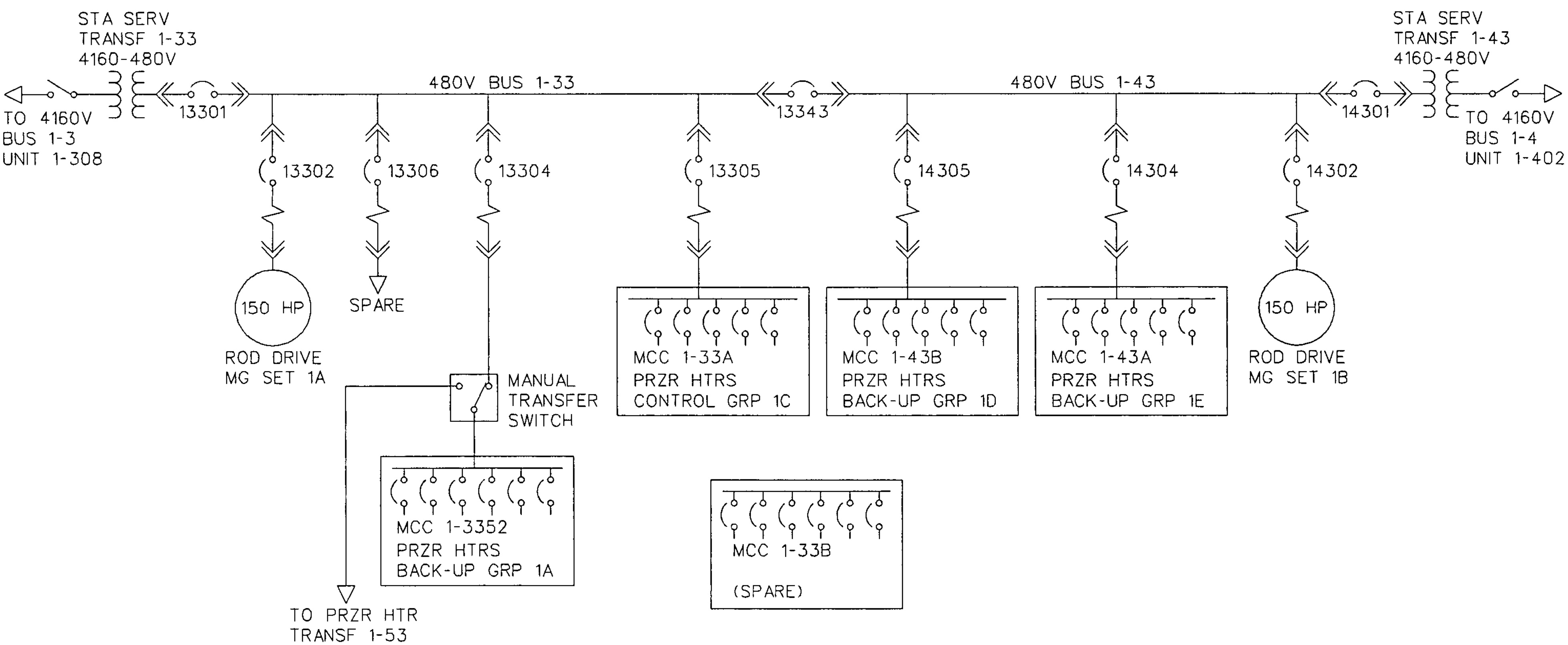
P4-10
E228
REV. AA
10-28-87

C:\DGN\FG\N3-14.DGN Sep. 28, 1992 13:49:14

FIGURE 3.2-15

480V SWITCHGEAR

NON-ESF BUS 1-33 AND 1-43

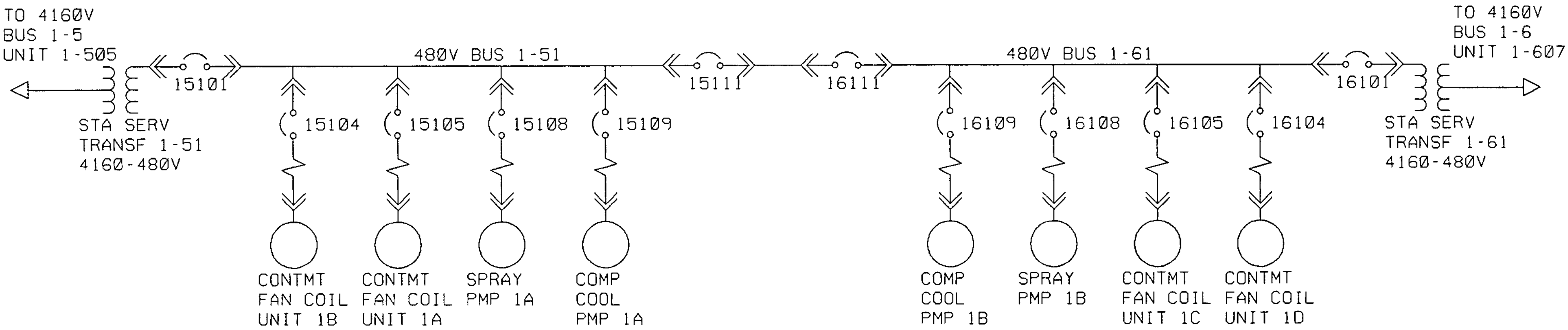


N3-15
E228
REV. AA
10-28-87

C:\DGN\FG\N3-15.DGN Oct. 02, 1992 10:24:29

FIGURE 3.2-16

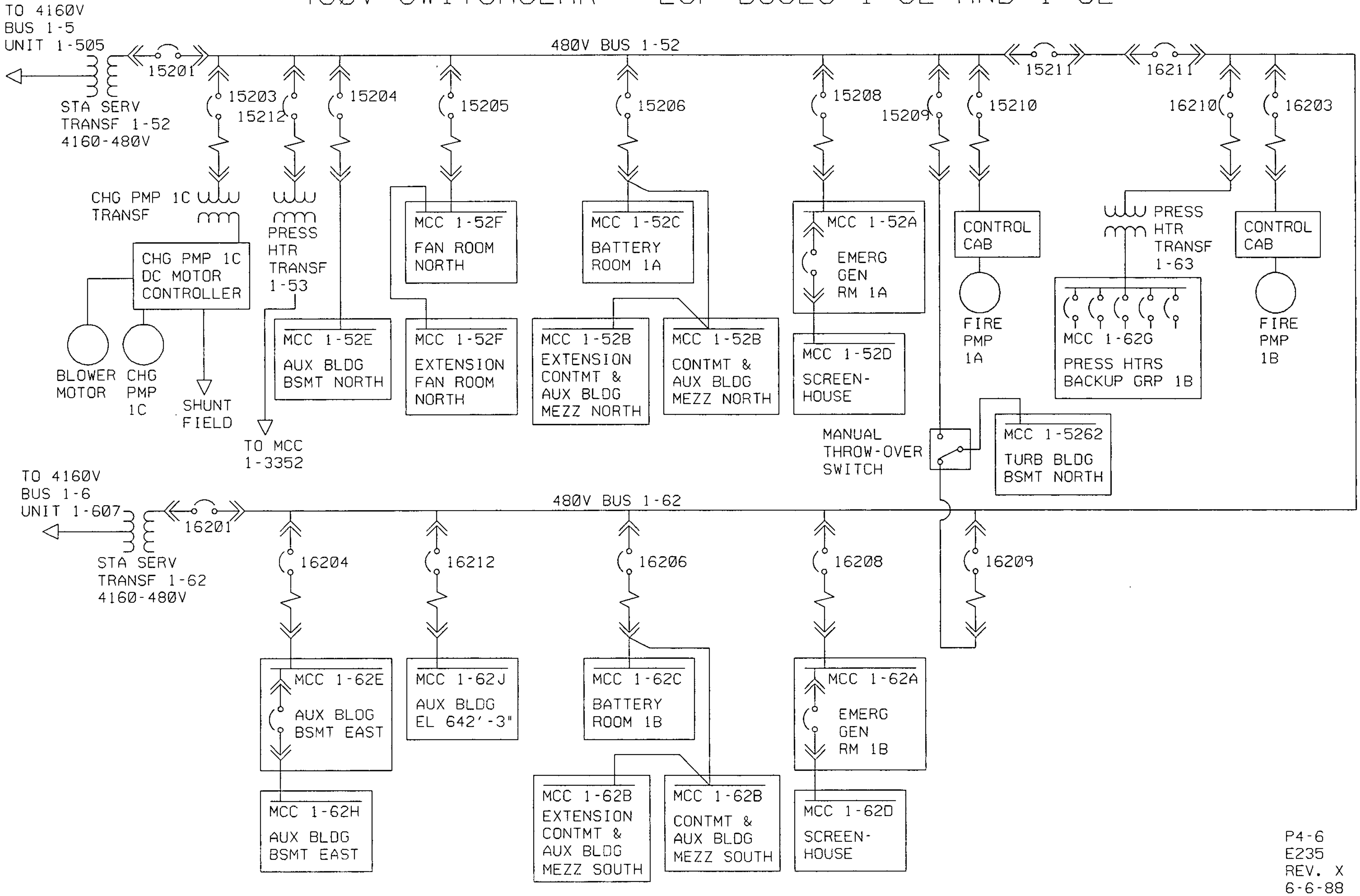
480V SWITCHGEAR - ESF BUSES 1-51 AND 1-61



P4-5
E235
REV. X
6-6-88

FIGURE 3.2-17

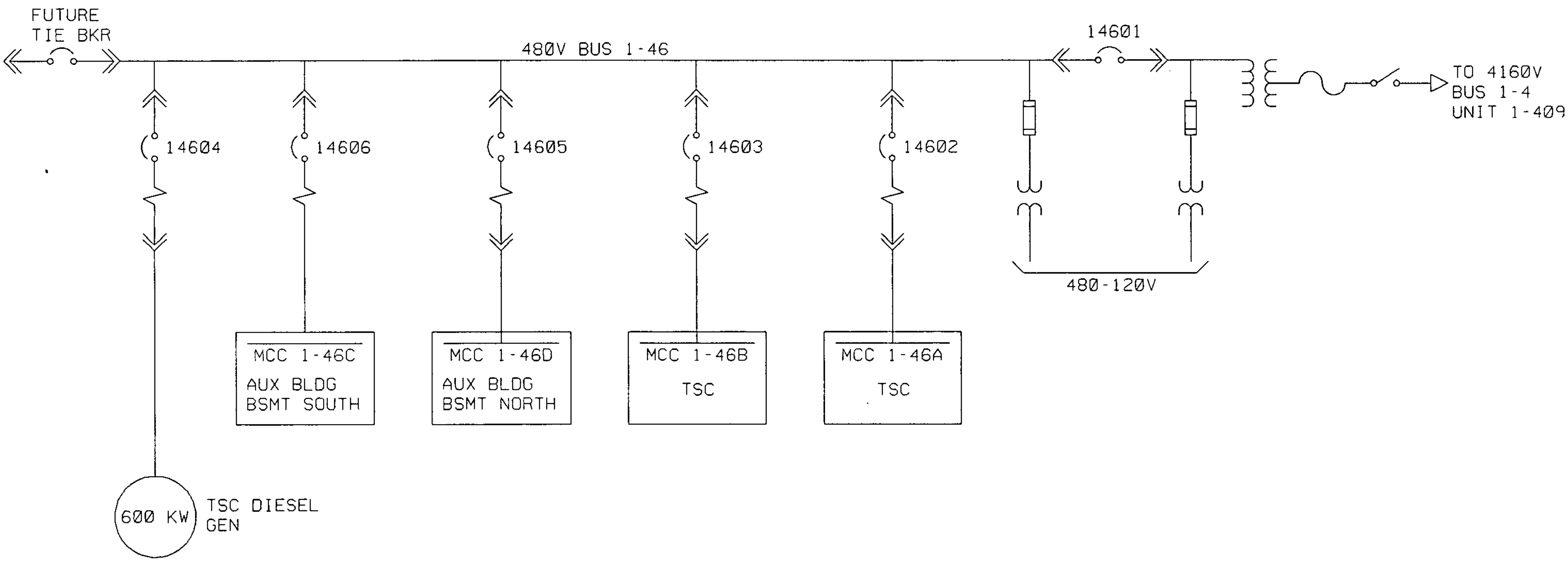
480Ø V SWITCHGEAR - ESF BUSES 1-52 AND 1-62



C:\DGN\FG\N3-17.DGN Sep. 28, 1992 13:51:30

FIGURE 3.2-18

TECHNICAL SUPPORT CENTER POWER SERVICE



P4-9
E2902
REV. D
10-28-87

C:\DGN\F6\N3-18.DGN Sep. 28, 1992 13:52:23

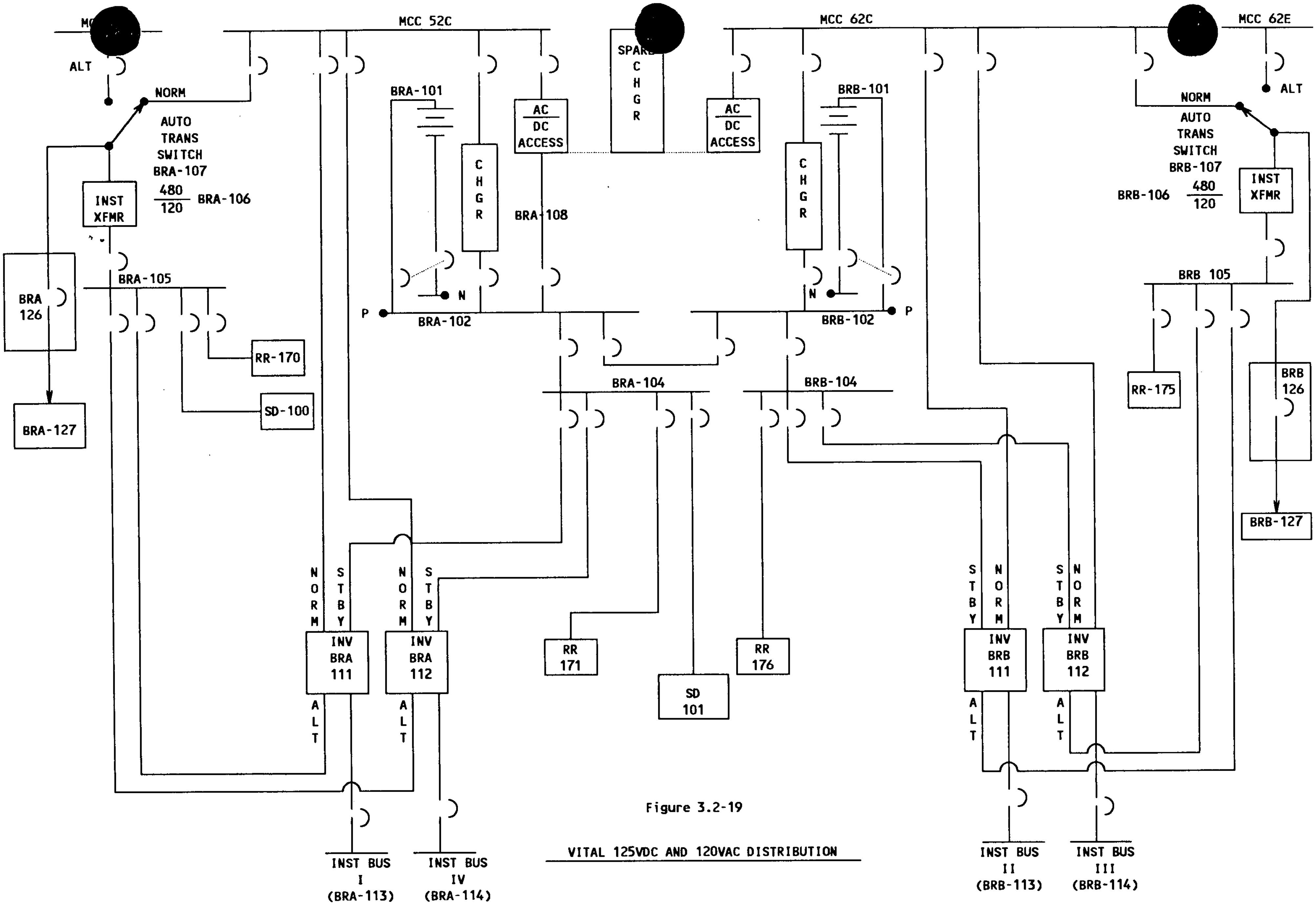


Figure 3.2-19

VITAL 125VDC AND 120VAC DISTRIBUTION

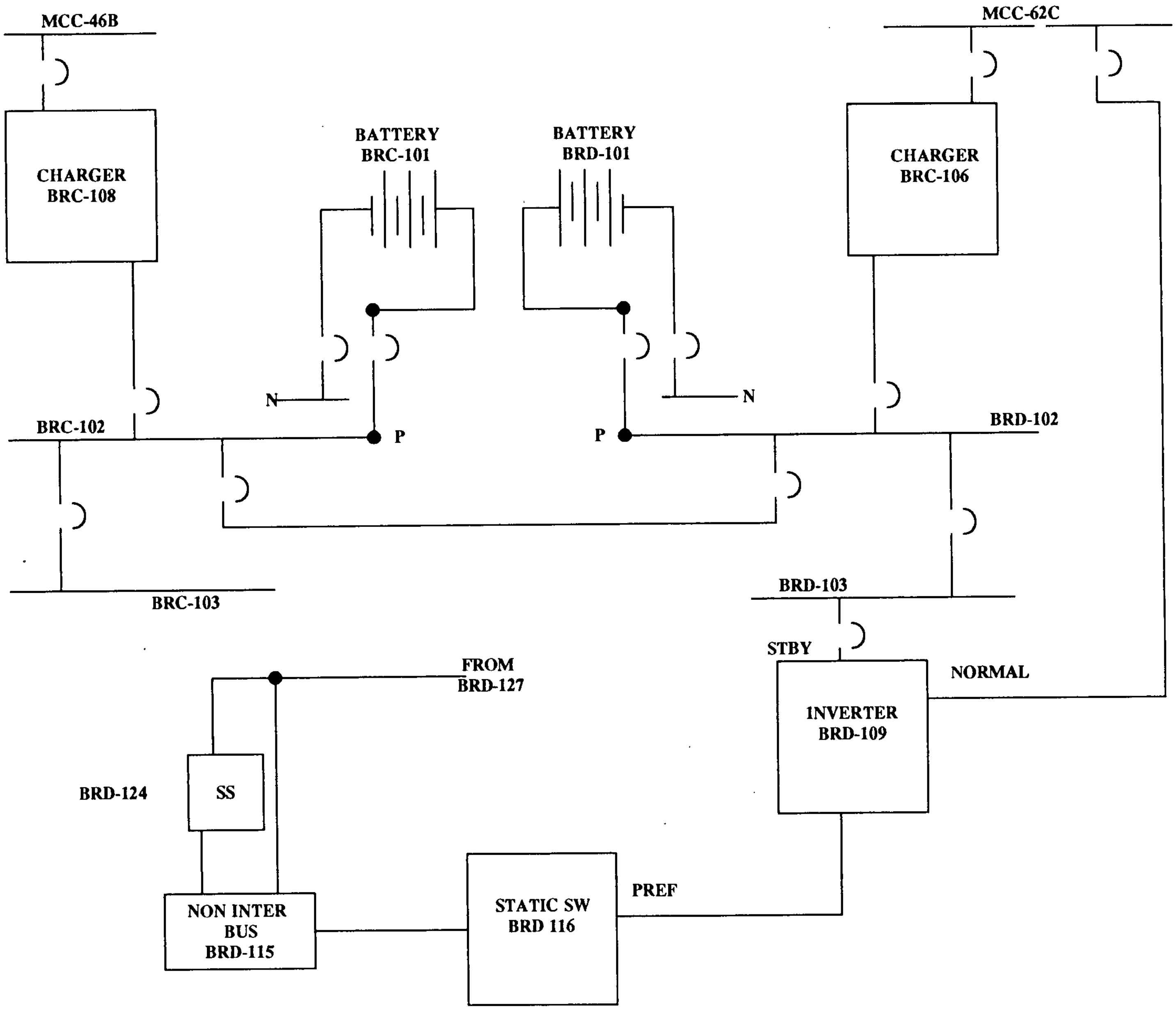

MCC-46B
MCC-62C
CHARGER
BRC-108
BATTERY
BRC-101
BATTERY
BRD-101
CHARGER
BRC-106
N
P
P
N
BRC-102
BRD-102
BRC-103
BRD-103
STBY
NORMAL
FROM
BRD-127
INVERTER
BRD-109
BRD-124
SS
NON INTER
BUS
BRD-115
STATIC SW
BRD 116
PREF

Figure 3.2-20

3.2.1.6 High Pressure Safety Injection

Function

The high pressure safety injection (HPSI) system is a subsystem of Kewaunee's emergency core cooling system (ECCS). It can be operated in the high pressure injection (HPI) mode, and the high pressure recirculation (HPR) mode.

The HPSI system provides emergency core cooling in the event of a break in either the reactor coolant system (RCS) or the secondary system. The purpose of the injection mode of operation is to terminate any reactivity increase following the postulated accidents, cool the core, and replenish coolant from the primary system.

Upon depletion of the refueling water storage tank (RWST), the recirculation mode of operation is initiated to provide long-term cooling by utilizing the water that accumulates in the containment sump.

Description

The HPSI System is one of several ECCSs that work together to perform two basic functions. The primary function of the ECCS following a loss of coolant accident (LOCA) is to remove the stored energy and fission product decay heat from the reactor core so that fuel damage is avoided or minimized. A secondary function is to provide shutdown capability for four design basis accidents by means of chemical boric acid injection. The systems that make up the ECCS include: The residual heat removal (RHR) system, safety injection (SI) system and the SI accumulators. The normal operating configuration is shown in Figure 3.2-21.

The following is a discussion on the high pressure ECCS operation.

High pressure injection (HPI) - During normal plant operation, HPI is in a standby mode that allows accident response with a minimum of active component operation. With HPI in the standby mode, the cold leg injection isolation valves, SI-11A/B are open and the reactor vessel injection isolation valves are closed. The supply valves from the RWST and boric acid tank (BAT) are closed, and the BAT selector switch is in the position for the BAT selected for HPI. The isolation valves on the recirculation line to the RWST, SI-208 and SI-209, are open to provide a miniflow path for the SI pumps. HPI is used to fill the accumulators with borated water when the plant is shut down.

On receipt of an SI signal, the SI pumps start automatically. The BAT suction isolation valves SI-2A and SI-2B both open to supply concentrated boric acid to the SI pumps. When RCS pressure drops below 2210 psig, the SI pumps begin to deliver flow to the RCS.

When the BAT reaches a low-low level of 10%, SI pump suction valves SI-4A and SI-4B from the RWST open and SI pump suction valves SI-2A and SI-2B automatically close. The common supply valve, SI-3, from the BAT to both SI pumps is open with the breaker locked in the off position.

Manually operated SI pump discharge cross-connect valves SI-8A and SI-8B are normally open to assure flow to both RCS cold legs if one pump fails to start. Manual throttling valves SI-10A and SI-10B and pressure reducing orifices on each cold leg injection line prevent runout flow and assure balanced flow to the RCS cold legs even for cold leg break LOCAs.

Reactor vessel injection isolation valves SI-15A and SI-15B are normally closed.

HPI Valve Interlocks - The BAT supply valves to the SI pumps, SI-2A/B, and the RWST supply valves, SI-4A/B, are interlocked with BAT level. When the BAT that is lined up for HPI reaches 10% level, SI-2A and SI-2B close and SI-4A and SI-4B open. In order for this interlock to work correctly for either set of valves, the following conditions must be met:

1. The BAT selector switch must be positioned to the BAT that will supply HPI.

2. The valves' control switches must be in the auto position.

3. Two relay coils must receive signals to energize. In order for each relay coil to become energized, one out of two level transmitters must transmit low level signals. Therefore, the BAT level interlock logic is 1 out of 2 low level signals from 2 independent transmitters.

High Pressure Recirculation (HPR) - After the injection phase of a LOCA, if the RCS pressure remains above 140 psig, HPR is used to provide long-term cooling. During the LOCA, HPI will continue to supply high pressure water to the RCS cold legs until the RWST level reaches 37%. At this time the operators are instructed to begin Emergency Operating Procedure ES-1.3, Transfer to Containment Sump Recirculation. The major operator actions in this procedure are:

1. Establish component cooling flow to the residual heat exchangers by opening valves CC-400A and B.

2. Stop train B SI pump and RHR pump.

3. Close SI pump to RWST recirculation valves, SI-208 and SI-209.

4. Open containment sump B isolation valve SI-350B.

5. Close RWST to RHR pump B suction isolation valve SI-300B.

6. Open containment sump B isolation valve, SI-351B.

7. Close residual heat exchanger B flow control valve, RHR-8B.

8. Start RHR pump B.

9. If RCS pressure is above 140 psig, establish recirculation flow with SI pump B.

10. Close SI pump B suction isolation valve, SI-5B.

11. Open RHR supply to SI pump B suction valve, RHR-300B.

12. Start SI pump B.

One SI/RHR train is now aligned for high pressure recirculation, and the other HPI train is aligned for cold leg injection taking suction from the RWST. This "split train" operation will continue until the RWST reaches 10%. The remaining SI/RHR train is then aligned in the standby mode for high pressure recirculation.

HPR Valve Interlocks - The SI pump recirculation valves to the RWST, SI-208 and SI-209, are interlocked with the containment sump isolation valves SI-350A/B and SI-351A/B. The sump isolation valves cannot be opened until SI-208 or SI-209 is closed. This is to prevent the release of radioactive water to the RWST from the containment sump.

The RHR supply valves to the SI pumps suction, RHR-300A/B can only be opened if RHR discharge pressure is below 210 psig and the associated SI pump suction valve, SI-5A or SI-5B, is closed. These interlocks are to prevent overpressurization of the suction piping and to prevent the pumping of contaminated sump water directly to the RWST through SI-5A or SI-5B.

<u>Fault Trees</u>

There are seven fault trees for HPI and three fault trees for HPR. They are:

High Pressure Injection (HI0) - HI0 is required for a medium break LOCA (MLO). HI0 provides low flow, high pressure concentrated boric acid from the BATs and borated water from the RWST to the primary system following a safety injection signal. The mission time for HI0 is 3.5 hours. The success criterion for HI0 is one of two HPI trains injecting the contents of one BAT, successfully switching to RWST suction, and delivering the contents of the RWST to the intact RCS cold leg. Upper Plenum injection with the HPSI is not modeled. Reactor vessel injection valves SI-15A/B are normally closed. Reactor vessel injection is not normally used during LOCAs because of phenomenological uncertainty over the effectiveness of upper head injection in smaller break LOCAs.

High Pressure Injection (HI1) - HI1 is similar to HI0. It is used for a steam generator tube rupture (SGR), however, instead a MLO. The difference is that there are two intact loops for a SGR and only one for a MLO. The mission time for HI1 is 3.5 hours. The success criterion for HI1 is one out of two HPI trains injecting the contents of one BAT, successfully switching to RWST suction, and delivering the contents of the RWST to one of two RCS cold legs.

High Pressure Injection (HI2) - HI2 is similar to HI0. It is used for a small LOCA, however, instead of a MLO. The difference is that HI2 allows for manual starting of any components that did not initiate on the automatic SI signal. Time is available in a small break LOCA to manually initiate, in the control room, a component capable of starting that did not initiate during the automatic SI signal. The mission time for HI2 is 3.5 hours. The success criterion for HI2 is the same as that for HI1.

High Pressure Injection (HI3) - HI3 is similar to HI0. It is used for a steam line break (SLB), however, instead of a MLO. Similar to HI1, HI3 has two intact loops for injection. Also, because the only use for SI in an SLB is boron addition, only the BAT injection is needed, and RWST injection is not required. Although the actual time required to inject one BAT is less than ten minutes, a conservatively high value of 3.5 hours is used. The success criterion for HI3 is one of two HPI trains injecting the contents of one BAT to one of two RCS cold legs.

High Pressure Injection (HI4) - HI4 is the same as HI1 except that HI4 has a 24 hour mission time. HI4 is used in the Interfacing Systems LOCA sequence (ISL). The success criterion for HI4 is the same as for HI1.

High Pressure Injection (HPI) - HPI is the manual initiation of SI for the purposes of bleed and feed in the event of a loss of heat sink. The descriptions in the HI0 section apply to the HPI as well. The mission time for HPI is 3.5 hours. The success criterion for HPI is the same as that for HI1.

High Pressure Injection (HPID) - HPID is the same as HPI except SI pump A is unavailable due to loss of BRA-104. The mission time for HPID is 3.5 hours. The success criterion for HPID is one available HPI train injecting the contents of one BAT, successfully switching to RWST suction, and delivering the contents of the RWST to one of two RCS cold legs.

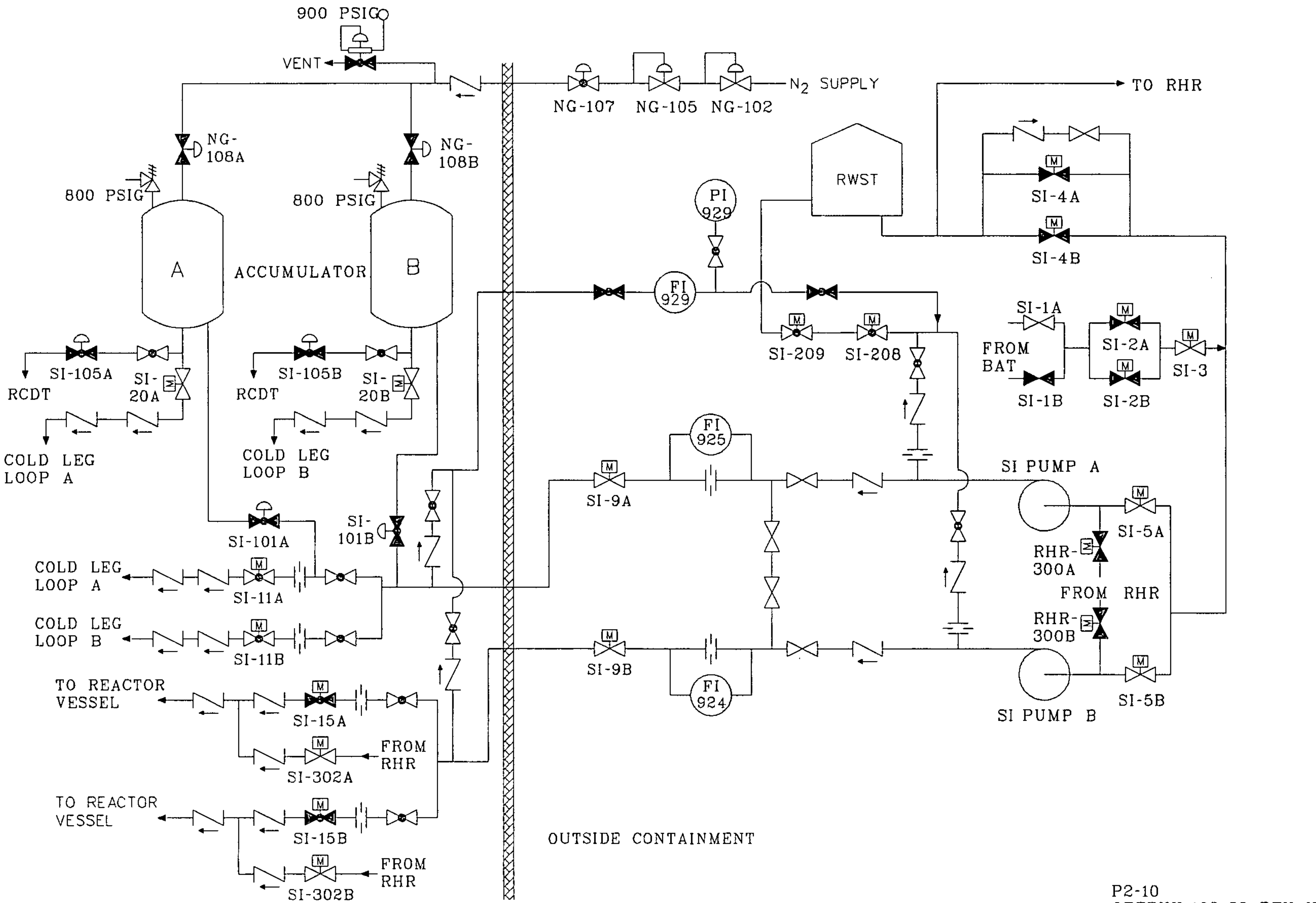
High Pressure Recirculation (HR0) - After the injection phase of a LOCA, if the primary system is above 140 psig, HPR is initiated. Coolant spilled from the break and water collected from the containment spray is cooled and returned to the RCS. HR0 is arranged so that the RHR pumps take suction from the containment sump B. The water is thus cooled as it passes through the residual heat exchangers and is delivered to the suction of the SI pumps. The SI pumps then inject the high pressure coolant into the RCS cold legs. The mission time for HR0 is 20.5 hours. The success criterion of HR0 is one out of two HPR trains taking suction from the containment sump line and delivering flow to the intact RCS cold leg. It is assumed in this fault tree and in HR1 and HR2 that auxiliary building basement cooling is needed for operation of the safety injection pumps when in the recirculation mode (refer to description of fault tree ABBC in the miscellaneous system description section 3.2.1.12 and Figures 3.2-22 and 3.2-23).

High Pressure Recirculation (HR1) - HR1 is similar to HR0 except HR0 is used only for a MLO, which assumes only one intact loop, where HR1 is for all other initiating events requiring HPR (except loss of a 125V DC bus) that assume two intact loops. The mission time for HR1 is 20.5 hours. The success criterion of HR1 is one out of two HPR trains taking suction from its containment sump suction line and delivering flow to one of two RCS cold legs.

High Pressure Recirculation (HR2) - HR2 is similar to HR0 except HR2 is for a loss of one 125V DC bus. HR2 assumes BRA-104 is lost and therefore, SI pump A is unavailable. The mission time for HR2 is 20.5 hours. The success criterion for HR2 is the same as that for HR1.

FIGURE 3.2-21

SAFETY INJECTION SYSTEM



P2-10
OPERXK-100-28, REV. X
OPERXK-100-29, REV. L

FIGURE 3.2-22

AUXILIARY BUILDING VENTILATION

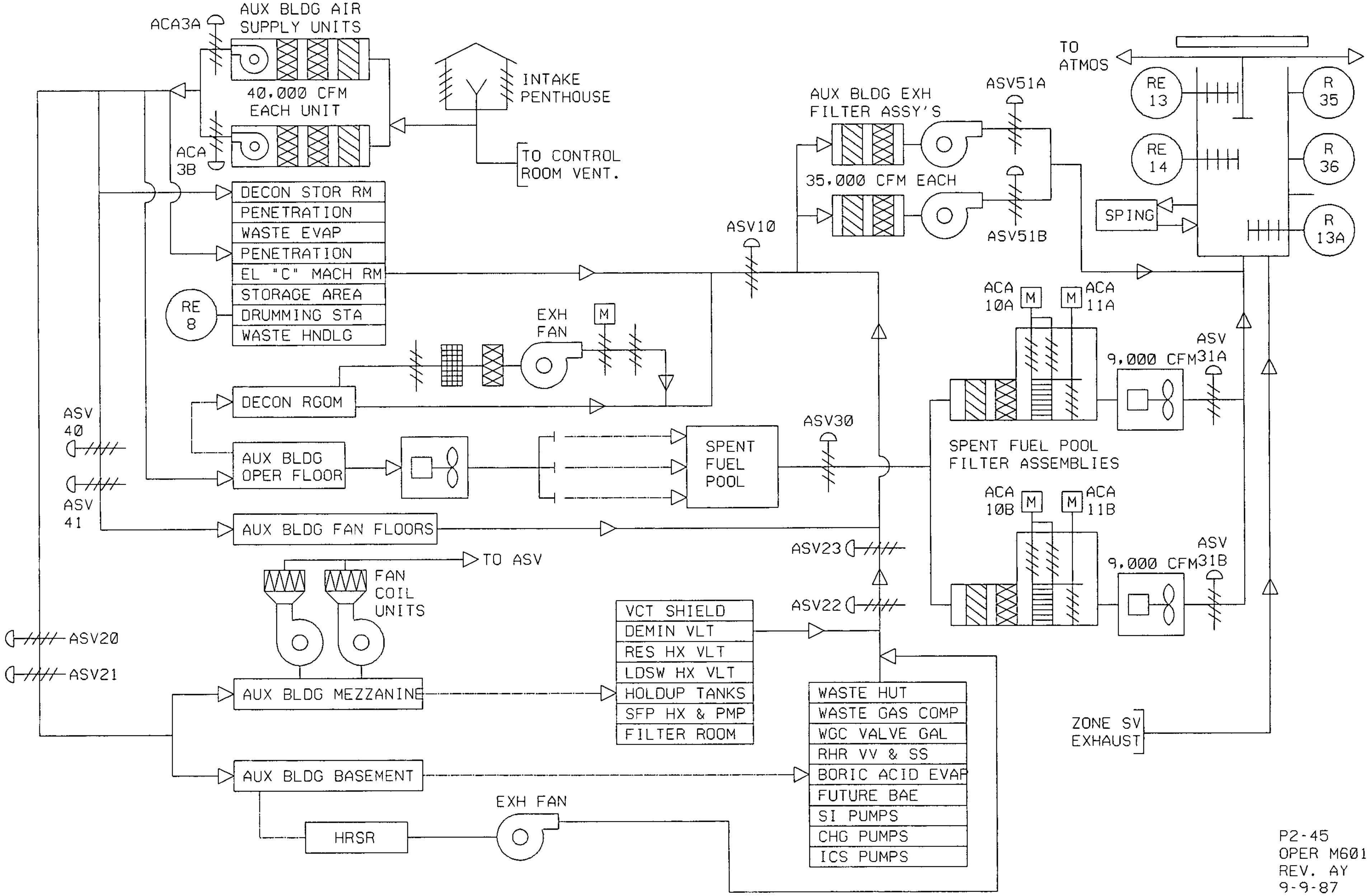


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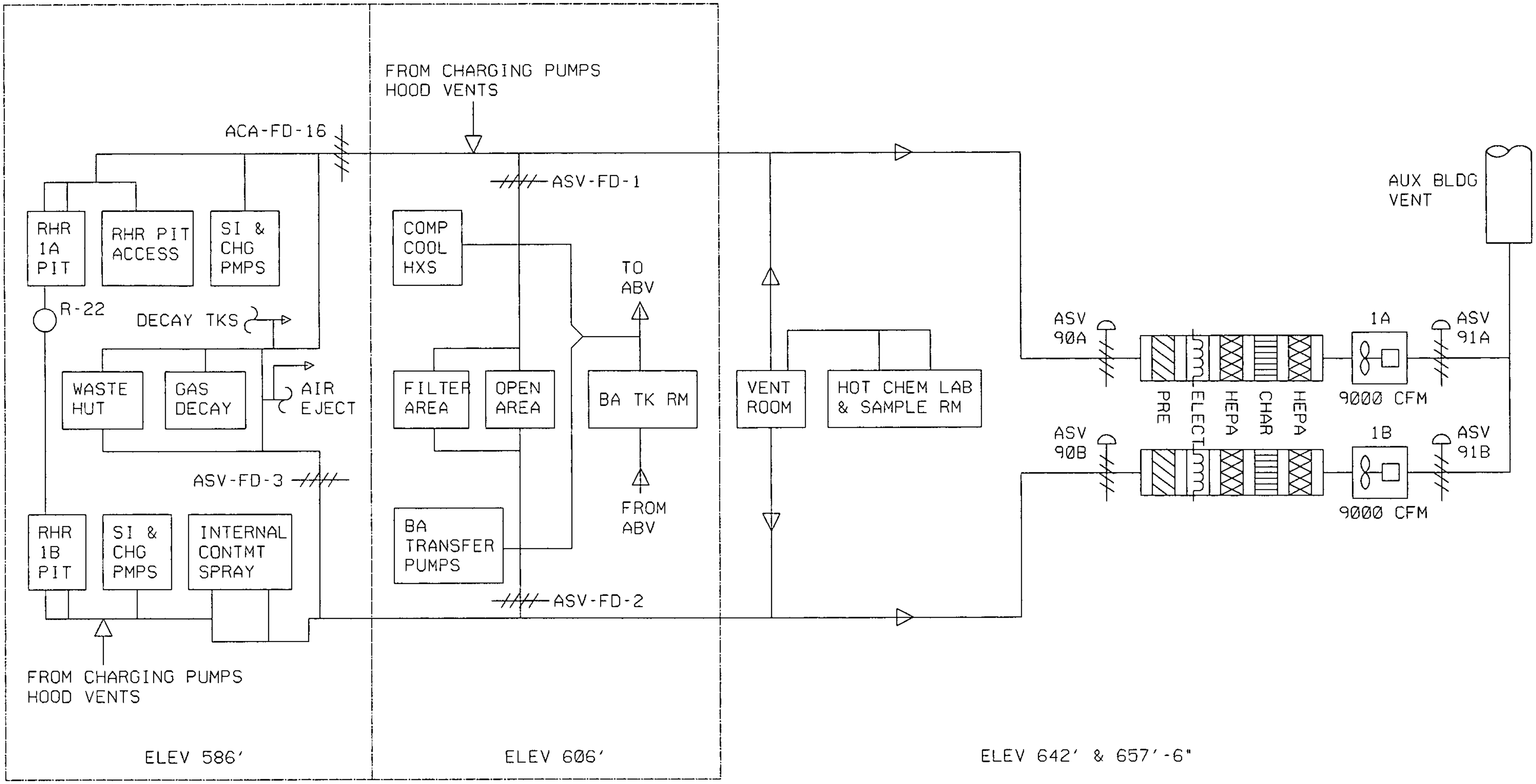
FIGURE 3.2-23
AUXILIARY BUILDING SPECIAL VENTILATION



P2-47
OPER M604
REV. AC
9-15-87

3.2.1.7 Internal Containment Spray

Function

The internal containment spray (ICS) system is designed to spray cool water into the containment atmosphere following a design basis accident. The spray provides sufficient heat removal capability to maintain the post-accident containment pressure below its design value. In addition, the spray is effective in scrubbing fission products from the containment atmosphere. Sodium hydroxide (NaOH) is added to the spray solution for pH adjustment. The resulting alkaline pH of the spray fluid enhances its ability to scavenge iodine fission products from the containment atmosphere and precludes the possibility of stress corrosion cracking of the stainless steel components that are exposed to the containment sump fluid.

Description

The ICS system does not operate under normal plant conditions other than for required testing. The ICS is designed to spray 2,600 gpm of borated water with NaOH added into containment on a coincidence of high-high containment pressure one-out-of-two, three times, or on manual initiation by the control room operator. The ICS system operates first in the injection phase and then the recirculation mode as dictated by containment pressure and temperature conditions. Figure 3.2-24 is a detailed flow diagram of the system.

The ICS system consists of two redundant spray trains each capable of delivering 1300 gpm. During the injection phase borated water is supplied from the refueling water storage tank (RWST) by a common suction line to each pump header.

The ICS system is also capable of supplying concentrated sodium hydroxide (NaOH) solution to the containment atmosphere. The NaOH enhances the ability of the spray to scavenge iodine fission products from the containment atmosphere during the washdown and ensure an alkaline pH for the containment sump solution. The alkaline pH of the containment sump water minimizes the evolution of iodine and the occurrence of chloride and caustic stress corrosion on mechanical systems and components exposed to the sump fluid.

Concentrated NaOH is provided to the ICS pumps from the caustic additive standpipe via two parallel air operated valves (AOVs), CI-1001A and CI-1001B, which open automatically on an ICS initiation.

During the recirculation phase the residual heat removal (RHR) system supplies the suction of the ICS pumps through motor operated valves RHR-400A/400B (MOVs) which are opened by the control room operator. Also in preparation for the recirculation phase motor operated valves (MOVs) ICS-2A/2B are closed by the control room operator to isolate the RWST from the ICS system. During both phases of operation, each ICS pump discharges through two parallel MOVs. These MOVs, ICS-5A/5B and ICS-6A/6B, are normally closed and open automatically on an ICS initiation. The discharge path is then through locked open manual valves, to the spray ring headers and spray nozzles which are located in the containment dome.

Test line isolation AOVs ICS-201 and ICS-202 are normally open valves used for pump surveillance testing. These valves would close automatically upon receipt of a containment isolation signal, which would occur prior to ICS system actuation.

Fault Trees

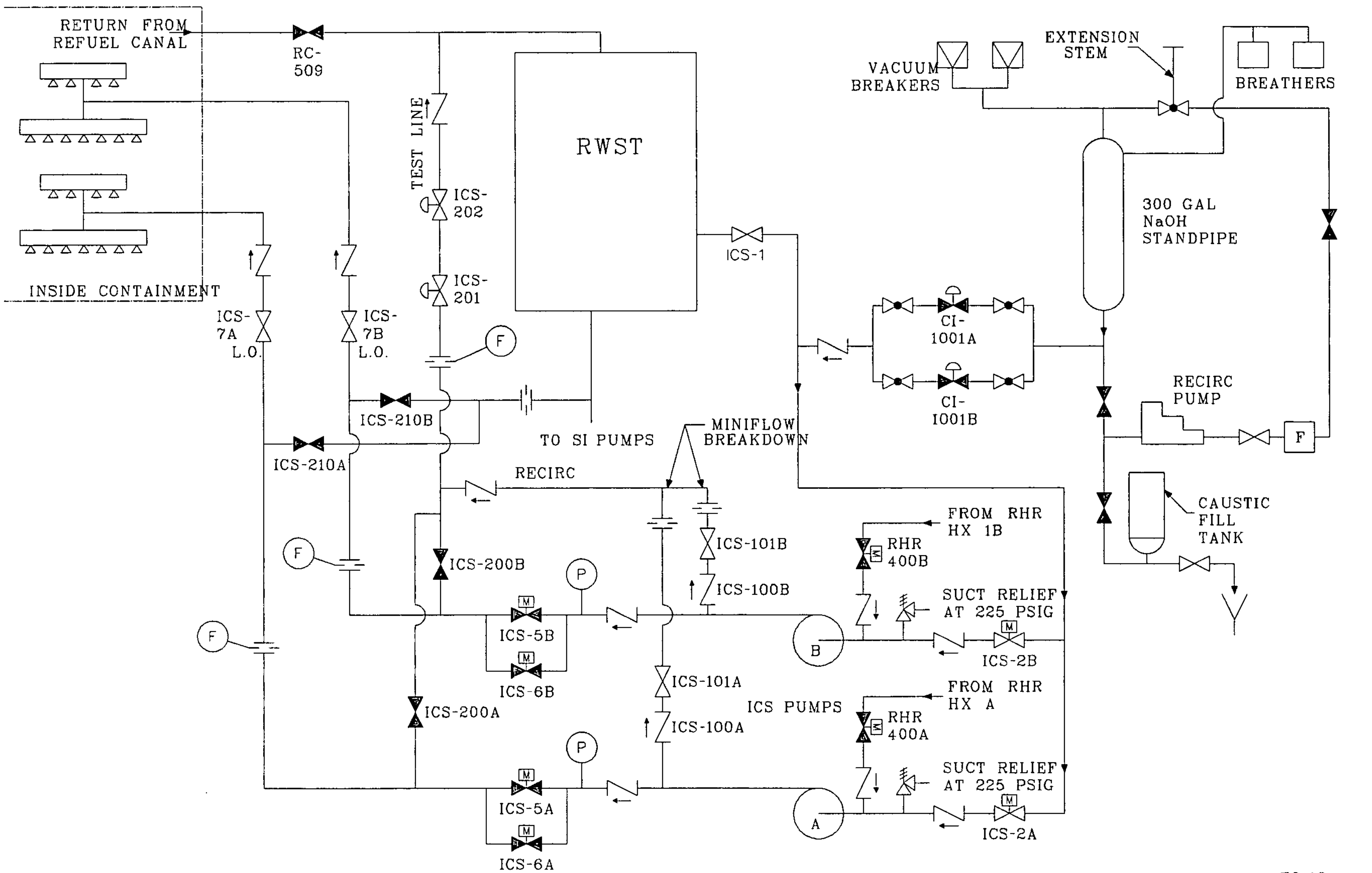
Two fault trees were developed for the ICS system. It is assumed in the models for both fault trees that auxiliary building basement cooling is needed for the operation of the containment spray system when in the recirculation mode (refer to description of fault tree ABBC in the miscellaneous system description section 3.2.1.12, and Figures 3.2-22 and 3.2-23). The two fault trees are:

ICS - The ICS fault tree is used to represent the ICS system in plant damage states quantification for all events except loss of DC power, loss of service water, loss of component cooling water, and station blackout events in which power is not restored in 24 hours. This fault tree models both the injection and recirculation phases of ICS operation. The success criterion is one of two trains of ICS. The mission time is 24 hours with 1 hour designated for injection and 23 hours for recirculation. The success of the recirculation phase is dependent on the operations crew successfully transferring low head safety injection to the containment sump recirculation mode.

CSD - The CSD fault tree is identical to the ICS fault tree except that this fault tree is used for loss of DC power events and assumes the loss of BRA-104 and therefore the unavailability of the A ICS train.

FIGURE 3.2-24

INTERNAL CONTAINMENT SPRAY SYSTEM



P2-12
OPERM-217
REV. Y

3.2.1.8 Low Pressure Safety Injection

Function

The low pressure safety injection (LPSI) system and the accumulators are subsystems of Kewaunee's emergency core cooling system (ECCS). It can be operated in the low pressure injection (LPI) mode and the low pressure recirculation (LPR) mode. The LPSI system and the accumulators provide emergency core cooling in the event of a break in either the reactor coolant system (RCS) or the secondary system. The purpose of the injection mode of operation is to terminate any reactivity increase following the postulated accidents, cool the core, and replenish coolant lost from the primary system. Upon depletion of the RWST, the recirculation mode of operation is initiated to provide long term heat removal by utilizing the water that accumulates in the containment sump.

Description

The residual heat removal (RHR) system/LPSI System is one of several ECCSs which work together to perform two basic functions. The primary function of the ECCS following a loss of coolant accident (LOCA) is to remove the stored energy and fission product decay heat from the reactor core so that fuel damage is minimized. A secondary function of the ECCS is to provide shutdown capability for four design bases accidents by means of chemical boric acid injection. The systems that make up the ECCS include the RHR system, the safety injection (SI) system, and the SI accumulators. The normal operating configuration is shown in Figure 3.2-25. In both the injection and recirculation modes of LPSI, water is injected into the reactor vessel.

The following discussion of the low pressure ECCS operation is presented according to those ECCS subsystems and components that are functional over each RCS pressure range that exists during the course of the accident. This breakdown of subsystems is as follows:

1. Accumulators
2. Low pressure injection.
3. Low pressure recirculation.

Accumulators - During power operation, the accumulators are in a stand-by mode that allows accident response without active component operation. In this configuration, the accumulators are filled with borated water (minimum boron concentration of 1900 ppm) and pressurized with nitrogen gas to a pressure of approximately 750 psig. The accumulator discharge isolation valves (SI-20A/B) are open with power locked out at RCS pressures of 1000 psig or greater.

In the event of a large LOCA, the two accumulators inject borated water into the cold legs when the RCS system pressure decreases to a value below the accumulator nitrogen gas pressure of 750 psig. Each accumulator injects into a separate RCS cold leg through a discharge isolation valve (SI-20A/B) in series with two check valves.

For the maximum large break size (double ended cold leg guillotine) LOCA, the accumulator is emptied in about 42 seconds after the break occurs seconds. For smaller break sizes, the RCS depressurization rate is decreased and initial accumulator injection is delayed and the elapsed time for injection is increased.

Low Pressure Injection - During normal plant operation, LPSI provides no direct support to the reactor plant. The system is in a stand-by mode that allows accident response with a minimum of active component operation. With LPSI in a stand-by mode, the RWST isolation valves (SI-300A/B), the flow control valves (RHR-8A/B), and the vessel injection isolation valves (SI-302A/B) are open. On receipt of a safety injection signal, the RHR pumps start automatically. When the RCS pressure drops below 150 psig (the RHR pump shutoff head), water from the RWST is pumped into the reactor vessel.

At RCS pressures greater than 600 psig, the RHR pumps discharge flow passes through a miniflow return line from the discharge side of residual heat exchangers to the pump suction lines.

A cross-tie between the two injection trains at the discharge of the flow control valves (RHR-8A/B) is normally isolated, thereby preventing flow from each pump to the opposite train.

Low Pressure Recirculation - During a large LOCA, LPSI will continue to supply low pressure water to the reactor vessel until the RWST level reaches 37%. When the RWST reaches 37% level, the operators are instructed to begin Emergency Operating Procedure ES-1.3, Transfer to Containment Sump Recirculation. The major operator actions in this procedure required to transfer to containment sump recirculation are:

1. Establish component cooling flow to the residual heat exchangers by opening valves CC-400A and B.

2. Stop train B of safety injection (SI pump and RHR pump).

3. Close SI pump recirculation valves, SI-208 and SI-209.

4. Open containment Sump B isolation valve, SI-350B.

5. Close RWST to RHR Pump B suction isolation valve, SI-300B.

6. Open containment sump B isolation valve, SI-351B.

7. Close residual heat exchanger B flow control valve, RHR-8B.

8. Start RHR Pump B.

9. Establish 1500 GPM recirculation flow by throttling open RHR 8B.

At this point, one train of LPSI is aligned for containment sump recirculation. The other train is aligned for vessel injection, taking suction from the RWST. This "split train" operation will continue until the RWST reaches 10%. The remaining LPSI train is then aligned in standby for containment sump recirculation.

Low Pressure Recirculation Support - The success of high pressure recirculation is dependent upon the success of the low pressure recirculation system. Containment spray recirculation is also dependent upon low pressure recirculation. If high pressure recirculation and/or containment spray recirculation are needed, cross connecting valves are opened to provide a portion of the low pressure recirculation flow to the suction of the high pressure safety injection (HPSI) and/or containment spray (ICS) pumps. In order to supply flow to the HPSI pumps motor operated valves (MOV) RHR-300A or B are opened from the control room to supply flow to the A or B pump respectively. MOV RHR-400A or B are opened from the control room to supply flow to ICS pump A or B respectively.

Fault Trees

There are three fault trees for low pressure injection, three fault trees for low pressure recirculation and 2 fault trees that are subtrees in the high pressure SI recirculation and containment spray recirculation fault trees, they are:

LI1 - The LI1 fault tree is used to represent LPSI in the injection mode for large break LOCAs. The success criterion for LI1 is one out of two trains of LPSI delivering flow from one RHR pump, taking suction from the RWST, to the reactor vessel. Mission time is assumed to be one hour (time to empty the RWST).

LI2 - The LI2 fault tree is used to represent LPSI in both medium and small break LOCA events. In both these events, the RHR (LPSI) pumps are manually stopped by procedure because RCS pressure would exceed the discharge pressure of the pumps. Following operator actions to successfully cooldown and depressurize the RCS, LPSI needs to be manually initiated by the operators. It is this manual initiation of LPSI that is modeled in LI2. The success criterion for LI2 is one of two trains and the mission time is one hour.

LPI - Fault tree LPI is similar to fault tree LI1 except that it is used as a subtree in the fault tree LR2, which is one of the LPSI recirculation trees described below. LPI models the failure of both LPSI trains in the injection mode which is one failure mode for low pressure recirculation (i.e. if the portion of the system common to LPI and LPR fails in the LPI mode it also fails in the LPR mode).

LR1 - Fault tree LR1 is used for large break LOCAs, medium break LOCAs and small break LOCAs. LR1 is the only LPSI recirculation fault tree for large break LOCAs whereas there are two LPR fault trees in medium and small break LOCAs (LR2 is the other). LR1 assumes that LPI is successful. The success criterion for LR1 is one of two RHR pumps taking suction from its containment sump suction lines and delivering flow to the reactor vessel. The mission time is 23 hours.

LR2 - Fault tree LR2 is used in medium and small break LOCAs after high pressure safety injection fails in the recirculation mode. Because there are several components that could fail high pressure recirculation but not LPR (such as the SI pumps). LPR may still be successful. Fault tree LPI is used as a subtree in LR2 to model the possible failure of both LPI trains. The success criterion and mission time are the same as those for LR1.

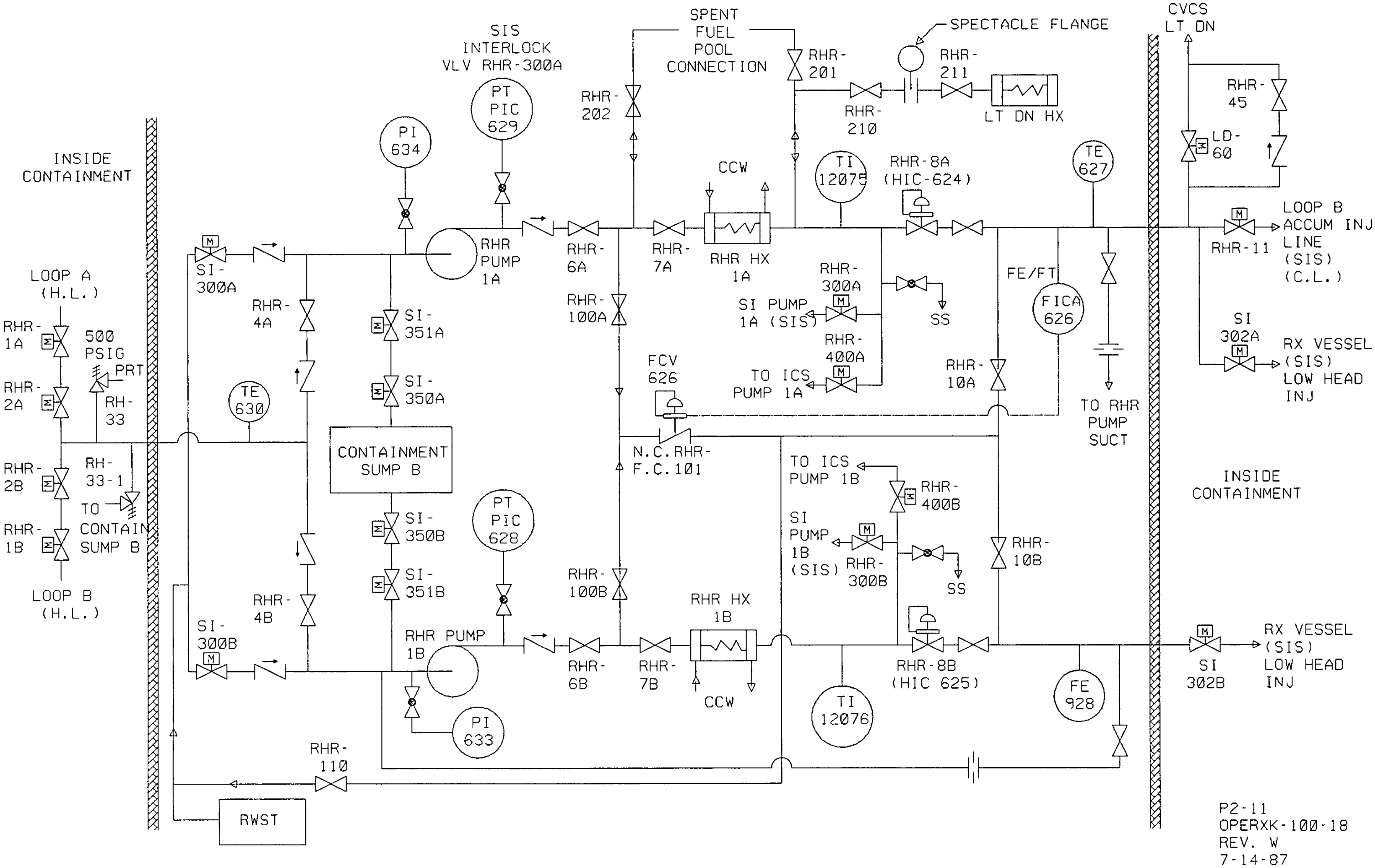
LR3 - LR3 is similar to LR1 except that it models the loss of BRA-104 and therefore all train A components. This fault tree is used in the plant damage states quantification for loss of DC bus events. The success criteria and mission time are the same as those for LR1.

RHRA, RHRB - Fault trees RHRA and RHRB are subtrees in the fault trees for high pressure safety injection (HPSI) in the recirculation mode (fault trees HR0, HR1 and HR2), and in the fault trees for the containment spray (ICS) system in the recirculation mode (fault trees ICS and CSD). The LPR system supplies water from the containment sump to the suction of the SI and ICS pumps in the event that HPSI or ICS recirculation are needed. The success criterion for RHRA and RHRB is either one of the two trains being successful in delivering flow from the containment sump to the cross connect valves to HPSI and/or ICS recirculation. The mission time is 20.5 hours to be consistent with the HPSI recirculation fault trees.

In all 8 fault trees, loss of heating, ventilation and air conditioning (HVAC) to the RHR pumps is directly modeled. The RHR pumps are located in small pump pits with their own fan coil units. It is assumed that if the fan coil unit fails, the RHR pump fails.

FIGURE 3.2-25

RHR SYSTEM - INJECTION



3.2.1.9 Main Feedwater

Function

The main feedwater (MFW) system, along with the condensate (CD) system, returns condensed steam from the main turbine condenser and drains from the regenerative feedwater heating cycle to the Steam Generators (SGs). The MFW system accepts flow from the CD pumps and the heater drain pumps, raises the temperature and pressure of the fluid, and then delivers the required flow rate to each SG. The SGs use the high temperature pressurized water to produce dry saturated steam to drive the turbine generator. The SGs are the interface between the reactor coolant system and the secondary system. The MFW system provides a sufficient quantity of high pressure feedwater to the SGs for water level control during plant startup, power operation, and plant shutdowns. The CD pumps provide the necessary net positive suction head (NPSH) for the MFW pumps.

Description

The MFW system is a closed system which operates in the secondary system. The CD pumps pump water collected in the condenser hotwells through four pairs of low pressure feedwater heaters and provide, in conjunction with the heater drain pumps the NPSH required by the MFW pumps. The outlets of the CD pumps combine into a single header. Condensate passes through the air ejector condenser, the gland steam condenser, and the low pressure feedwater heaters before it is delivered to the MFW pump suction. Each MFW pump discharges through a motor-operated gate valve FW-2A or FW-2B, and a flow nozzle to a common 22 inch header. The feedwater is then sent through a parallel pair of high pressure feedwater heaters. The parallel flow paths are then joined together to allow the feedwater to be mixed properly and equalize temperature. The flow path then splits into two 16 inch lines that contain feedwater control stations and flow nozzles to feed both SGs. Each 16 inch line contains a MFW flow control valve, FW-7A or FW-7B, and a bypass feedwater flow control valve, FW-10A or FW-10B, in parallel with the main control valve. Downstream of these control valves are motor-operated feedwater isolation valves, FW-12A or FW-12B. The piping then connects with the auxiliary feedwater piping and enters the SG. The normal operating configuration is shown on Figures 3.2-26, 3.2-27, and 3.2-28.

Fault Trees

Five fault trees were developed for the MFW system based on a review of the initiating events. The MFW fault trees are:

Fault Tree OM0 - This fault tree was developed for medium LOCA, small LOCA and interfacing systems LOCA events. The basis for this fault tree is that both SGs, feed lines and main steam lines are intact. The defined mission time is 24 hours. The success criterion is a 200 gpm flow rate to 1 of 2 SGs delivered from 1 of 2 FW trains.

Fault Tree OM1 - The basis for this fault tree is that one of two SGs is ruptured or faulted and applies to the steam generator tube rupture and steam line break events. Under these conditions, the operators are directed by emergency operating procedures to isolate the effected SG. The mission time is 24 hours. The success criterion is a 200 gpm flow rate to 1 of 1 SG delivered from 1 of 2 FW trains.

Fault Tree OM2 - This fault tree was developed for the loss of component cooling water and transients with MFW events. The mission time is 24 hours. The success criterion is a 200 gpm flow rate to 1 of 2 SGs delivered from 1 of 2 FW trains.

Fault Tree OM3 - This fault tree was developed for the loss of station and instrument air event. The mission time is 24 hours. The success criterion is a 200 gpm flow rate to 1 of 2 SGs delivered from 1 of 2 FW trains.

Fault Tree OM4 - This fault tree was developed for the loss of 125V DC bus event. The mission time is 24 hours. The success criterion is a 200 gpm flow rate to 1 of 2 SGs delivered from 1 of 2 FW trains.

FIGURE 3.2-26

MAIN FEEDWATER SYSTEM

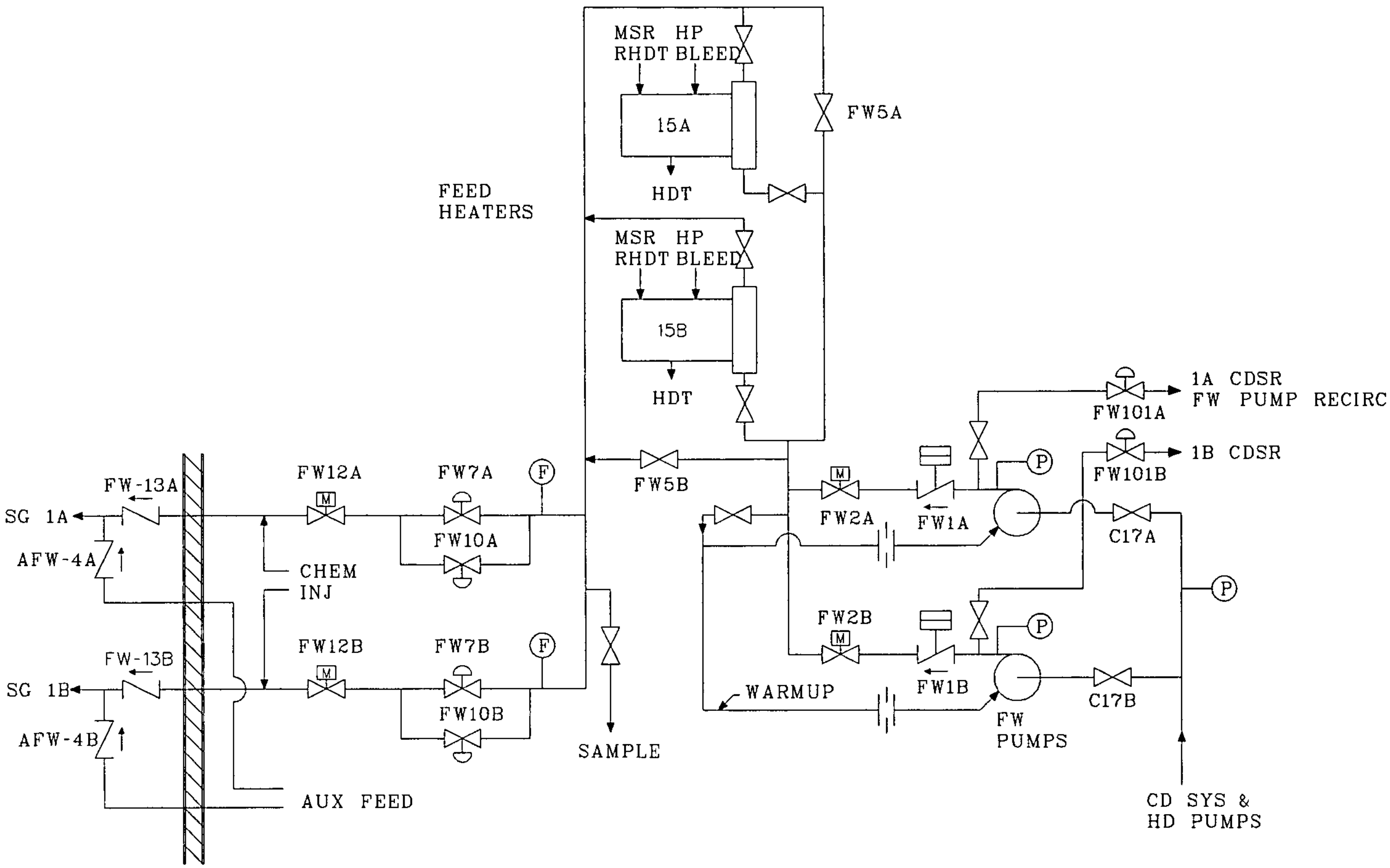
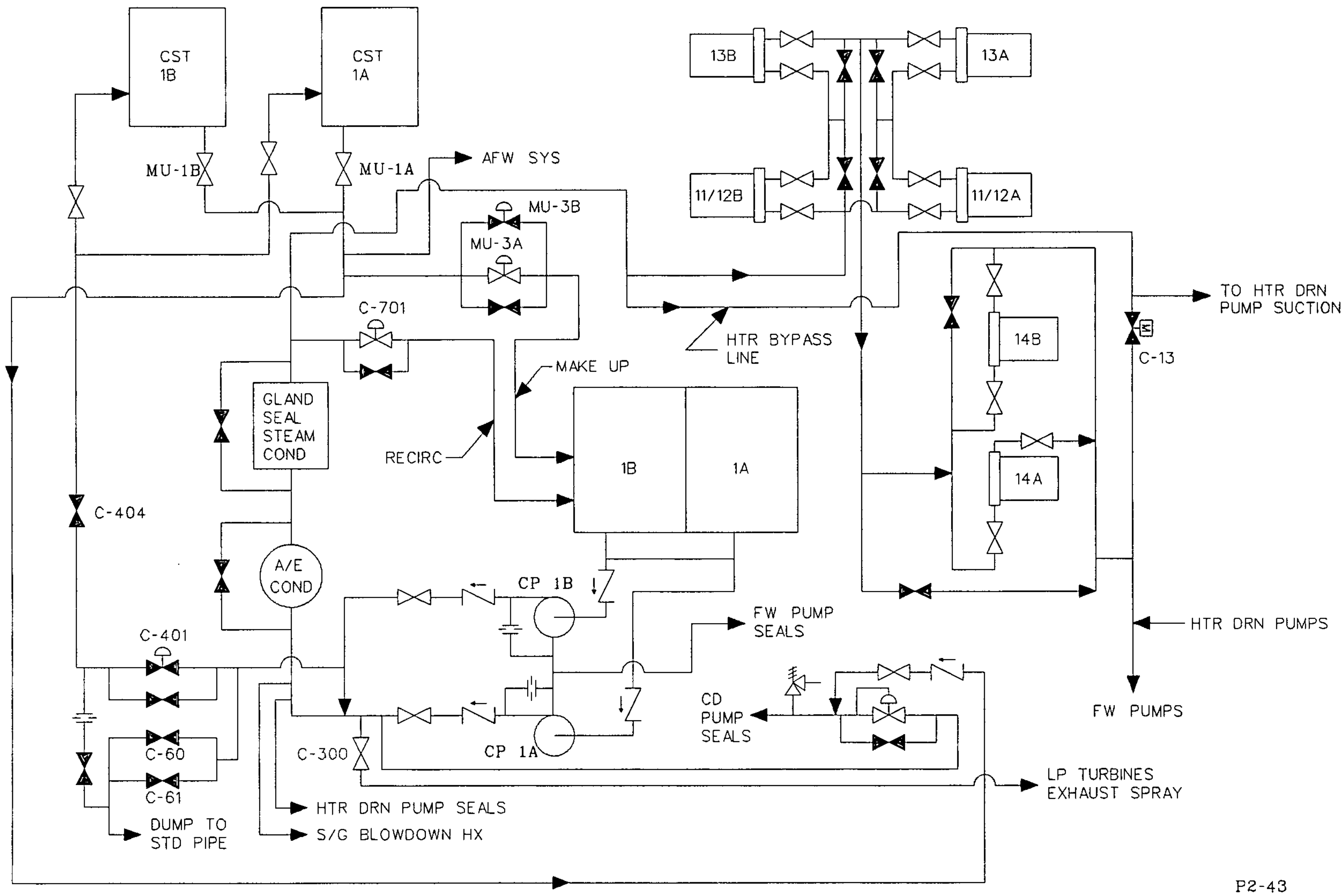


FIGURE 3.2-27

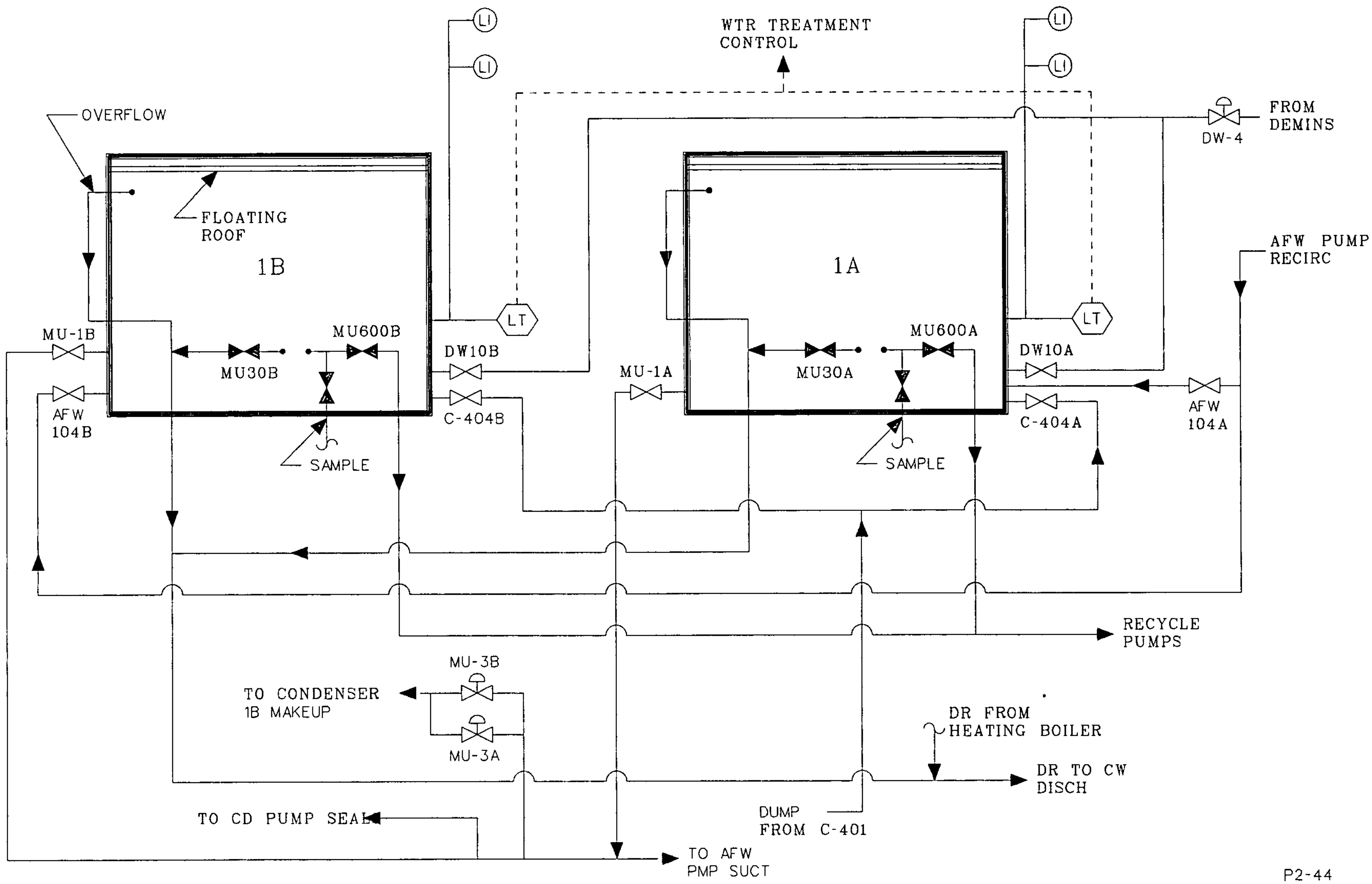
CONDENSATE SYSTEM



P2-43
OPERM-204
REV. GN

FIGURE 3.2-28

CONDENSATE STORAGE TANKS



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3.2.1.10 Reactor Protection

Function

The protective actions initiated by the reactor protection system (RPS) are broadly classified into two major categories, reactor trips and actuations of engineered safety features (ESFs). Therefore, the RPS protective functions are addressed from two functionally defined subsystems, the reactor trip system (RTS) and the ESF actuation system (ESFAS). These two subsystems perform all of the safety related actions associated with the RPS.

The RTS functions to prevent reactor operation outside of prescribed safe operating limits. The limits of safe reactor operation are defined by the correlation of reactor power, reactor coolant system temperatures, pressure, and flow; pressurizer level and secondary system heat removal capability. The basic reactor operating philosophy is to define an allowable region of power, pressure and coolant temperature conditions. This allowable region is defined by the three primary tripping functions: the overpower delta-T trip, the overtemperature delta-T trip and the nuclear overpower trip. The operating region below these trip settings is designed so that no combination of power, temperature, and pressure could result in departure from nucleate boiling ratio (DNBR) less than 1.3 for any credible operational transient with all reactor coolant pumps in operation. Additional tripping functions to those stated above are provided to back up the primary tripping functions for specific abnormal conditions.

Rapid reactivity shutdown is provided by the insertion of rod cluster control assemblies (RCCA) by free fall. Duplicate series-connected circuit breakers supply all power to the control rod drive mechanisms (CRDM). The CRDMs must be energized for the RCCAs to remain withdrawn from the core. Automatic reactor trip occurs upon loss of power to the CRDMs. If the RTS receives signals indicative of an approach to unsafe operating conditions, the system actuates alarms, prevents control rod withdrawal, initiates load runback, and/or opens the reactor trip breakers. At various power levels, permissive signals are generated which permit the operator to block certain reactor trip signals where they are not required for safety.

In addition to the requirements for a reactor trip for anticipated abnormal transients, the plant is provided with adequate instrumentation and controls to sense accident situations and initiate the operation of necessary ESFs. The occurrence of a limiting fault, such as a loss of coolant accident or a steam line break, requires a reactor trip plus actuation of one or more of the ESFs in order to prevent or mitigate damage to the core and reactor coolant system components, and ensure containment integrity.

Generation of a safety injection (SI) trip signal results in a reactor trip, feedwater and containment isolation and emergency diesel generator startup. Once a SI signal is generated, the SI sequencer will sequentially energize safeguards equipment providing that power is available for the associated ESF bus and no conflict exists between blackout (BO) and SI sequences. This is to ensure proper loading of the diesel generators in the event that outside power supplies to the safeguards buses (5 and 6) are interrupted and the diesel generators must assume the loads on those buses. Equipment energized by the SI sequencer includes: SI, residual heat removal pumps, containment spray pumps (if diesel generator is supplying and a containment high pressure signal exists), shield building ventilation, service water pumps, containment fan coil units and dome fan, auxiliary feedwater pumps, component cooling pumps, instrument air compressors, control room air conditioner and auxiliary building fans.

Description

The overall RPS consists of: Foxboro process protection instrumentation system, nuclear instrumentation system, reactor protection, and safeguards logic relay cabinets, SI sequencer, and reactor trip switchgear. For reactor trip and engineered safety features actuation, the safeguards and reactor protection logic relay cabinets each contain two redundant logic trains, A and B, that are physically and electrically independent. The logic relay cabinets receive inputs from process instrumentation, nuclear instrumentation, field contacts, and directly from main control board switches. The following is a description of the RPS sub-systems.

Process Instrumentation System - Process instrumentation includes those devices that measure temperature, pressure, fluid flow and fluid level. A typical process instrumentation channel includes a sensor, loop power supply, signal conditioning devices, bistables, indicators, recorders, alarm actuating devices and controllers necessary for operation of the nuclear steam supply system as well as for monitoring the plant and providing initiation of protective functions upon approach to unsafe plant conditions. The Foxboro process instrumentation racks house the process instrumentation after the sensors and supply outputs to the reactor protection relay racks and the safeguards relay racks.

Nuclear Instrumentation System - The nuclear instrumentation system (NIS) uses information from instrumentation channels to provide protection at three discrete power levels. Each range of instrumentation, source, intermediate and power, provides the necessary overpower reactor trip protection required during operation in that range. The overlap of instrument ranges provides reliable continuous protection beginning with source level through the intermediate and power level.

Logic Relay Racks - The reactor protection system and safeguards system logic relay racks perform logic decisions and cause accident mitigating actuations based upon process or nuclear instrumentation system bistable settings that may have been exceeded as well as the condition of various bypasses and permissive interlocks.

Reactor Trip Switchgear - Each of the reactor protection trains is capable of opening a separate and independent reactor trip breaker, RTA and RTB, and a bypass breaker, BYB and BYA (refer to Figure 3.2-29). The two trip breakers in series connect three phase AC power from the control rod drive motor-generator sets to the control rod drive power cabinets. During plant power operation, a DC undervoltage coil on each reactor trip breaker holds a trip plunger out against its spring, allowing the power to be available at the rod control power supply cabinets. For reactor trip, a loss of DC voltage to the undervoltage coil, as well as energization of the shunt trip coil, trips open the breaker. When either of the trip breakers opens, power is interrupted to the rod drive power supply, and the control rods fall by gravity into the core. The rods cannot be withdrawn until the abnormal condition which initiated the trip is corrected.

Sequencer - The sequencer contains combinational and sequential logic circuitry that is used to accomplish equipment loading on an ESF bus, inhibit automatic starting of certain equipment, or initiate load shedding under special conditions, by tripping load breakers on an ESF bus.

In response to an SI signal, an SI sequence is initiated and the sequencer controls the sequential loading of selected equipment on an ESF bus. During an SI sequence, an auto inhibit signal is generated to prevent certain ESF automatic starts and assure the DG is not overloaded. In response to a BO signal, a BO is initiated and the sequencer controls sequential loading and shedding of selected equipment on an ESF bus. A load shed signal may also be generated from the interaction of the BO and SI signals (or test signal). During a BO sequence, an auto inhibit signal is also generated.

AMSAC - The AMSAC (ATWS mitigation system actuation circuitry) provides an alternate means to automatically trip the main turbine and start the turbine driven and motor driven auxiliary feed pumps. The AMSAC actuation signals are generated by four SG level transmitters. The output of the level transmitters is supplied via current isolators to a three-out-of-four analog to digital programmable logic controller (PLC). When tripped, the PLC in turn sends actuation signals to the turbine driven auxiliary feed pump steam supply valve and motor driven auxiliary feed pump start circuitry as well as to the main turbine trip circuit. Figure 3.2-30 provides a logic diagram of the AMSAC system.

Fault Trees

Seventy one fault trees were developed for use in core melt quantification. These fault trees are generally used as subtrees in the various frontline and support system fault trees. The RPS fault trees that were developed are described below.

ESF1 and ESF3 - These models define the logic associated with the unavailability of the train A and train B safety injection actuation signal.

CS1 and CS3 - These models define the logic associated with the unavailability of the train A and train B containment spray actuation signal.

AI5B, AI6B, BOSEQ5, BOSEQ6, D1AUV, D1BUV, DGABO, DGBBO, LDSHED5, LDSHED6, SISEQ5 and SISEQ6 - These models define the logic associated with the unavailability of the Train A and Train B Sequences.

AMS, BAT, MANRT, RHR300A and RHR300B - These models define the logic associated with the unavailability of the special functions provided by these fault trees as described below.

AMS - Used as a node in the ATWS without main feedwater (AWS) event tree and depicts the failure of the AMSAC.

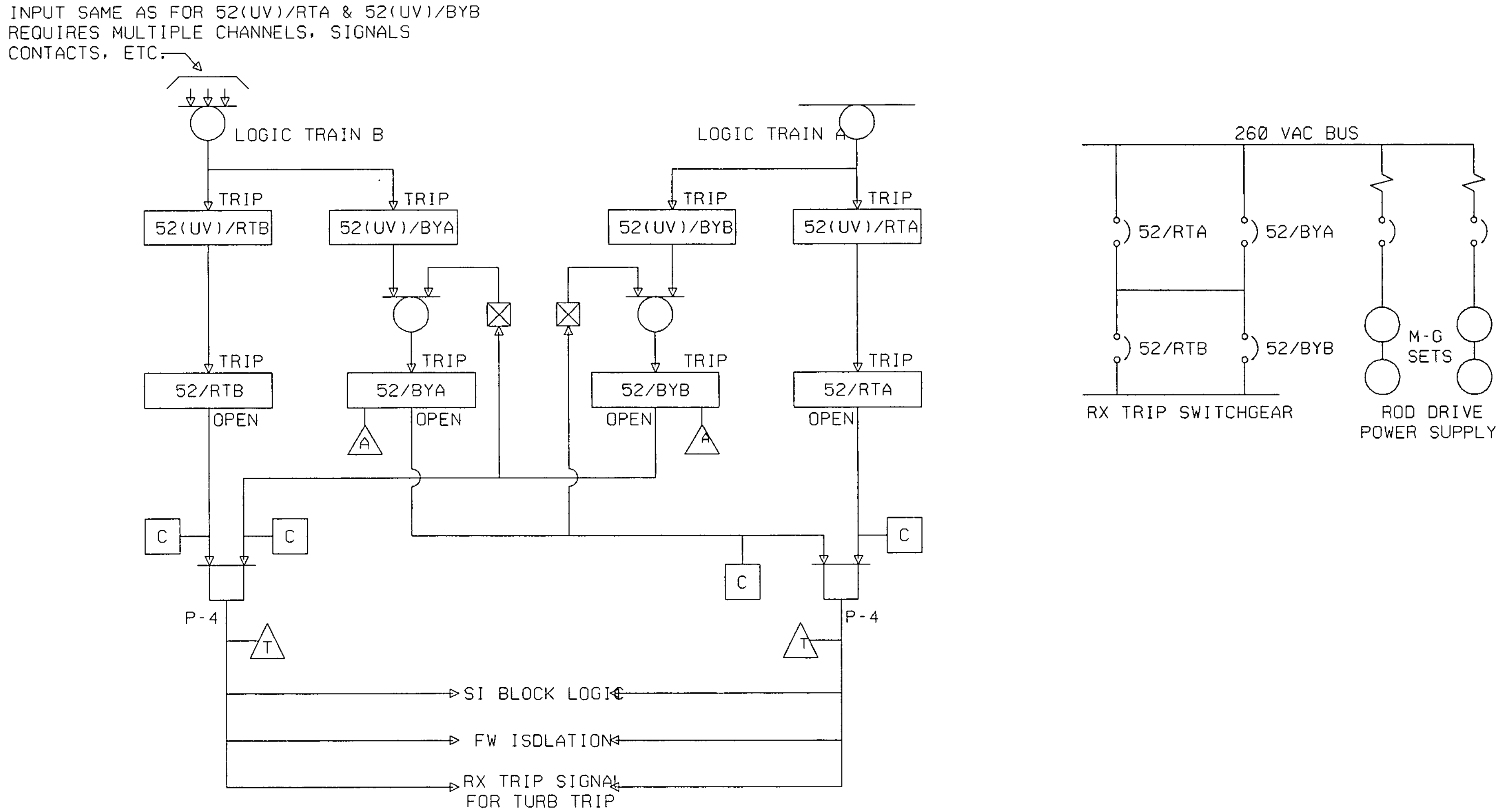
BAT - Used in the high pressure injection fault trees and depicts the failure of the interlock associated with the automatic switch over of the SI pumps suction from the boric acid tank (BAT) to the refueling water storage tank (RWST).

MANRT - Used in the AWS accident sequence and depicts the failure of the manual reactor trip function.

RHR300A/RHR300B - Used in the high pressure recirculation fault trees and depicts the failure of the interlock which prevents the opening of motor operated valves (MOVs) RHR-300A/RHR-300B unless MOVs SI-5A/SI-5B are closed, which prevents a diversion of low pressure recirculation flow to the RWST.

Fifty one component actuation modules were developed which define the unavailability of the specific component controls associated with the starting of pumps, fans and air compressors as well as the repositioning of valves given the presence of one of the actuation signals described previously.

FIGURE 3.2-29
REACTOR TRIP BREAKER LOGIC



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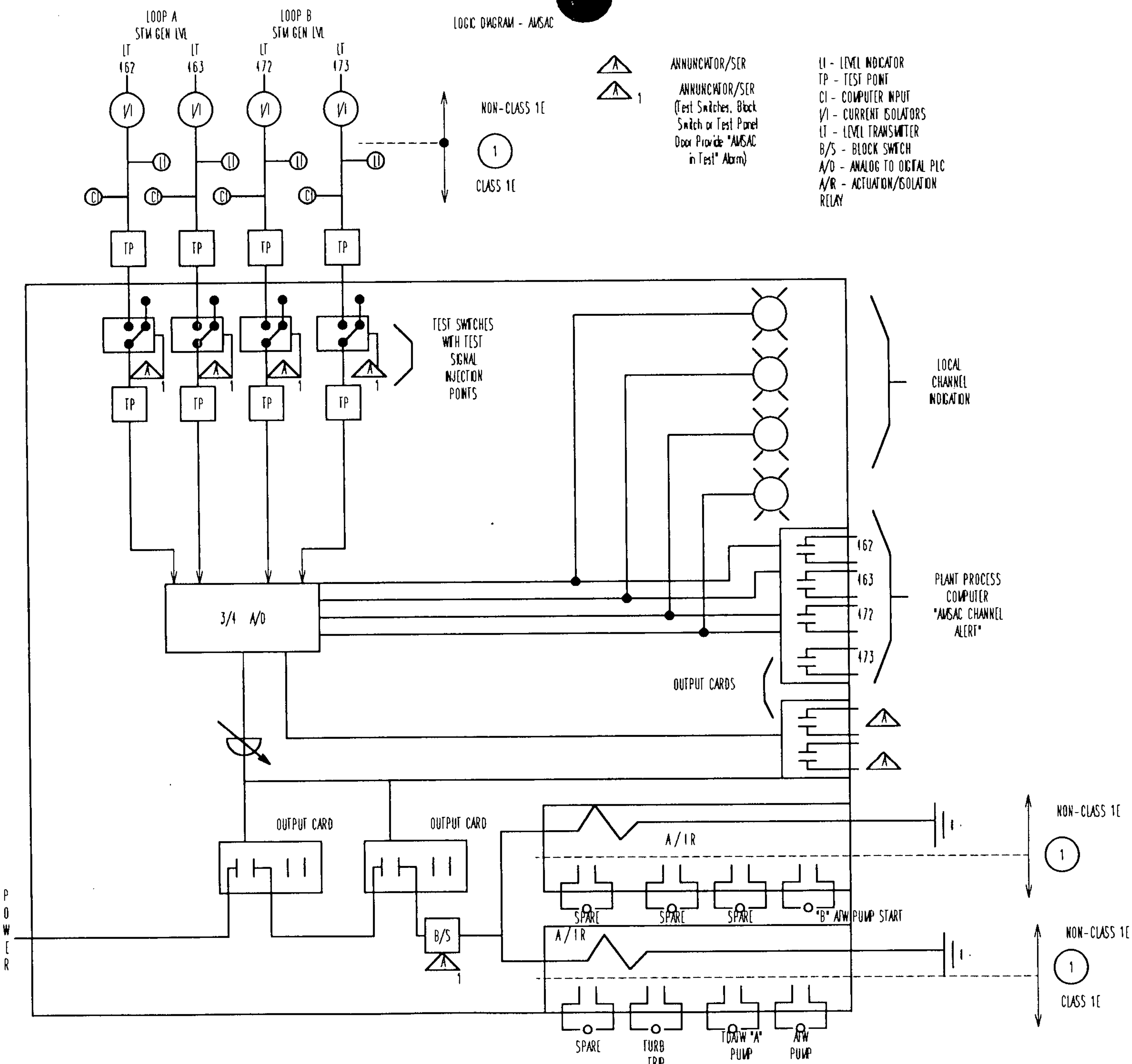


Figure 3.2-30

233

3.2.1.11 Service Water

Function

The service water (SW) system is designed to provide redundant cooling water supply to the diesel generators, air compressors, safety injection pumps, containment fan coil units, turbine and auxiliary building safeguard fan coil units, control room air conditioners, component cooling heat exchangers, charcoal filter deluge in the control room post accident recirculation, special zone ventilation, and shield building ventilation. The SWS system provides emergency supply of makeup to the component cooling system, spent fuel pool and a backup source of water to the auxiliary feedwater system. The SW System also provides non-redundant cooling water supplies for balance of plant equipment.

Description

The SW system flow diagram is shown in Figures 3.2-31, 3.2-32 and 3.2-33.

Service water is normally withdrawn from the circulating water system, which takes water from Lake Michigan through a deep-water, multiple inlet, submerged conduit. The water passes through the submerged intake to a forebay in the screenhouse structure, through four traveling water screens and to a screen well from which service water pumps take suction.

Provision for intaking service water are included in the sizing of the circulating water intake, which is the normal source of SW. In addition, two alternate sources of SW are provided. One alternate source consists of two auxiliary intakes on the circulating water intake pipe downstream of the main intakes. Each of these intakes is capable of providing the required amount of service water. The design of the auxiliary intakes are such that they are not damaged by frazzle ice. An alternate source of SW is provided by an interconnecting pipe between the circulating water discharge structure and the screenhouse forebay. This interconnecting provides a redundant source of SW in the extremely unlikely event the main intake line becomes blocked. The redundant path is always available since the valve CW-500 in the interconnecting line is locked open.

The traveling water screen backwash is supplied by two independent branch lines from each service water header. Each branch line supplies two screens through a normally open manual isolation valve.

The SW system is a plant support system and as such is in continuous operation for all plant operating modes. The load on the SW system is determined by plant load and environmental conditions. Normally three pumps are operating with the fourth in standby. The pump selected for standby operation is rotated on a bi-weekly basis.

Water is pumped from the pump bay by four centrifugal SW pumps through individual check valves to a rotating strainer and manual butterfly isolation valve. Two SW pumps (A1 and A2) discharge to SW header A and two SW pumps (B1 and B2) discharge to SW header B. The headers are connected by two normally open, air operated butterfly valves, SW-3A and SW-3B. These valves separate the headers when necessary. Each SW header supplies water to an auxiliary building SW header through motor operated butterfly isolation valves SW-10A and SW-10B respectively. These two auxiliary building headers can be connected through two normally closed manual butterfly valves, SW-11A and SW-11B. Various redundant safeguard equipment and coolers are supplied with SW from each of these headers. Only one of the two valves (SW-4A, SW4B that supply the turbine building header is open at any given time with SW-4A normally selected. When a SW pump is removed from service, the SW header with two operable SW pumps is selected to supply the turbine building header by positioning SW-4A and SW-4B appropriately.

The SW supply to the emergency diesel generators is provided by way of branch lines off of the respective train A and train B SW headers. These connections are located upstream of MOVs SW-10A and SW-10B. SW flow to the diesel generators is provided when normally closed AOVs SW-301A and SW-301B open when the associated unit starts.

Fault Trees

Based on a review of the frontline and support systems used for all the selected initiating events, ten fault trees were developed. The basis for each of these is described below. The mission time for all SW system fault trees is 24 hours.

Fault Tree SWA - This fault tree was developed to represent the SW support requirements for all train A engineered safety feature (ESF) equipment except the diesel generator. This fault tree is used for all events except for the loss of service water system, loss of offsite power and station blackout events. The success criterion is 1 of 2 pumps providing flow to train A SW header.

Fault Tree SWB - This fault tree was developed to represent the SW support requirements for all train B ESF equipment except the diesel generator. This fault tree is used for all events except loss of service water system, loss of offsite power and station blackout events. The success criterion is 1 of 2 pumps providing flow to train B SW header.

Fault Tree SWAP - This fault tree was developed to represent the SW support requirements for all train A ESF equipment except the diesel generator. This fault tree is used for the loss of offsite power event. The success criterion is 1 of 2 pumps providing flow to train A SW header.

Fault Tree SWBP - This fault tree was developed to represent the SW support requirements for all train B ESF equipment except the diesel generator. This fault tree is used for the loss of offsite power event. The success criterion is 1 of 2 pumps providing flow to train B SW header.

Fault Tree SWAG - This fault tree was developed to represent the SW support of diesel generator A for all events except the loss of service water system and the station blackout events. The success criterion is 1 of 2 pumps providing flow to diesel generator A.

Fault Tree SWBG - This fault tree was developed to represent the SW support requirements of diesel generator B for all events except the loss of service water system and the station blackout events. The success criterion is 1 of 2 pumps providing flow to diesel generator B.

Fault Tree SW - This fault tree was developed for use in the loss of component cooling system event. The success criterion is 1 of 2 SW trains providing flow to train A or train B SW headers.

Fault Tree SWT - This fault tree was developed to represent the SW support requirements of main feedwater and instrument air systems. The success criterion is 1 of 2 SW trains providing flow to the turbine building SW header.

Fault Tree SWAD - This fault tree was developed for use in the loss of 125V DC bus event. The success criterion is 1 of 2 pumps providing flow to train A SW header.

Fault Tree SWTD - This fault tree was developed for use in the loss of 125V DC bus event. The success criterion is 1 of 2 SW trains providing flow to the turbine building SW header.

FIGURE 3.2-31

SERVICE WATER

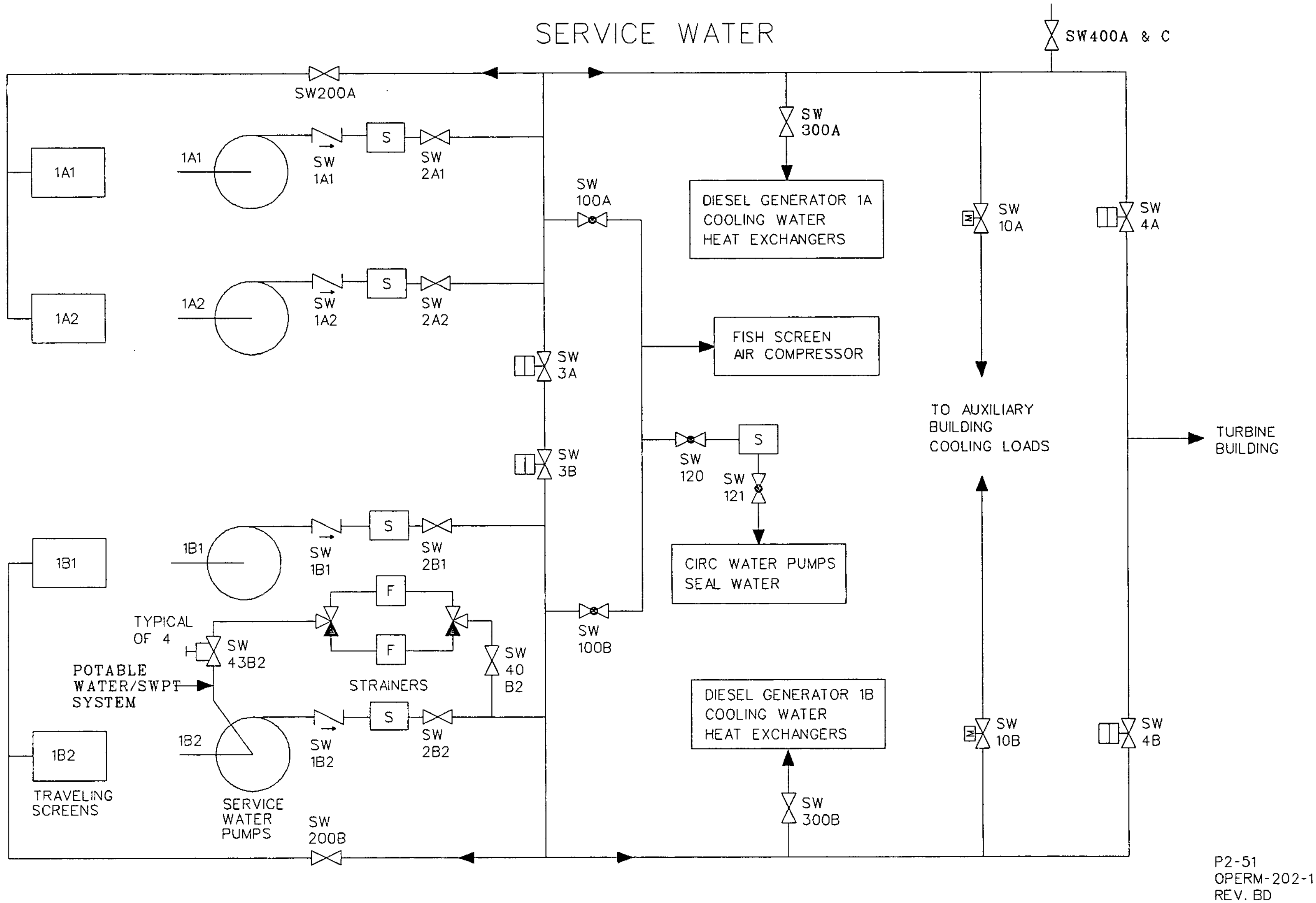
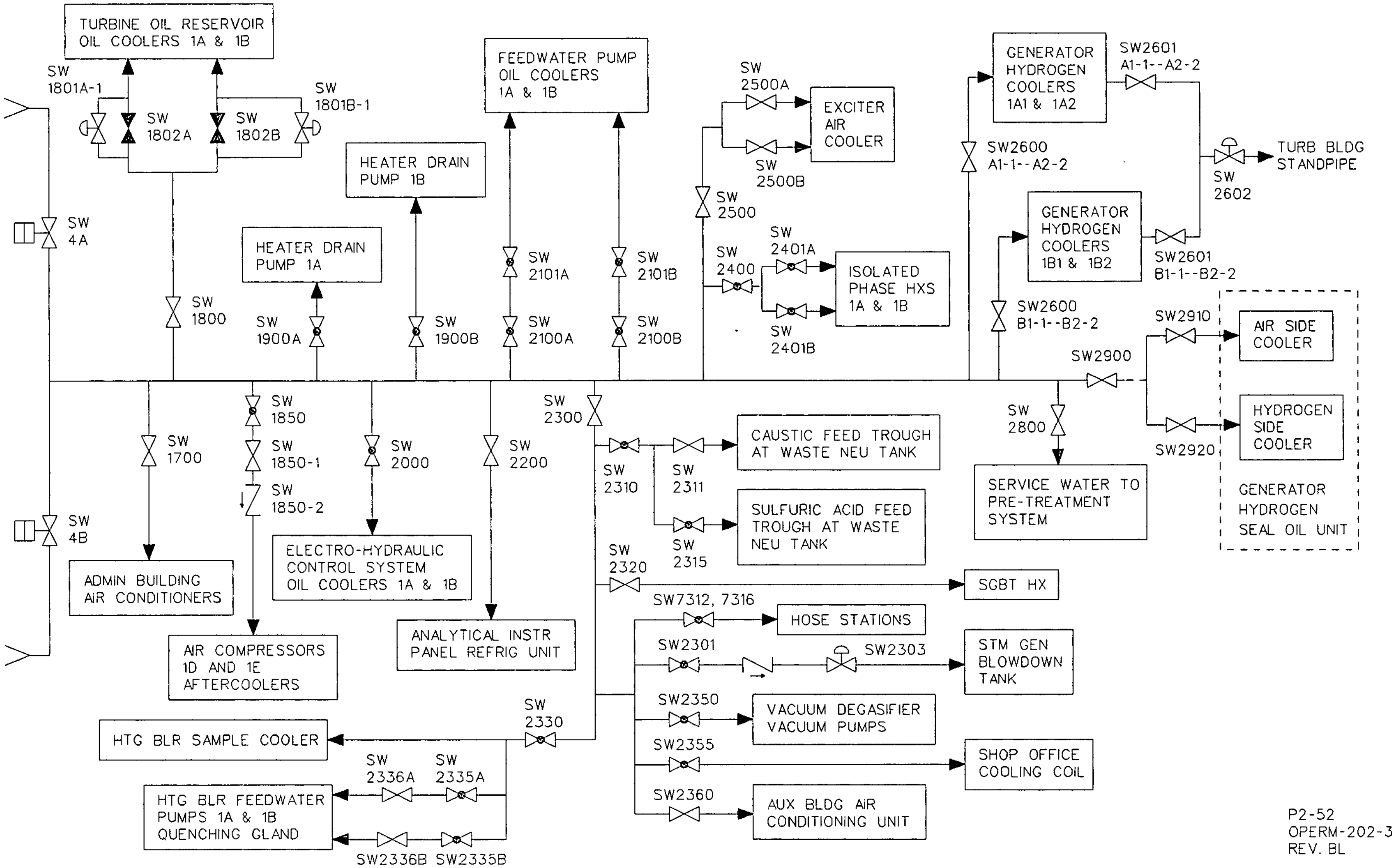


FIGURE 3.2-32

SERVICE WATER SYSTEM



P2-52
OPERM-202-3
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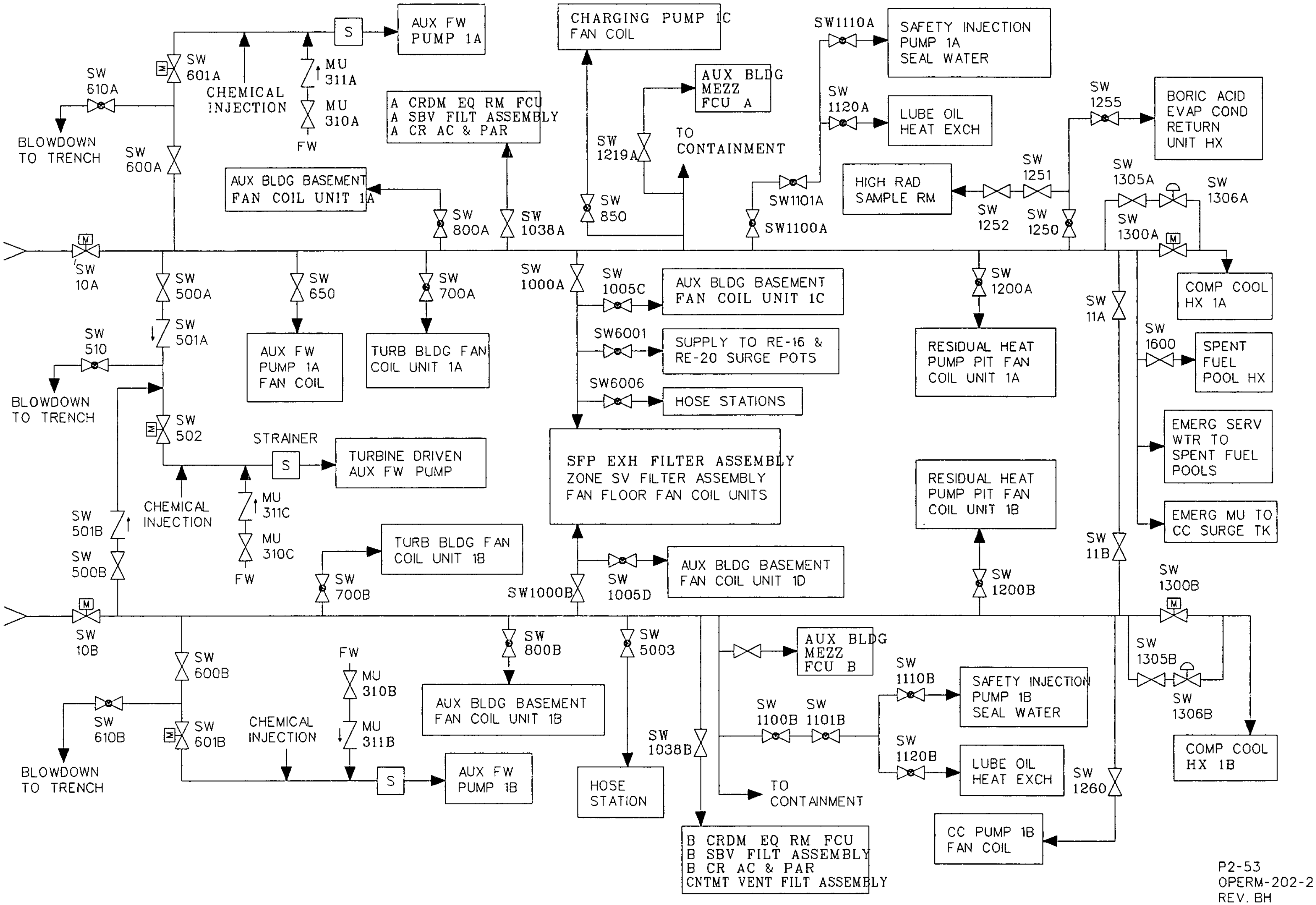
FIGURE 3.2-33

SERVICE WATER SYSTEM



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3.2.1.12 Miscellaneous Systems

This section documents the calculation of those event tree nodes that are not defined in a separate system notebook in the PRA study. A cross reference between those event tree nodes and the event or fault trees that they are used in is shown in Table 3.2-2.

The description and use of these miscellaneous event tree/fault tree nodes in the Kewaunee IPE can be found in the text that follows.

Each of the following subsections in this section contains:

- an identification of the accident sequences and or fault trees to which the node applies,
- a description of the node and assumptions that may apply
- a description of the fault tree and the success criteria

TABLE 3.2-2

MISCELLANEOUS NODE USAGE TABLE

SECTION	NODE	EVENT TREE	FAULT TREE	OPERATOR ACTION
3.2.1.12.1	ABBC		HR0, HR1, ICS, HR2 CSR, CSD	No
3.2.1.12.2	CHB	SBO		Yes
3.2.1.12.3	CHG	CCS, TRA, TRS, LSP		No
3.2.1.12.4	CHS	SWS		No
3.2.1.12.5	EC3	SGR		Yes
3.2.1.12.6	EC4	SGR		Yes
3.2.1.12.7	ES1	SLO		Yes
3.2.1.12.8	IS1	SLB		No
3.2.1.12.9	ISO	SGR		Yes
3.2.1.12.10	LTS	AWS		Yes
3.2.1.12.11	MRT	AWS		Yes
3.2.1.12.12	OB1	SLO		Yes
3.2.1.12.13	OB2	TRA, TRS		Yes
3.2.1.12.14	OB3	TDC		Yes
3.2.1.12.15	OB4	SLB		Yes
3.2.1.12.16	OB5	LSP		Yes
3.2.1.12.17	OCD	SBO		Yes
3.2.1.12.18	OP1	MLO		Yes

TABLE 3.2-2

MISCELLANEOUS NODE USAGE TABLE (Continued)

SECTION	NODE	EVENT TREE	FAULT TREE	OPERATOR ACTION
3.2.1.12.19	OP2	SLO		Yes
3.2.1.12.20	ORI	SBO		Yes
	ORT	AWS		Yes
3.2.1.12.22	OS1	SGR		Yes
3.2.1.12.23	OS2	SGR		Yes
3.2.1.12.24	OSD	SGR		Yes
3.2.1.12.25	OSP	LSP, TRA, TRS		Yes
3.2.1.12.26	OSR	ISL		Yes
3.2.1.12.27	OIP	ISL		Yes
3.2.1.12.28	PPR	AWS		Yes
3.2.1.12.29	RHR	SLO, SGR	ES1, EC3, EC4	No
3.2.1.12.30	RVC	ISL		No
3.2.1.12.31	SSV	SGR		No
3.2.1.12.32	IAS			No
3.2.1.12.33	IASP	LSP	OB5	No
3.2.1.12.34	IASPT	LSP	SWAP, SWBP	No
3.2.1.12.35	IAST		SWA, SWB, SWT, SWIE	No
3.2.1.12.36	IASTA		SWAG	No
3.2.1.12.37	IASTB		SWBG	No

TABLE 3.2-2

MISCELLANEOUS NODE USAGE TABLE (Continued)

SECTION	NODE	EVENT TREE	FAULT TREE	OPERATOR ACTION
3.2.1.12.38	IASD	TDC	OB3, OM4	No

3.2.1.12.1 ABBC - Failure of Auxiliary Building Basement Cooling

Function

ABBC is used as a subtree in HR0, HR1, HR2, CSR, CSD and ICS fault trees and represents area cooling support for the safety injection (SIS) and the containment spray (ICS) systems.

Description

The 4 auxiliary building basement (ABB) fan coil units (FCU) are modeled as a subtree in both the high pressure SI recirculation and the ICS fault trees. The A and B ABB FCUs are capable of removing 400,000 BTU/hr of heat at 11,000 scfm air flow. The C and D ABB FCUs are capable of removing 240,000 BTU/hr of heat at 8,500 scfm air flow. All four fan coil units are cooled by service water. For ABBC to be successful at least 2 ABB FCUs must operate in any of five combinations except C and D together for a defined mission time of 24 hours. Refer to Figures 3.2-22 and 3.2-23 for simplified flow diagrams.

Fault Tree

The analysis for node ABBC consists of construction of a fault tree that predicts the failure of this node with respect to the requirements and assumptions listed above. This fault tree ABBC is considered to fail if none of the acceptable combinations of the ABB FCUs are available.

3.2.1.12.2 CHB - No Flow From 1 of 2 Charging Pumps

Function

The node applies to the station blackout (SBO) accident sequence and represents the loss of charging flow for reactor coolant pump (RXCP) seal injection.

Description

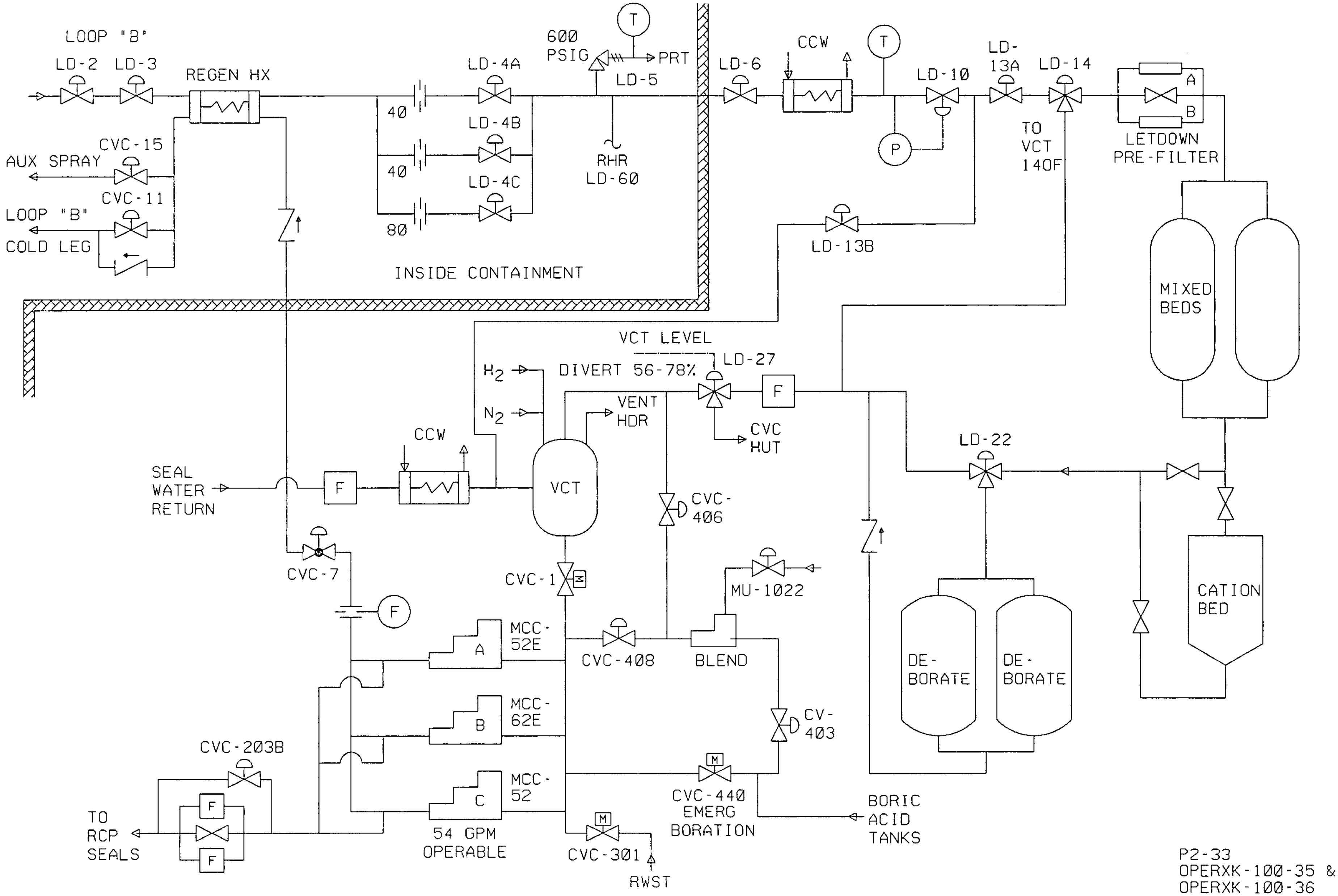
With loss of component cooling water (CCW) to the RXCP thermal barrier, the operator must maintain a minimum amount of charging flow to supply RXCP seal injection. One charging pump provides adequate RXCP seal cooling and thereby prevent a small break loss of coolant accident (LOCA) due to seal degradation following loss of all seal cooling. Continued post-trip operation of the charging pump plus operator training ensure that seal injection is maintained with little or no interruption following reactor trip. In the case of an SBO, the operators are instructed to strip bus 52 of all loads, start the technical support center diesel generator, power bus 52 through bus 46, and start one of the two charging pumps (A or C) that are powered from bus 52. Based on the expected RXCP seal response to the loss of all cooling described in Reference 24, a normal seal flow requirement of 3 to 5 gpm per pump (less than 10 gpm total) is expected if seal injection is restored by 10 minutes, i.e., prior to the transient heatup phase.

Even if seal injection is delayed until about 30 minutes, the seal leakage rate is expected to be less than 21 gpm per pump or 42 gpm total. This is still within the capacity of one of the 60 gpm positive-displacement charging pumps. Based on the above description, success of this node is 1 of 2 charging pumps supplying the minimum flow needed for seal injection for a defined mission time of 24 hours. Figures 3.2-34 and 3.2-35 provide flow diagrams for the chemical and volume control system.

Fault Tree

The analysis for node CHB consists of a fault tree that predicts the failure of this node with respect to the requirements and assumptions listed above. This fault tree CHB is considered to fail if the charging pumps fail to deliver flow to cool the RXCP seals.

FIGURE 3.2-34
SIMPLIFIED CHEMICAL & VOLUME CONTROL SYSTEM

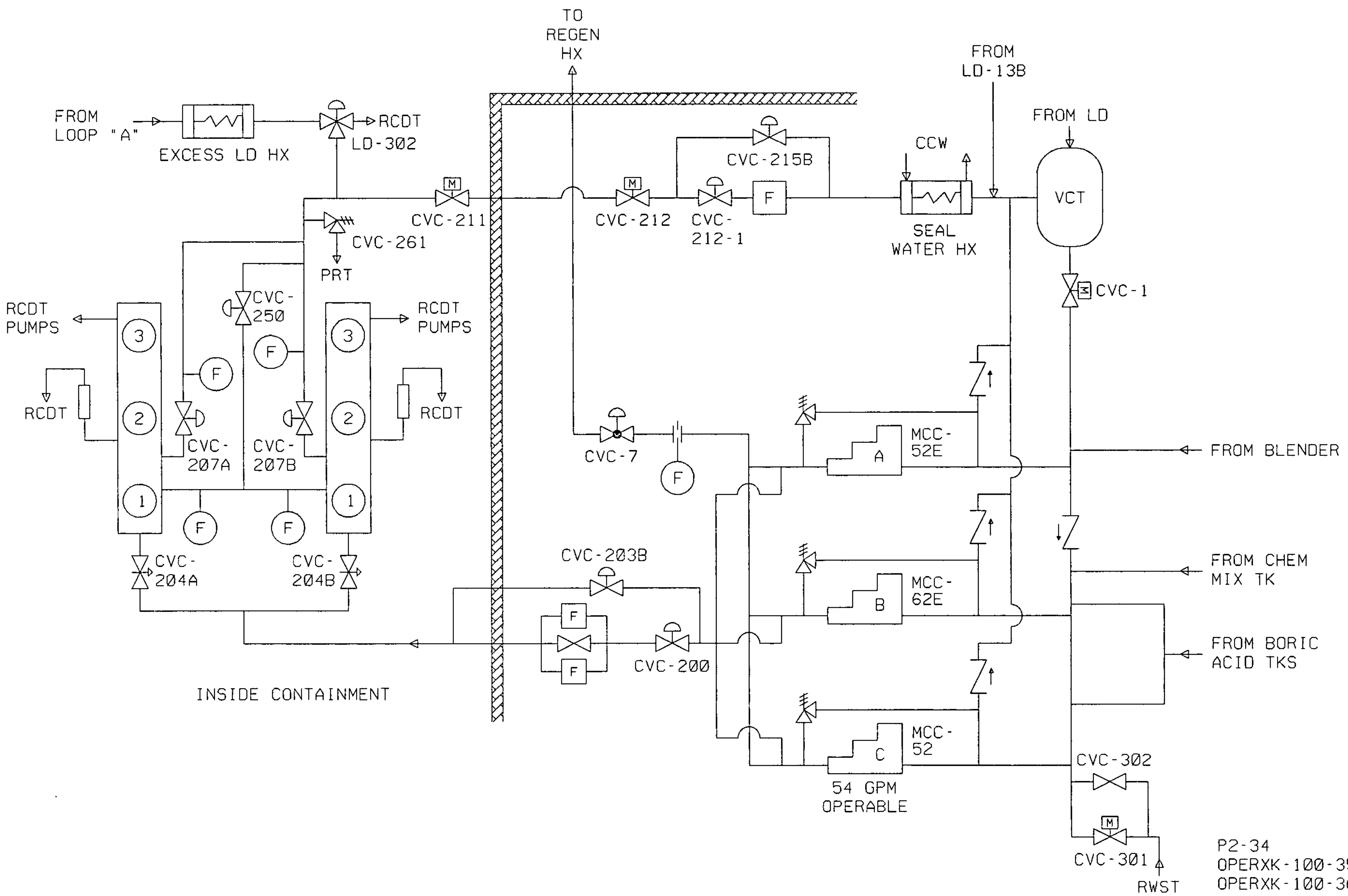


P2-33
OPERXK-100-35 &
OPERXK-100-36
REV. R & AA
4-16-87

246

C:\DGN\FG\M3-34.DGN Sep. 28, 1992 15:04:08

FIGURE 3.2-35
CHARGING



P2-34
OPERXK-100-35 &
OPERXK-100-36
REV. R & AA
7-10-87

247

C:\DGN\FG\H3-35.DGN Sep. 28, 1992 15:05:36

3.2.1.12.3 CHG - No Flow From 1 of 3 Charging Pumps

Function

This node applies to accident sequences loss of component cooling (CCS), transients with main feedwater (TRA), transients without main feedwater (TRS) and loss of offsite power (LSP) and represents the loss of charging flow for RXCP seal injection.

Description

With loss of CCW to the RXCP thermal barrier, the operator must maintain a minimum amount of charging flow to supply RXCP seal injection. One charging pump provides adequate RXCP seal cooling and thereby prevents a small LOCA due to seal degradation following loss of all seal cooling. Continued post-trip operation of the charging pump plus operator training ensure that seal injection is maintained with little or no interruption following reactor trip. In addition, step 4 of Emergency Operating Procedure (EOP) ES-0.1, Reactor Trip Response, explicitly directs the operator to verify or establish charging flow. It is highly probable that step 4 of EOP ES-0.1 would be reached within 10 minutes following reactor trip since the EOP E-0 to EOP ES-0.1 transition occurs very quickly via step 4 of EOP E-0. Based on the expected RXCP seal response to the loss of all cooling described in Reference 24, a normal seal flow requirement of 3 to 5 gpm per pump (less than 10 gpm total) is expected if seal injection is restored by 10 minutes, i.e., prior to the transient heatup phase. Even if seal injection is delayed until about 30 minutes, the seal leakage rate is expected to be less than 21 gpm per pump or 42 gpm total. This is still within the capacity of one of the 60 gpm positive-displacement charging pumps. Based on the above description, success for the CHG top event requires either; continued operation of 1 of 3 charging pumps for seal injection for a defined mission time of 24 hours, or operator action (based on training or the EOPs) to start a charging pump for seal injection within 30 minutes following reactor trip.

Fault Tree

The analysis for node CHG consists of construction of a fault tree that predicts the failure of this node with respect to the requirements and assumptions listed above. This fault tree CHG is considered to fail if the charging pumps fail to deliver enough flow to cool the RXCP seals.

3.2.1.12.4 CHS - No Flow From 1 of 3 Charging Pumps

This node applies to the loss of service water (SWS) accident sequence and represents the loss of charging flow for RXCP seal injection.

Description

With loss of CCW to the RXCP thermal barrier, the operator must maintain a minimum amount of charging flow to supply RXCP seal injection. One charging pump will provide adequate RXCP seal cooling and thereby prevent a small LOCA due to seal degradation following loss of all seal cooling. Continued post-trip operation of the charging pump plus operator training ensure that seal injection is maintained with little or no interruption following reactor trip. In addition, step 4 of EOP ES-0.1, Reactor Trip Response, explicitly directs the operator to verify or establish charging flow. It is highly probable that step 4 of EOP ES-0.1 would be reached within 10 minutes following reactor trip since the EOP E-0 to EOP ES-0.1 transition occurs very quickly via step 4 of EOP E-0. Based on the expected RXCP seal response to the loss of all cooling described in Reference 24 a normal seal flow requirement of 3 to 5 gpm per pump (less than 10 gpm total) is expected if seal injection can be restored by 10 minutes, i.e., prior to the transient heatup phase. Even if seal injection is delayed until about 30 minutes, the seal leakage rate is expected to be less than 21 gpm per pump or 42 gpm total. This is still within the capacity of one of the 60 gpm positive-displacement charging pumps. Based on the above description, success for the CHS top event requires either; continued operation of 1 of 3 charging pumps for seal injection for a defined mission time of 24 hours, or operator action (based on training or the EOPs) to start a charging pump for seal injection within 30 minutes following reactor trip.

Fault Tree

The analysis for node CHS consists of construction of a fault tree that predicts the failure of this node with respect to the requirements and assumptions listed above. This fault tree CHG is considered to fail if the charging pumps fail to deliver enough flow to cool the RXCP seals.

3.2.1.12.5 EC3 - Failure to Cooldown and Depressurize RCS

Function

This node applies to the Steam Generator Tube Rupture (SGR) accident sequence and represents the operator actions and system hardware failure that result in the failure to cooldown and depressurize the reactor coolant system.

Description

The ECA-3.1 and possibly ECA-3.2 recovery actions need to be followed if the ruptured SG can not be isolated from the intact SG used for cooldown (ISO fails).

In ECA-3.1, the operator initiates a cooldown at a maximum rate of 100°F/hr using the intact SG. Once residual heat removal (RHR) entry conditions are established reactor coolant system (RCS) pressure less than 425 psig and coldest RCS wide range temperature less than 380°F), the RHR system can be placed in service to continue the cooldown to cold shutdown (200°F). As the cooldown progresses, the high pressure safety injection (SI) pumps are sequentially stopped (based on specified subcooling and RCS inventory criteria) until the charging pumps are able to supply the required makeup. The RCS is also depressurized either using pressurizer spray or a PORV to minimize the break flow to the ruptured SG and the environment. If RWST level has decreased to below 52% or narrow range level in the ruptured SG has increased to above 92%, step 13 of ECA-3.1, SGTR With Loss of Reactor Coolant - Subcooled Recovery Desired, instructs the operators to determine if a transition to ECA-3.2, SGTR With Loss of Reactor Coolant - Saturated Recovery Desired, is appropriate.

In ECA-3.2, the subcooling and inventory criteria are relaxed to allowed a more expedited recovery. The ultimate objective of the ECA-3.1/3.2 recovery is to depressurize the RCS and ruptured SG to near atmospheric pressure and thereby terminate the leak. Although the actions appear complex, the operator has roughly 6 to 10 hours for them to be successful for a design basis SGR provided high pressure SI is available. This is the approximate RWST depletion time for a break with an average injection flow requirement ranging from 300 to 600 gpm.

For the success for EC3, successful RHR system operation is required. A requirement for operation of at least 2 of 3 charging pumps, 1 of 2 PORVs or 1 of 1 auxiliary spray valves are also included since this would allow the SI pumps to be stopped at reasonable subcooling values and would allow makeup control after the SI pumps are secured. The defined mission time is 24 hours.

It is assumed that the failure of EC3 always results in core melt.

Fault Tree

The analysis for node EC3 consists of construction of a fault tree that predicts the failure of this node with respect to the requirements and assumptions listed above. This fault tree EC3 is considered to fail if the operator fails to follow the EOPs, the RHR system fails, neither the PORVs or pressurizer spray is available to depressurize the RCS, or none of the steam dump valves is operable.

3.2.1.12.6 EC4 - Failure to Cooldown and Depressurize RCS

Function

This node applies to the SGR accident sequence and represents the operator actions and system hardware failure that result in the failure to cooldown and depressurize the RCS.

Description

In addition to the scenario described above for EC3, ECA-3.1 and possibly ECA-3.2 recovery actions need to be followed if SG overflow occurs and a secondary side relief valve on the ruptured SG sticks open (OS1 and SSV fail).

EC4 represents the same operator actions as EC3. Since it follows the failure of OS1, however, it has a higher failure probability due to operator dependence.

It is assumed that failure of EC4 always results in core melt.

Fault Tree

The analysis for node EC4 consists of construction of a fault tree that predicts the failure of this node with respect to the requirements and assumptions listed above. This fault tree EC4 is considered to fail if the operator fails to follow the EOPs, the RHR system fails, neither the PORVs or pressurizer spray is available to depressurize the Reactor Coolant System (RCS), or none of the steam dump valves is operable.

3.2.1.12.7 ES1 - Failure to Cooldown and Depressurize RCS

Function

This node applies to the small break LOCA (SLO) accident sequence and represents the operator actions and system hardware failures that result in the failure to cooldown and depressurize the reactor coolant system.

Description

In EOP ES-1.2, Post LOCA Cooldown and Depressurization, the operator initiates a cooldown at a maximum rate of 100°F/hr using the available intact SGs. Once RHR system entry conditions are established (RCS pressure less than 425 psig, coldest RCS wide range temperature less than 380°F), the RHR system is placed in service to continue the cooldown to cold shutdown (200°F). As the cooldown progresses, the high pressure SI pumps are sequentially stopped (based on specified subcooling and RCS inventory criteria) with the charging pumps able to supply the required makeup. The RCS would also be depressurized (using pressurizer spray or a PORV) to increase inventory and to minimize the break flow. Using EOP ES-1.2, it may be possible for very small break cases to depressurize the RCS to near atmospheric pressure and thereby terminate or substantially reduce the break flow. By doing so, the charging flow can be reduced and switchover to high pressure recirculation can be avoided.

Although the ES-1.2 actions appear complex, the operator would have a long period of time to perform these actions. With the ES-1.2 actions, it is likely that for break sizes 0.7 inch diameter or smaller (it is assumed that 50% of the breaks will be in this range), the operator is able to avoid switchover for at least the 24 hour time frame assumed for the event tree and fault tree modeling. Success also requires steam dump from at least one of the SGs (i.e., steam dump to condenser, if available, or operation of the atmospheric steam dump valve). The active SGs used for the cooldown also need a supply of auxiliary feedwater (AF0 success) until the RHR system can be aligned for service. Another function to ensure ES1 success is a means for RCS depressurization by either pressurizer spray or operation of one PORV. Normal spray requires operation of an RXCP. In order to achieve cold shutdown conditions, it is assumed that operation of at least one train of RHR is required. A requirement for operation of at least 2 of 3 charging pumps, 1 of 2 PORVs or 1 of 1 auxiliary spray valves are also included since this allows the SI pumps to be stopped at reasonable subcooling values and allows makeup control after the SI pumps are secured. The defined mission time is 24 hours.

Fault Tree

The analysis for node ES1 consists of construction of a fault tree that predicts the failure of this node with respect to the requirements and assumptions listed above. This fault tree ES1 is considered to fail if the operator fails to follow the EOPs, the RHR system fails, neither the PORVs or pressurizer spray is available to depressurize the RCS, or none of the steam dump valves is operable.

3.2.1.12.8 IS1 - Failure of Isolation After SLB Event

Function

This node applies to the steam line break (SLB) accident sequence and represents the failure of main steam and main feedwater isolation functions.

Description

Main steam and feedwater isolation (IS1) are necessary to stop the cooldown. To allow for an arbitrary break location, success of IS1 requires closure of both of the two MSIVs and isolation of both feedwater lines. Operator action is not required for IS1 success. The defined mission time is 24 hours.

Fault Tree

The analysis for node IS1 consists of construction of a fault tree that predicts the failure of this node with respect to the requirements and assumptions listed above. This fault tree IS1 is considered to fail if either MSIV fails to close or feedwater isolation is not achieved on either train.

3.2.1.12.9 ISO - Failure to Isolate 1 of 2 Steam Generators

Function

This node applies to the SGR accident sequence and represents the operator actions and hardware failures that result in the failure to isolate at least one steam generator.

Description

In the EOP E-3, Steam Generator Tube Rupture recovery, the ruptured SG is isolated from the intact SG by closure of an MSIV. Other paths to and from the ruptured SG also require isolation (e.g., blowdown, steam supply to the turbine-driven auxiliary feedwater (AFW) pump, etc.). Isolation of these paths, however, is not as crucial to the recovery as main steam isolation. It is preferable to close the MSIV for the ruptured SG since this gives the operator the option of using steam dump to condenser, if available, for the subsequent cooldown using the intact SGs. Should the MSIV for the ruptured SG fail to close, the MSIV for the intact SG is closed and the corresponding SG PORV used for the cooldown. Since the initial cooldown is limited (i.e., to about 500°F), only one SG is required for the cooldown. Therefore, success for the ISO function is determined by the ability to close at least one MSIV on any SG. For a design basis SGR, it is assumed that the operator must identify the ruptured SG and perform the ISO isolation function by 15 minutes for ISO to be successful.

If this node fails, the recovery actions in ECA-3.1 or ECA-3.2 (EC3) need to be addressed.

Fault Tree

The analysis for node ISO consists of construction of a fault tree that predicts the failure of this node with respect to the requirements and assumptions listed above. This fault tree ISO is considered to fail if the operator fails to follow the EOPs or both MSIVs fail to close.

3.2.1.12.10 LTS - Failure to Maintain Long Term Shutdown

Function

This node applies to the ATWS Without Main Feedwater (AWS) accident sequence and represents the operator actions and system hardware failures that result in the failure to maintain long term shutdown.

Description

If automatic trip, manual trip, or manual/local opening of the breakers for the rod drive motor-generator sets do not shut down the reactor early in the transient and the peak RCS pressure does not exceed the stress criterion within the first few minutes of the transient, then alternate means to achieve subcriticality and maintain the shutdown condition are available. Emergency Operating Procedure FR-S.1, Response to Nuclear Power Generation/ATWS, instructs the operator to begin manual rod insertion. Boration of the RCS is also initiated with the high pressure charging pumps in emergency boration via the boric acid tank.

If there is no mechanical failure associated with the control rods, the operators are able to manually insert the control rod banks to achieve long term shutdown. Successful manual rod insertion is adequate to ensure long term shutdown at the hot zero power condition. Analysis shows that reactor shutdown can be achieved within 20 minutes by manual insertion of the control rods.

If the control rods can not be inserted, emergency boration is used to achieve long term shutdown. The limiting boration time is estimated by assuming the RCS boron concentration must be increased from full power beginning-of-life equilibrium xenon conditions to one percent, xenon free shutdown conditions with two boric acid transfer pumps taking suction from the boric acid tank and feeding directly to the suction of two charging pumps, the one percent shutdown boron concentration is achieved within 15 minutes. The success of this node requires 2 of 3 charging pumps, 1 of 2 boric acid transfer pumps and 1 of 1 emergency boration valves to operate for a defined mission time of 24 hours.

If long term shutdown fails, it is assumed core damage occurs.

Fault Tree

The analysis for node LTS consists of construction of a fault tree that predicts the failure of this node with respect to the requirements and assumptions listed above. This fault tree LTS is considered to fail if the operator fails to follow the EOPs or the charging system fails.

3.2.1.12.11 MRT - Manual Reactor Trip Failure

Function

This node applies to the AWS accident sequence and represents the operator action and reactor protection system (RPS) hardware failure that result in the failure of a manual reactor trip.

Description

Top event MRT models the manual trip of the reactor by the operators. If the failure of the automatic reactor trip function is due to failure of the RPS logic, the operator may be successful at manually tripping the reactor using 1 of 2 pushbuttons. Success for this event tree node means that an ATWS has not occurred.

Fault Tree

The analysis for node MRT consists of construction of a fault tree that predicts the failure of this node with respect to the requirements and assumptions listed above. This fault tree MRT is considered to fail if the operator fails to follow the EOPs, or neither of the manual reactor trip pushbuttons causes a reactor trip.

3.2.1.12.12 OB1 - Failure to Estahlish Bleed and Feed

Function

The node applies to the SLO accident sequence and represents the operator actions and system hardware failures that result in the failure to establish bleed and feed.

Description

If secondary cooling via AFW or main feedwater (MFW) is unavailable, the operators are instructed to initiate primary system bleed and feed. Emergency Operating Procedure FR-H.1, Response to Loss of Secondary Cooling, instructs the operators to initiate bleed and feed if secondary cooling is lost and wide range SG level in either SG drops below 15% (RCS pressure and hot leg temperature increasing for adverse containment) or pressurizer pressure increases above 2335 psig. The operators use the SI pumps for injection and establish an RCS bleed path by opening at least one of two pressurizer PORVs. It is likely that bleed and feed cooling using FR-H.1 would be established by 30 minutes. SG secondary dryout is expected at approximately one hour.

Success of OB1 is 1 of 2 high pressure SI trains delivering flow to 1 of 2 RCS cold legs with 1 of 2 pressurizer PORVs open. Bleed and feed initiation prior to SG dryout with this success criterion is expected to result in effective decay heat removal. For simplicity, it is assumed that bleed and feed initiated by 30 minutes using one SI pump and one pressurizer PORV results in success. The success of this node requires 1 of 2 high pressure injection trains and 1 of 2 PORVs to operate for a defined mission time of 24 hours.

It is assumed that failure of this node results in early core melt due to loss of all secondary cooling.

Fault Tree

The analysis for node OB1 consists of construction of a fault tree that predicts the failure of this node with respect to the requirements and assumptions listed above. This fault tree OB1 is considered to fail if the operator fails to follow the EOPs or the PORVs fail.

3.2.1.12.13 OB2 - Failure to Establisb Bleed and Feed

Function

This node applies to the TRA and TRS accident sequences and represents the operator actions and system hardware failures that result in the failure to establish bleed and feed.

Description

If AFW and MFW fail, the operators are instructed to initiate primary system bleed and feed. Emergency operating procedure FR-H.1, Response to Loss of Secondary Cooling, instructs the operators to initiate bleed and feed if secondary cooling is lost and wide range steam generator level in either steam generator drops below 15% or pressurizer pressure increases above 2335 psig. The operators start at least one high pressure SI pump and establish an RCS bleed path by opening at least one of two pressurizer PORVs. It is likely that bleed and feed cooling, according to FR-H.1 instructions, would be established by 30 minutes. SG secondary dryout is expected at approximately one hour.

Success of this node is 1 of 2 high pressure SI trains delivering flow to 1 of 2 RCS cold legs with at least one pressurizer PORV open. Bleed and feed initiation prior to SG dryout with this success criterion is expected to result in effective decay heat removal. For simplicity, it is assumed that bleed and feed initiated by 30 minutes using one high pressure SI pump and one pressurizer PORV will result in success. The success of this node requires 1 of 2 high pressure injection trains and 1 of 2 PORVs to operate for a defined mission time of 24 hours.

It is assumed that failure of this node results in early core melt due to loss of all secondary cooling.

Fault Tree

The analysis for node OB2 consists of construction of a fault tree which predicts the failure of this node with respect to the requirements and assumptions listed above. This fault tree OB2 will be considered to fail if the operator fails to follow the EOPs, the PORVs fail, or high pressure safety injection fails.

3.2.1.12.14 OB3 - Failure to Establish Bleed and Feed

Function

This node applies to the Loss of 125V DC Bus (TDC) accident sequence and represents the operator actions and system hardware failures which would result in the failure to establish bleed and feed.

Description

If secondary cooling with AFW and the MFW is unavailable, the operators are instructed to initiate primary system bleed and feed. Emergency operating procedure FR-H.1, Response to Loss of Secondary Cooling, instructs the operators to initiate bleed and feed if secondary cooling is lost and wide range steam generator level in either steam generator drops below 15% or pressurizer pressure increases above 2335 psig. The operators start at least one high pressure SI pump and establish a RCS bleed path by opening the available pressurizer PORV. The A SI pump has to be started locally, while the B SI pump can be started manually. According to FR-H.1 instructions, it is likely that bleed and feed cooling would be established by 30 minutes. SG secondary dryout is expected at approximately one hour.

Success of OB3 is 1 of 2 high pressure SI trains delivering flow to 1 of 2 RCS cold legs with the available pressurizer PORV open. Bleed and feed initiation prior to SG dryout with this success criterion is expected to result in effective decay heat removal. For simplicity, it is assumed that bleed and feed initiated by 30 minutes using one high pressure SI pump and one pressurizer PORV results in success. The success of this node requires 1 of 2 high pressure injection trains and 1 of 2 PORVs to operate for a defined mission time of 24 hours.

It is assumed that failure of this node results in early core melt due to loss of all secondary cooling.

Fault Tree

The analysis for node OB3 consists of construction of a fault tree that predicts the failure of this node with respect to the requirements and assumptions listed above. This fault tree OB3 is considered to fail if the operator fails to follow the EOPs, the PORVs fail, or SI fails.

3.2.1.12.15 OB4 - Failure to Establish Bleed and Feed

Function

This node applies to the SLB accident sequence and represents the operator actions and system hardware failures which would result in the failure to establish bleed and feed.

Description

If AFW and MFW fail, the operators are instructed to initiate primary system bleed and feed. Emergency operating procedure FR-H.1, Response to Loss of Secondary Cooling, instructs the operators to initiate bleed and feed if secondary cooling is lost and wide range SG level in either SG drops below 15% or pressurizer pressure increases above 2335 psig. The operators use the SI pumps for injection and establish an RCS bleed path by opening at least one of two pressurizer PORVs. Bleed and feed initiated with EOP FR-H.1 is performed at a comparatively early time if all SGs loose some of their inventory prior to MSIV closure. Most likely, the loss of heat sink symptom is not reached early, and bleed and feed is not performed until after the RCS and intact SG heat up to no-load and additional inventory is boiled from the intact SG. Since the residual heat is not event dependent (precluding any significant nuclear heat due to a return to criticality) and since the RCS gains additional inventory and heat capacity due to addition of cold SI water, the intact SG dryout times for the secondary break transients exceed the one hour SG dryout time previously noted for the other transient events. Thus, it is appropriate to apply the same success criterion for the steamline break as the other transient cases.

Success of OB4 is 1 of 2 high pressure safety injection trains delivering flow from the RWST to 1 of 2 RCS cold legs with 1 of 2 pressurizer PORVs open. Bleed and feed initiation prior to intact SG dryout with this success criterion is expected to result in effective decay heat removal. For simplicity, it is assumed that bleed and feed initiated by 30 minutes results in successful recovery.

With a secondary heat sink available, the only SI required following a large secondary side break is the contents of a BAT. However, with no secondary heat sink available, safety injection from the RWST is required for a successful bleed and feed recovery. Top event OB4 includes the automatic transition of the suction of the SI pumps from the BAT to the RWST. The SI pumps are running since the emergency operating procedures direct the operator to leave them running if a secondary heat sink is unavailable. Thus, no manual action by the operator is required for top event OB4 to either transfer the SI pumps' suction to the RWST or to start the SI pumps. He is required to manually open at least one pressurizer PORV to provide an RCS bleed path. The success of this node requires 1 of 2 high pressure injection trains and 1 of 2 PORVs to operate for a defined mission time of 24 hours.

It is assumed that failure of this node results in early core melt due to loss of all secondary cooling.

Fault

The analysis for node OB4 consists of construction of a fault tree that predicts the failure of this node with respect to the requirements and assumptions listed above. This fault tree OB4 is considered to fail if the operator fails to follow the EOPs, the PORVs fail, or SI fails.

3.2.1.12.16 OB5 - Failure to Estahlisb Bleed and Feed

Function

This node applies to the LSP accident sequence and represents the operator actions and system hardware failures that result in the failure to establish bleed and feed.

Description

If AFW fails, the operators are instructed to initiate primary system bleed and feed. Emergency Operating Procedure FR-H.1, Response to Loss of Secondary Cooling, instructs the operators to initiate bleed and feed if secondary cooling is lost and wide range steam generator level in either SG drops below 15% or pressurizer pressure increases above 2335 psig. The operators actuate SI and establish a RCS bleed path by opening at least one of two pressurizer PORVs. The availability of OB2 is dependent on the availability of emergency onsite power supply to the vital buses. It is likely that bleed and feed cooling would be established according to FR-H.1 instructions by 30 minutes. SG secondary dryout would be expected at approximately one hour.

Success of OB5 is 1 of 2 high pressure SI trains delivering flow to 1 of 2 RCS cold legs with 1 of 2 pressurizer PORVs open for a defined mission time of 24 hours. Bleed and feed initiation prior to SG dryout with this success criterion is expected to result in effective decay heat removal. For simplicity, it is assumed that bleed and feed initiated by 30 minutes using one high pressure SI pump and one pressurizer PORV will result in success.

It is assumed that failure of this node results in early core melt due to loss of all secondary cooling.

Fault Tree

The analysis for node OB5 consists of construction of a fault tree that predicts the failure of this node with respect to the requirements and assumptions listed above. This fault tree OB5 is considered to fail if the operator fails to follow the EOPs, the PORVs fail, or SI fails.

3.2.1.12.17 OCD - Failure to Cooldown RCS

Function

This node applies to the station blackout (SBO) accident sequence and represents the operator action and system hardware failures that result in the failure to cool down the RCS.

Description

The emergency procedures instruct the operator to depressurize the intact SGs to 300 psig by locally dumping steam at the maximum rate using the steam generator PORVs. By establishing AFW to and using the associated SG PORV for one or both of the steam generators, it is possible to cool the RCS to around 410°F within a one hour time period. By reducing the temperature to 410°F or less, the RXCP seal leak rate is significantly reduced because the cooldown results in an RCS depressurization causes most of the contents of the accumulators to be injected. The success of this node requires 1 of 2 steam generator PORVs to operate.

Fault Tree

The analysis for node OCD consists of construction of a fault tree that predicts the failure of this node with respect to the requirements and assumptions listed above. This fault tree OCD is considered to fail if the operator fails to follow the EOPs or either steam generator PORV fails to open.

3.2.1.12.18 OP1 - Failure to Cooldown and Depressurize RCS

Function

This node applies to the medium break LOCA (MLO) accident sequence and represents the operator actions and system hardware failures that result in the failure to cooldown and depressurize the RCS.

Description

Upon failure of the high pressure SI, RCS inventory can be provided by the low pressure SI system. This requires operator action to cool down and depressurize the RCS by dumping steam from an intact steam generator to maintain a maximum 100 °F/hour cooldown rate. These operator actions are provided in Emergency Operating Procedure ES-1.2, Post LOCA Cooldown and Depressurization, which is entered from E-1, Loss of Reactor or Secondary Coolant. It is assumed that the operators have 15 minutes to initiate this action and that failure of this node results in early core melt because of the unavailability of all emergency core cooling systems (ECCS) to provide core cooling. The success of this node requires 1 of 4 steam dump valves operate for a defined mission time of 1 hour.

Fault Tree

The analysis for node OP1 consists of construction of a fault tree that predicts the failure of this node with respect to the requirements and assumptions listed above. This fault tree OP1 is considered to fail if the operator fails to follow the EOPs, or none of the steam dump valves are operable.

3.2.1.12.19 OP2 - Failure to Cooldown and Depressurize RCS

Function

This node applies to the SLO accident sequence and represents the operator actions and system hardware failures which would result in the failure to cooldown and depressurize the RCS.

Description

Upon failure of the HI2, RCS inventory can be provided by the accumulators and the low pressure SI system. This requires operator action to cool down and depressurize the RCS by dumping steam from an intact SG to maintain a maximum 100 °F/hour cooldown rate. These operator actions are provided in Emergency Operating Procedure ES-1.2, Post LOCA Cooldown and Depressurization, which is entered from E-1, Loss of Reactor or Secondary Coolant. It is assumed that the operators have 30 minutes to initiate this action, and that failure of this node results in early core melt because of the inability to provide inventory to the core by high pressure SI, the accumulators or low pressure SI. The success of this node requires 1 of 4 steam dump valves operate for a defined mission time of 24 hours.

Fault Tree

The analysis for node OP2 consists of construction of a fault tree that predicts the failure of this node with respect to the requirements and assumptions listed above. This fault tree OP2 is considered to fail if the operator fails to follow the EOPs, or none of the steam dump valves are operable.

3.2.1.12.20 ORI - Failure to Restore RCS Inventory

Function

The node applies to the SBO accident sequence and represents the operator actions and system hardware failures that result in the failure to restore RCS inventory.

Description

When power is restored, the emergency operating procedures instruct the operator to restore the safeguard systems. SI to restore RCS inventory is required and decay heat removal must be established or maintained. The operator actions and systems required depend on the postulated accident progression at the time that power is restored. To keep the analysis for ORI manageable, it is sufficient to model the basic actions and systems used in the recovery for a small LOCA.

If core uncovery has not occurred (success of fault tree CCV) and RXCP seal leakage is assessed as a small LOCA, high pressure SI with 1 out of 2 SI pumps is required to mitigate the event. The operator actions needed to restore the safeguard systems also include operation of at least one AFW pump with injection to at least one SG (otherwise bleed and feed recovery is used; at least one pressurizer PORV is opened for the bleed and feed contingency actions). Based on prior success for AF2, a secondary heat sink is still available (or can be quickly restored) at various times during the accident (ACX), so the success criterion is consistent with that required for small LOCA or bleed and feed recovery. The success of this node requires 1 of 3 AFW trains, 1 of 2 high pressure injection trains and 1 of 2 PORVs operate for a defined mission time of 24 hours.

Fault Tree

The analysis for node ORI consists of construction of a fault tree which predicts the failure of this node with respect to the requirements and assumptions listed above. This fault tree ORI is considered to fail if the operator fails to follow the EOPs, high pressure injection is not established, or auxiliary feedwater or bleed and feed is not established.

3.2.1.12.21 ORT - Failure to Deenergize 480V Buses 33 and 43

Function

This node applies to the AWS accident sequence and represents the operator action and system hardware failures that result in the failure to deenergize the electrical busses that supply the rod drive MG sets.

Description

Following an AWS, the operators are instructed by procedure to first manually trip the reactor. The failure of this action is accounted for in the MRT event tree node. If a manual trip is unsuccessful the operators are instructed to manually insert the control rods, manually open the supply breakers to the buses supplying the control rod drive MG sets, locally trip the reactor trip breakers, locally open the MG set supply breakers, and verify turbine trip and AFW pumps running. Due to the length of time required to achieve reactor trip from manually inserting the control rods, this method of tripping the reactor is considered in top event LTS.

Success for this event tree node requires the reactor to be tripped by the operators prior to steam generator dryout (within 2 minutes for an AWS at full power). This is accomplished manually by opening the supply breakers for buses 33 and 43 (which deenergizes the MG sets). The reactor can be tripped locally within 2 minutes by either opening 1 of 2 of the reactor trip breakers or by opening both of the MG set supply breakers. The success of this node requires the opening of 2 of 2 MG sets bus supply breakers.

Success of ORT precludes the need for AMSAC to function to mitigate the transient.

Fault Tree

The analysis for node ORT consists of construction of a fault tree that predicts the failure of this node with respect to the requirements and assumptions listed above. This fault tree ORT is considered to fail if the operator fails to follow the EOPs, or one supply breaker fails to open.

3.2.1.12.22 OS1 - Failure to Cool Down and Depressurize RCS

Function

This node applies to the SGR accident sequence and represents the operator actions and system hardware failures that result in the failure to cool down and depressurize the RCS.

Description

Success of this action requires the operator to successfully complete the actions in EOP E-3, Steam Generator Tube Rupture, to stabilize RCS pressure less than the ruptured SG pressure before the ruptured SG fills due to the addition of AFW and break flow. This normally requires three different high level operator actions: initial cooldown, RCS depressurization, and SI termination. The initial cooldown is performed using the intact SG supplied with feedwater, which has been isolated from the ruptured SG. The RCS depressurization is accomplished using normal or auxiliary spray, if available, or by opening one pressurizer PORV (and its associated block valve, if necessary). Of the three high level actions for OS1, the initial cooldown is the most essential one to model in the fault tree analysis. This is because success for SI termination is comparatively easy to demonstrate and the leak eventually causes the RCS to depressurize to the ruptured SG pressure if the SI pumps are secured and the intact SG is maintained at a lower pressure than the ruptured SG. These actions should be completed in about 30 minutes to prevent SG overfill for a design basis SGR event. This assumes simultaneous completion of all three actions at 30 minutes. The expected SG overfill time if the cooldown and depressurization are performed earlier could be significantly longer since the break flow is reduced while these actions are being completed.

Success of this node requires operator action to cool down and depressurize the RCS in order to stop the primary to secondary leak. These actions must be completed within 30 minutes. Success of OS1 results in no core melt if the RCS cooldown is successfully stopped. The success of this node requires 2 of 3 charging pumps and 1 of 2 normal spray valves or 1 of 1 auxiliary spray valve or 1 of 2 PORVs to operate for a defined mission time of 24 hours.

Fault Tree

The analysis for node OS1 consists of construction of a fault tree that predicts the failure of this node with respect to the requirements and assumptions listed above. This fault tree OS1 is considered to fail if the operator fails to follow the EOPs, the steam generator PORVs and steam dump valves fail or the pressurizer spray and PORVs fail.

3.2.1.12.23 OS2 - Failure to Cooldown and Depressurize RCS and Stop SI

Function

This node applies to the SGR accident sequence and represents the operator actions and system hardware failures which would result in the failure to cool down and depressurize the RCS and terminate SI flow.

Description

This event models the same actions as OS1. It is assumed, however, that the ruptured SG overfills prior to the completion of these actions. The 1982 Ginna event is an example of this case. For this SGR, the ruptured SG overfilled and one of the safety valves briefly opened, possibly several times. Upon SI termination, the safety valve did re-seat and the recovery proceeded normally.

Success of this node requires operator action to cooldown and depressurize the RCS and stop the primary to secondary leak after the ruptured SG has overfilled. Failure of this node results in late core melt. Success of this node requires 2 of 3 charging pumps and 1 of 2 normal spray valves or 1 of 1 auxiliary spray valves or 1 of 2 PORVs to operate for a defined mission time of 24 hours.

Fault Tree

The analysis for node OS2 consists of construction of a fault tree that predicts the failure of this node with respect to the requirements and assumptions listed above. This fault tree OS2 is considered to fail if the operator fails to follow the EOPs, the steam generator PORVs and steam dump valves fail or the pressurizer spray and PORVs fail.

3.2.1.12.24 OSD - Failure to Terminate Depressurization

Function

This node applies to the SGR accident sequence and represents the operator actions and system hardware failures that result in the failure to terminate RCS depressurization.

Description

Nodes OS1 and OS2 require a depressurization of the RCS. If the pressurizer PORVs are used for this purpose, there is a possibility that they do not close, in which case RCS inventory is lost through the PORVs.

Success of this node requires that either the pressurizer sprays fail and that the one PORV used for depressurization is successfully closed.

Success of OSD always results in no core melt. Failure of OSD and either SI injection or SI recirculation always results in core melt.

Fault Tree

The analysis for node OSD consists of construction of a fault tree that predicts the failure of this node. This fault tree OSD is considered to fail if the operator fails to close either the PORV or its associated block valve.

3.2.1.12.25 OSP - Failure of On-Site Power

Function

This node applies to the proper quantification of accident sequences LSP, TRA and TRS to eliminate cutsets that only apply to the SBO accident sequence.

Description

If offsite power is lost at any time during the transient, the emergency diesel generators (EDGs) are designed to automatically start and come up to speed within 10 seconds (see Reference 14). If the EDG fails to start the first time, several restarts will be attempted automatically. Operator action is then required for additional start attempts. Success for the OSP top event is to have emergency AC power (either from the EDGs or from offsite power) available to at least one of the two 4.16 kV emergency buses. If onsite power is lost, the failure to have AC power to these buses may be due to failure of the EDGs to start or to run, failure of the buses to shed loads, or failure of the EDGs to load. For a loss of offsite power with successful reactor trip, provision of emergency AC power could be delayed for as long as 30 minutes, i.e., a limiting time for SG secondary dryout. For event tree modeling, it is assumed that AC power is required for 24 hours. During this time, a fuel oil transfer pumps would be required to operate periodically to replenish the 850 gallon day tanks for the EDGs.

Fault Tree

The analysis for node OSP consists of construction of a fault tree that predicts the failure of this node with respect to the requirements and assumptions listed above. This fault tree OSP is considered to fail if power cannot be restored to both safeguard busses.

3.2.1.12.26 OSR - Failure to Throttle SI Flow

Function

This node applies to the Interfacing Systems LOCA (ISL) accident sequence and represents the operator action and system hardware failures that result in the failure to throttle SI flow.

Description

This node models the operator actions necessary to minimize ECCS flow upon recognition of loss of recirculation capability. These actions are directed by Emergency Operating Procedure ECA-1.1, Loss of Emergency Coolant Recirculation, which is entered from either step 16 of E-1, Loss of Reactor or Secondary Coolant, or step 3 of ECA-1.2, LOCA Outside Containment. The success of this node requires the operator to throttle manual valves SI-7A and SI-7B.

It is assumed that failure of this node will result in early core melt as the RWST would be depleted thus eliminating ECCS injection into the RCS.

Fault Tree

The analysis for the OSR node consists of construction of a fault tree that predicts the failure of this node with respect to the requirements and assumptions listed above. This fault tree OSR is considered to fail if either valve fails to close or the operator fails to take the appropriate actions.

3.2.1.12.27 OIP - Failure to Isolate RHR Pumps

Function

This node applies to the ISL accident sequence and represents the operator action and system hardware failures that result in the failure to isolate the RHR pumps.

Description

This node determines whether or not the operators are successful in manually closing valves RHR-4A and 4B to isolate both RHR pumps assuming that each pump's seal is leaking. Because of the length of time associated with this action, it is assumed that by the time this isolation is complete both RHR pumps are inoperable and RCS pressure is low enough, due to the pressure relief provided by the RHR relief valves, to not cause RHR piping failure. If this node is unsuccessful, core damage occurs unless a water source is available once the RWST is depleted.

Success of OIP requires 2 of 2 RHR pump manual isolation valves, RHR-4A and RHR-4B, to be closed.

Fault Tree

The analysis for the OIP node consists of construction of a fault tree that predicts the failure of this node with respect to the requirements and assumptions listed above. The fault tree OIP is considered to fail if either valve fails to close or the operator fails to take the appropriate actions.

3.2.1.12.28 PPR - Failure to Relieve RCS Pressure

Function

This node applies to the AWS accident sequence and represents the operator actions and system hardware failures which would result in the failure to relieve RCS pressure.

Description

This event tree node addresses the probability that pressurizer pressure relief capacity is adequate to prevent a peak RCS pressure in excess of 3200 psig. 3200 psig is the maximum RCS pressure limit for Westinghouse plants corresponding to the ASME Boiler and Pressure Vessel Code Level C service limit stress criteria. Early core damage is assumed to occur if this pressure limit is exceeded.

In Reference 3, an unfavorable exposure time (UET) is defined as the time during the fuel cycle life when the reactivity feedback is not sufficient to limit the RCS pressure for ATWS to less than 3200 psig for a given plant configuration (power level, manual rod insertion, auxiliary feedwater flow, and pressurizer PORV availability). Although the PORVs may be blocked for part of the cycle life, the pressurizer safety valves are assumed to be available through the cycle life. Success for this top event assumes both safety valves are operable.

To determine the success criterion of top event PPR for Kewaunee, an evaluation of the analysis described in Appendix B of Reference 3 was performed. From this evaluation, a success criterion of 2 safeties and 1 PORV available for RCS pressure relief is bounding for all but the first 40 days of the fuel cycle if top event AFG is successful (i.e., if 2 out of 3 AFW pumps are delivering at least 400 gpm to the SGs). Thus, this success criterion bounds about 90% of the days in the 12 month fuel cycle of Kewaunee. It should also be pointed out that the pressure requirements in the Reference 3 analysis are based on a "worst case" initiating event, i.e., an ATWS with loss of load and concurrent loss of main feedwater. In view of this, the success criterion is judged to be adequate for the purpose of event tree modeling. The success of this node requires the operation of 2 of 2 pressurizer safety valves and 1 of 2 PORVs for a defined mission time of 24 hours.

Fault Tree

The analysis for node PPR consists of construction of a fault tree that predicts the failure of this node with respect to the requirements and assumptions listed above. This fault tree PPR is considered to fail if either pressurizer safety valve fails to open or both PORVs fail to open.

3.2.1.12.29 RHR - No Residual Heat Removal Flow for RCS Cooldown

Function

This node is used as a subtree in ES1 for the SLO accident sequence and as a subtree in EC3 and EC4 for the SGR accident sequence and represents the RHR system hardware failures that result in the failure of the RHR system to provide flow for RCS cooldown.

Description

The RHR system consists of two 100% capacity redundant trains. Each RHR train consists of a RHR pump, residual heat exchanger, and associated piping, valves, and instrumentation. When the RHR System is used to remove core decay heat during plant cooldown, RCS flows from either or both of the loop hot legs, to the RHR pumps, through the residual heat exchangers, and returns into the B RCS cold leg. Heat loads from the residual heat exchanger are transferred to the CCW system on the shell side of the heat exchangers. The heat is eventually transferred to the service water (SW) system. Figure 3.2-36 provides a flow diagram of the RHR system in the cooldown mode.

The RCS cooldown rate is controlled by the remote/manual regulation of RCS flow through the tube side of the residual heat exchangers. This flow control is accomplished by the use of AOVs RHR-8A and RHR-8B located on the discharge of each heat exchanger. To maintain a constant flow through the RCS loops, a portion of the RCS flow is diverted, via a bypass line around the residual heat exchangers. This ensures a constant mixing and cooling flow with a relatively constant temperature drop across the heat exchanger. The bypass flow control valve AOV RHR-101 is automatically positioned, by flow controllers, to maintain a constant flow rate of 2000 gpm with one RHR pump running or 4000 gpm with both RHR pumps running.

At RCS pressure in the range of 0 to 425 psig and temperature less than 400°F, the RHR system is aligned in accordance with N-RHR-34 to provide low temperature overpressure protection and normal RCS cooling.

The RWST suction valves, MOVs SI-300A and SI-300B are closed. The manual isolation valves RHR-10A, RHR-10B, RHR-100A, and RHR-100B are opened. CCW flow is established to both residual heat exchangers by opening CC-400A and CC-400B.

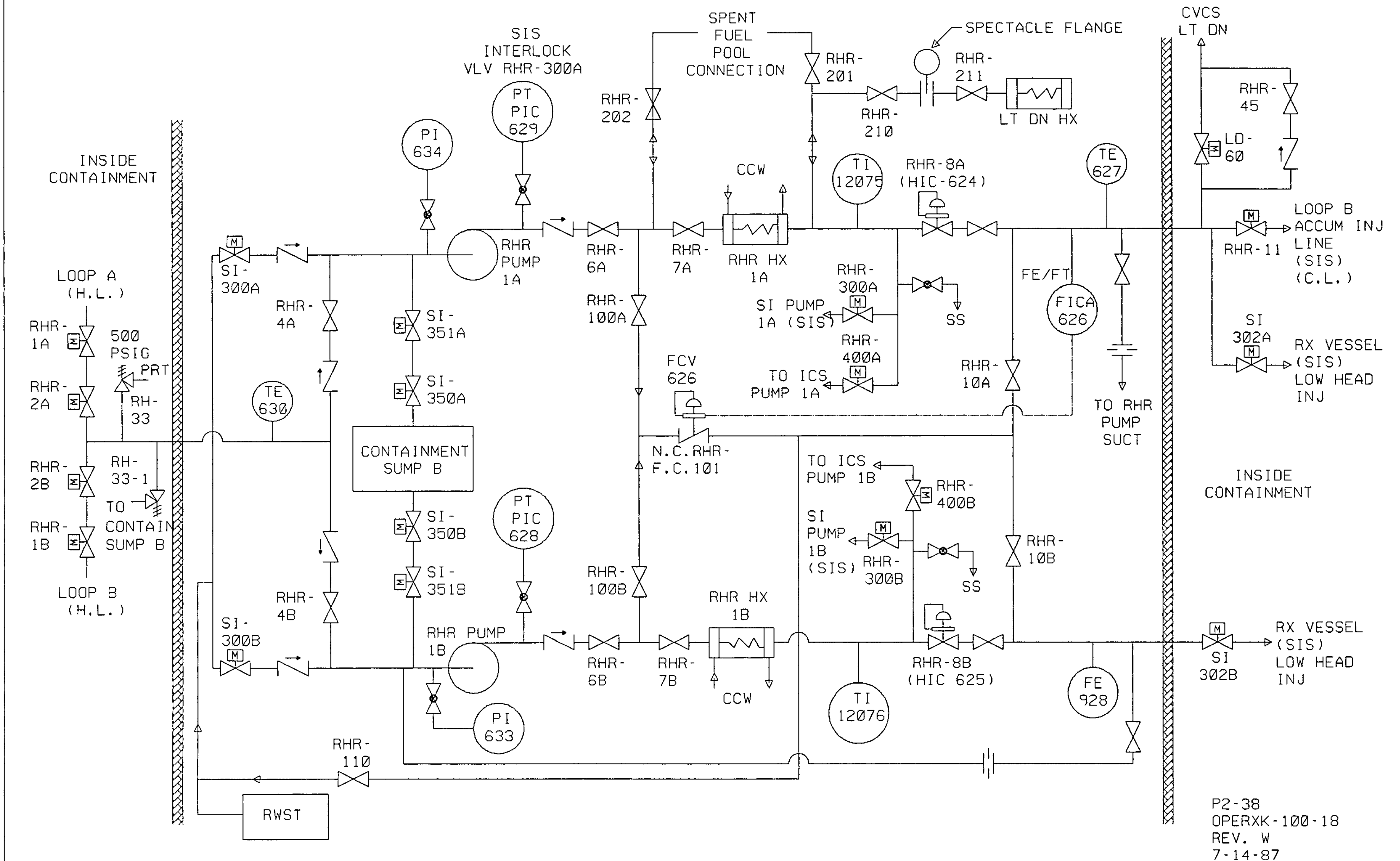
Prior to opening the RHR isolation valves MOVs RHR-1A, RHR-1B, RHR-2A and RHR-2B, LD-60 is opened to crossconnect the RHR System to the chemical and volume control system (CVCS). LD-10 is adjusted to equalize CVCS and RHR pressure with the RCS pressure. RHR-1A, RHR-1B, RHR-2A, and RHR-2B are opened to connect RHR System to the RCS hot legs. Relief valves RHR-33 and RHR-33-1 provide overpressure protection for the RCS.

The RHR system bypass flow control valve is adjusted to 10% open and a RHR pump is started. The RHR system is slowly warmed by controlling flow through RHR-101. Prior to opening RHR-11, the boron concentration in the RHR System is verified to be not more than 100 ppm lower than the RCS. RHR-11 is then opened and flow is set at 2000 gpm for one RHR pump (4000 gpm for two RHR pumps) operating. RHR-8A and/or RHR-8B are now slowly opened to warm the heat exchangers and establish the desired cooldown rate in the RCS. Success of this node requires 1 of 2 RHR trains operate in the cooldown mode for a defined mission time of 24 hours.

Fault Tree

The analysis for node RHR consists of construction of a fault tree that predicts the failure of this node with respect to the requirements and assumptions listed above. This fault tree RHR is considered to fail if both trains of RHR fail.

FIGURE 3.2-36
RHR SYSTEM - COOLDOWN



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3.2.1.12.30 RVC - RHR Pump Relief Valves Fail to Reclose

Function

This node applies to the ISL accident sequence and represents the system hardware failures that would result in the residual heat removal pump relief valves failing to close.

Description

This node models the success of the RHR relief valves to close once RCS/RHR pressure is below the relief setpoint of both valves (approximately 480 psig).

Success of RVC is 2 of 2 relief valves closed. Failure of this node results in a failure to isolate the LOCA through the relief valves. This results in early core melt unless ECCS flow is minimized.

Fault Tree

The analysis for the RVC node consists of construction of a fault tree that predicts the failure of this node with respect to the requirements and assumptions listed above. The fault tree RVC is considered to fail if either valve fails to reclose.

3.2.1.12.31 SSV - Integrity Not Maintained or Restored in Ruptured Steam Generator

Function

This node applies to the SGR accident sequence and represents the system hardware failures that result in the failure to restore and maintain the integrity of the ruptured SG.

Description

If one of the secondary relief valves sticks opens following overfill of the ruptured SG, the SGR recovery strategy becomes somewhat more complicated. The operator transitions to EOP ECA-3.1, SGTR With Loss of Reactor Coolant - Subcooled Recovery Desired, and possibly to ECA-3.2, SGTR With Loss of Reactor Coolant - Saturated Recovery Desired. Success is defined as all 5 safety valves and the PORV closing to maintain or restore secondary integrity after the E-3, Steam Generator Tube Rupture, actions are complete.

It is assumed that if SSV fails and high pressure SI is available, recovery actions in ECA-3.1 or ECA-3.2 (EC4) must be addressed. If SSV fails and high pressure SI is unavailable, early core melt is assumed.

Fault Tree

The analysis for node SSV consists of construction of a fault tree that predicts the failure of this node with respect to the requirements and assumptions listed above. This fault tree SSV will be considered to fail if any of the 6 valves (5 safety, 1 PORV) fails to close.

3.1.2.12.32 IAS - Loss of Instrument Air

Function

Used as a subtree in the various system fault trees that require instrument air support. This fault tree is also used to calculate loss of instrument air initiating event frequency in fault tree IAIE but with different mission times. This node represents the system hardware failures that result in a loss of instrument air (IA).

Description

The station air (SA) and IA system provides compressed air for instruments, control systems, air operated valves, and maintenance operations. The SA and IA system filters, compresses, cools, and stores oil-free air at approximately 100 psig. The SA and IA system consists of six air compressors designated as A, B, C, D, E and F which are shown on Figure 3.2-37.

The six compressors supply two SA headers and thirteen IA headers. Prior to distribution to the IA headers, the oil-free compressed air is further dried and filtered by dryer/filter units A, B and C. Normally, C Dryer is in service, and units A and B are valved out in standby, IA is supplied for various vital services and is uncontaminated by oil, water vapor, or dirt.

Since the IA headers are necessary for normal plant operation, the SA headers have automatic pressure regulating valves SA-200 and SA-400 to shut off SA usage in the event of a leak in order to maintain the IA supply pressure. A continued drop in SA and IA header pressure to 90 psig causes the automatic closing of isolation valves SA-200 and SA-400. In an emergency, portable sources of oil-free compressed air may be connected to supply the system from either the fuel cask shipping area or the turbine building truck access area. Hand operators are provided to open isolation valves SA-200 and SA-400 when supplying air from an emergency/portable source.

The IA system is required to support normal plant operation. However, it is not a safeguards system. Air operated equipment is designed to fail in the safe position upon loss of air. This failure mode is a requirement for loss of air to particular pieces of equipment. The failure mode does not pertain to a systemic loss of all air to plant systems. The dedicated and alternate IA headers, together with IA accumulators have been installed to provide additional assurance of IA supply to certain plant components as described herein.

The SA and IA system design includes three high-capacity air compressors, which are supported by non-vital power supplies and three small-capacity air compressors, which are supported by vital power supplies. The system is designed to supply oil-free air to the SA and IA headers and to remove moisture from the air supplied to the IA system to a dew point of -40°F at normal pressure. The IA system operates to maintain greater than 90 psig minimum air pressure.

Large line sizes, which result in low flow rates through the system, are used in the design of the SA and IA System. Low air flow rates are desirable to minimize pressure losses due to flow.

During normal operation, one or two high-capacity compressors supply sufficient compressed air. In addition, one small-capacity air compressor is running unloaded. If SA or IA system pressure falls, the backup small-capacity compressors automatically start at their given setpoint and load as necessary to maintain system pressure.

To enhance the reliability of the IA system and assure the capability to shutdown the plant, Appendix R design modifications subdivided the IA System. Appendix R design criteria identified two IA headers, the dedicated and alternate headers, to enable a controlled shutdown of the plant in the event of a fire. These headers provide a reliable air supply, primarily to air operated valves outside containment, to maintain designated equipment operable. All other valves that require an air supply to maintain valve position or to be cycled a specified number of times are provided with local air accumulators. These local air accumulators provide a quantity of pressurized air to maintain the valves operable for a specified time period.

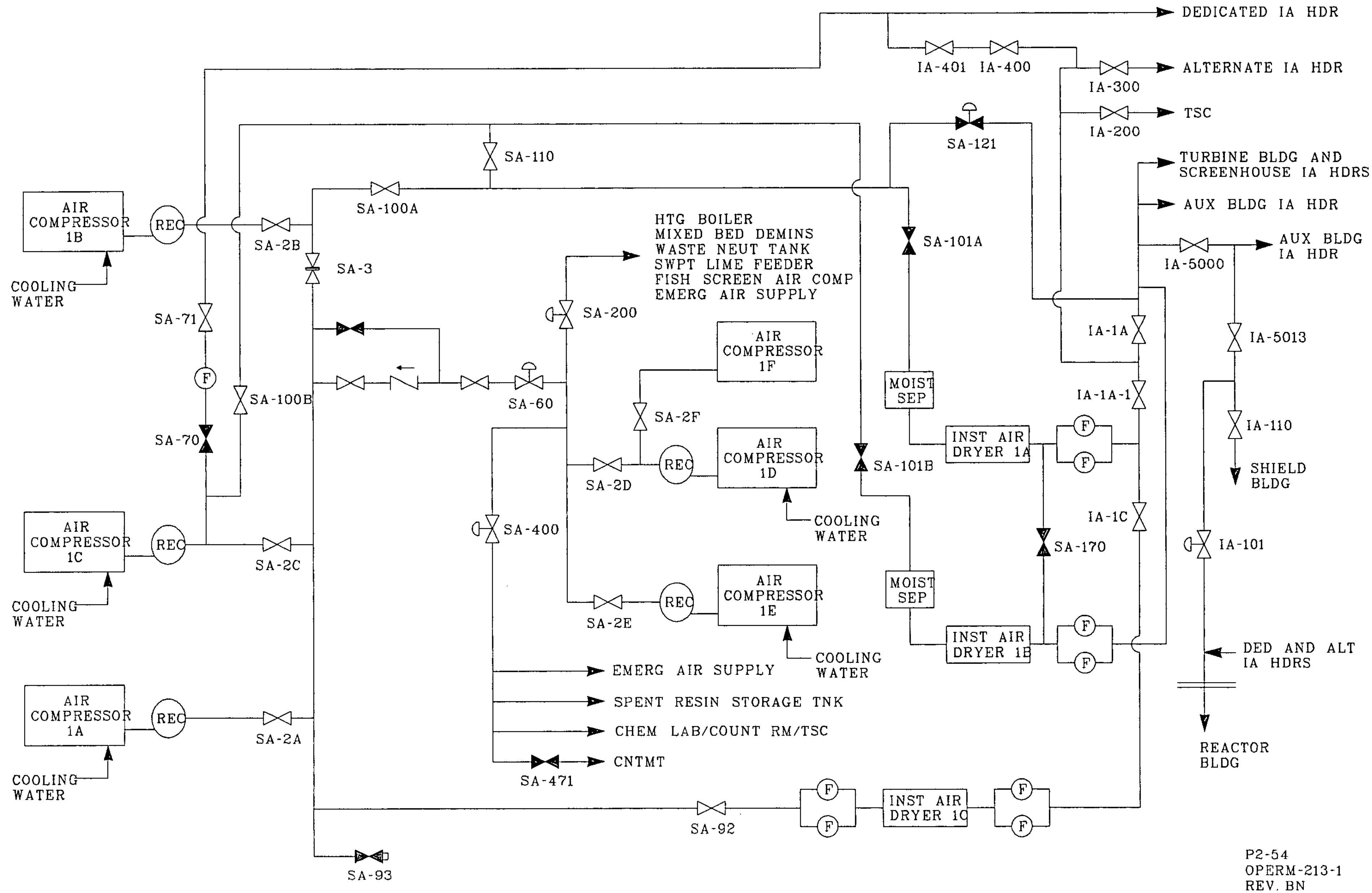
The dedicated IA header is isolated from the balance of the IA system by closing IA-401. Air compressor C is powered from DG A and is the dedicated IA header compressor. Manual valves SA-2C and SA-100B are closed to isolate air compressor C from the balance of the IA system. Valve SA-70 is opened to supply air directly from air compressor C to the dedicated IA header. The success of this node requires the operation of 1 of 3 non-vital compressors or 1 of 3 vital air compressors with 2 of 2 air header isolation valves closed or 2 of 3 vital air compressors if either of the air header isolation valves fails to close.

Fault Tree

The analysis for node IAS consists of construction of a fault tree that predicts the failure of this node with respect to the requirements and assumptions listed above. This fault tree IAS is considered to fail if all of the instrument air compressors fail or the instrument air piping fails, or the air filter system fails and the bypass valve fails to open.

FIGURE 3.2-37

STATION AND INSTRUMENT AIR SYSTEM



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3.2.1.12.33 IASP - Loss of Instrument Air (LSP)

Function

This fault tree is used as a subtree in OB2 which is used for the LSP accident sequence. This node represents the system hardware failures which would result in the loss of IA.

Description

There are 6 air compressors in the station and instrument air system is stated in Section 3.2.1.12.32. For a LSP event, compressors D, E and F are unavailable. The success of this node requires the operation of 1 of 3 vital air compressors with 2 of 2 air headers isolation vales close or 2 of 3 vital air compressors if either air header isolation valve fails to close.

Fault Tree

The analysis for node IASP consists of construction of a fault tree that predicts the failure of this node with respect to the requirements and assumptions listed above. This fault tree IASP will be considered to fail if all of the instrument air compressors fail or the instrument air piping fails, or the air filter system fails and the bypass valve fails to open.

3.2.1.12.34 IASPT - Loss of Instrument Air Termination for LSP

Function

This node is used as a subtree in service water system fault trees SWAP and SWBP in Loss of Offsite Power LSP accident sequence. This node represents the system hardware failures that result in the loss of IA to the SW system.

Description

This fault tree was created to break the SW system - IA system dependency loop.

Fault tree

The analysis for this node is the same as for IASP in section 3.2.1.12.33.

3.2.1.12.35 IAST - Loss of Instrument Air Termination

Function

This node is used as a subtree in SW system fault trees SWA, SWB, SWIE and SWT. This node represents the system hardware failures that result in the loss of IA to the service water system.

Description

This fault tree was created to break the SW system - IA system dependency loop.

Fault Tree

The analysis for this node is the same as for IAS in section 3.2.1.12.32.

3.2.1.12.36 IASTA - Loss of Instrument Air Termination

Function

This node is used as a subtree in SW system fault tree SWAG. This node represents the system hardware failures that result in loss of IA to the SW system that supplies diesel generator A.

Description

This fault tree was created to break the SW system - IA system dependency loop.

Fault Tree

The analysis for this node is the same as for IAS in section 3.2.1.12.32.

3.2.1.12.37 IASTB - Loss of Instrument Air Termination

Function

This node is used as a subtree in SW system fault tree SWBG. This node represents the system hardware failures that result in the loss of IA to the SW system that supplies diesel generator B.

Description

This fault tree was created to break the SW system - IA system dependency loop.

Fault Tree

The analysis for this node is the same as for IAS in section 3.2.1.12.32.

3.2.1.12.38 IASD - Loss of Instrument Air for TDC

Function

This node is used as a subtree in OB3 and OM4 fault trees for the Loss of 125V DC Bus accident sequence. This node represents system hardware failure that result in a loss of IA.

Description

This fault tree calls SW system fault trees SWAD and SWTD where a DC bus has failed.

Fault Tree

The analysis for this node is the same as for IAS in section 3.2.1.12.32.

3.2.2 System Analysis

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

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

The approach used to develop the fault tree models is consistent with the guidance provided in the Kewaunee Nuclear Power Plant Fault Tree Guidelines, which were developed to ensure that a consistent approach was used in establishing modeling assumptions and in structuring the models. Guidance is provided in areas such as the selection of random hardware failures, 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 modeled system to provide the level of detail required for the modeling of the system. 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 Kewaunee drawings from which the simplified diagrams were derived are identified in the system notebooks.

STEP 2 Develop Fault Tree

Step 2.1 Establish scope of fault tree - The fault tree guidelines were used to establish what modes and basic events should be modeled. They provided guidance in the 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 do not need to be included due to their low probability of occurrence relative to other events (e.g., certain passive failures).

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 completely 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). Rules were applied to determine the node level modules to be used based on the system success criteria and flow requirements. The fault tree was developed graphically with the Westinghouse GRAFTER Code System.

STEP 3 Quantify Fault Tree

The fault trees were quantified using the GRAFTER Code System to determine an initial system failure probability and to obtain the minimum cutsets.

Step 3.1 Calculate basic event probabilities - Using 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 Section 4.3 of the Kewaunee PRA notebooks entitled "Data Analysis" (refer to section 3.3.1/2 of this report).

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 Technique for Human Error Rate Prediction (THERP) methodology and described in section 4.15 of the Kewaunee PRA notebooks entitled "Human Reliability Analysis" (refer to section 3.3.3 of this report).

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. The common cause analysis is described in section 4.2 of the Kewaunee PRA notebooks entitled "Common Cause Model" (refer to section 3.3.4 of this report).

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 Documentation Process

The entire process of fault tree development including key assumptions, boundary conditions, and other important information was documented in the system notebooks. The quantification of the fault trees and key insights were identified and also documented in the system notebooks.

3.2.3 System Dependencies

One of the important aspects of system modeling is correctly taking into account the dependencies that equipment from one system have on equipment from other systems. Two tables were developed, the first of which is the traditional dependency matrix at a system support level. This table includes both front line and support system dependencies and is summarized in Table 3.2-3. The second set of tables are at a component level and are provided as Table 3.2-4.

The component level dependency set of tables include all active mechanical-type equipment. The support functions listed in the table include electrical, coolant and air along with the component position in both normal and post safety injection signal conditions, if applicable. This set of tables was especially helpful in developing the system fault trees in the Kewaunee PRA.

TABLE 3.2-3

PLANT SYSTEM DEPENDENCY MATRIX

	SWS	CCW	4160V	480V	120V	DC	IAS	RPS	LPSI	HVAC	EDG
AFW	X		X	X		X		X		X	
CAT	X			X		X		X			
CI				X				X			
ICS		X		X		X		X	X	X	
LPSI	X	X	X	X		X		X		X	
HPSI	X	X	X	X		X		X	X	X	
MFW	X		X	X		X	X				
SWS			X	X	X	X	X	X			
CCW	X			X		X		X			
IAS	X			X							
RPS					X	X					
4160V						X		X			X
480V			X			X					
120V				X		X					
DC				X							
EDG	X					X		X		X	
HVAC	X			X				X			
CVCS				X			X				

TABLE 3.2-4

COMPONENT LEVEL SUPPORT REQUIREMENTS
ECCS INJECTION

Active Components	Electrical	Support System Coolant (Water)	Air	Other Coolant	Position Normal	Position SI Signal
Low Pressure Injection						
RHR Pump A	4160V Bus 5 125V DC BRA-104	Note 1	-	Note 2	Standby	Start/Run
RHR Pump B	4160V Bus 6 125V DC BRB-104	Note 1	-	Note 2	Standby	Start/Run
SI-300A	MCC 52E	-	-	-	Open	-
SI-300B	MCC 62E	-	-	-	Open	-
SI-302A	MCC 52B	-	-	-	Open	Open
SI-302B	MCC 62B EXT	-	-	-	Open	Open
High Pressure Injection						
SI Pump A	4160V Bus 5 125V DC BRA-104	Component Cooling	-	Service Water	Standby	Start/Run

TABLE 3.2-4

COMPONENT LEVEL SUPPORT REQUIREMENTS (Continued)
ECCS INJECTION

Active Components	Electrical	Support System Coolant (Water)	Air	Other Coolant	Position Normal	Position SI Signal
SI Pump B	4160V Bus 6 125V DC BRB-104	Component Cooling	-	Service Water	Standby	Start/Run
SI-2A	MCC 52E	-	-	-	Closed	Open (Note 4)
SI-2B	MCC 62E	-	-	-	Closed	Open (Note 4)
SI-4A	MCC 52E	-	-	-	Closed	- (Note 4)
SI-4B	MCC 62E	-	-	-	Closed	- (Note 4)
SI-5A	MCC 52E	-	-	-	Open	-
SI-5B	MCC 62H	-	-	-	Open	-

TABLE 3.2-5

COMPONENT LEVEL SUPPORT REQUIREMENTS
ECCS RECIRCULATION

Active Components	Electrical	Support System Coolant (Water)	Air	Other Coolant	Position Normal	Position SI Signal	Position Recirc.
Low Pressure Recirculation							
RHR Pump A	4160V Bus 5 125V DC BRA-104	Component Cooling (Pump and RHR Hx Cooling)	-	Service Water (RHR Pump Pit Cooling)	Standby	Start/Run	Run
RHR Pump B	4160V Bus 6 125V DC BRB-104	Component Cooling (Pump and RHR Hx Cooling)	-	Service Water (RHR Pump Pit Cooling)	Standby	Start/Run	Run
SI-300A	MCC 52E	-	-	-	Open	-	Closed
SI-300B	MCC 62E	-	-	-	Open	-	Closed
SI-350A	MCC 52 EXT	-	-	-	Closed	-	Open
SI-350B	MCC 62B	-	-	-	Closed	-	Open
SI-351A	MCC 52E	-	-	-	Closed	-	Open
SI-351B	MCC 62H	-	-	-	Closed	-	Open
SI-302A	MCC 52B	-	-	-	Closed	Open	Open

TABLE 3.2-5

COMPONENT LEVEL SUPPORT REQUIREMENTS (Continued)
ECCS RECIRCULATION

Active Components	Electrical	Support System Coolant (Water)	Air	Other Coolant	Position Normal	SI Signal	Recirc.
SI-302B	MCC 62B EXT	-	-	-	Closed	Open	Open
CC-400A	480V 52B	-	-	-	Closed	-	Open
CC-400B	480V 62E	-	-	-	Closed	-	Open
RHR-8A	Non-Safeguards	-	-	-	Open	-	Open
RHR-8B	Non-Safeguards	-	-	-	Open	-	Open
RHR Pump Pit Fan Coil 1A	MCC 52E	Service Water	-	-	Standby	Start/Run	Run
RHR Pump Pit Fan Coil 1B	MCC 62E	Service Water	-	-	Standby	Start/Run	Run
Cooling Water Valve (SW-1211A)	120V AC BRA-105	-	-	-	Standby	*	*
Cooling Water Valve (SW-1221B)	120V AC BRB-105	-	-	-	Standby	*	*

*Temperature Control

TABLE 3.2-5

COMPONENT LEVEL SUPPORT REQUIREMENTS (Continued)
ECCS RECIRCULATION

Active Components	Electrical	Support System Coolant (Water)	Air	Other Coolant	Position Normal	SI Signal	Recirc.
High Pressure Recirculation							
SI-Pump A	4160V Bus 5 125V DC BRA-104	Component Cooling	-	Service Water	Standby	Start/Run	Run
SI-Pump B	4160V Bus 6 125V DC BRB-104	Component Cooling	-	Service Water	Standby	Start/Run	Run
SI-2A	MCC 52E	-	-	-	Closed	Open	Closed
SI-2B	MCC 62E	-	-	-	Closed	Open	Closed
SI-4A	MCC 52E	-	-	-	Closed	-	Open
SI-4B	MCC 62E	-	-	-	Closed	-	Open
SI-5A	MCC 52E	-	-	-	Open	-	Closed
SI-5B	MCC 62H	-	-	-	Open	-	Closed
RHR-300A (Note 6)	MCC 52E	-	-	-	Closed	-	Open
RHR-300B (Note 6)	MCC 62H	-	-	-	Closed	-	Open

TABLE 3.2-5

COMPONENT LEVEL SUPPORT REQUIREMENTS (Continued)
ECCS RECIRCULATION

Active Components	Electrical	Support System Coolant (Water)	Air	Other Coolant	Position Normal	SI Signal	Recirc.
SI-101A	RR-174 CKT27	-	To Open	-	Closed	-	Closed
SI-101B	RR-174 CKT27	-	To Open	-	Closed	-	Closed

NOTES FOR TABLES 3.2-4 AND 3.2-5

All ECCs pumps fail stopped on loss of motive power; fail as is on loss of control power.

Note 1: CCW is supplied to the RHR pumps for cooling. However, during low pressure injection, the coolant flow from the RWST is considered sufficient to keep the RHR pump cool.

Note 2: SW is supplied to the RHR pump pit fan coolers to provide room cooling. During low pressure injection, the coolant flow from the RWST is considered to be sufficient to preclude the need for room cooling.

Note 3: Valve has its breaker locked in the open position

Note 4: Interlock with BAT level, SI-2A/2B close and SI-4A/4B open.

Note 5: Deleted.

Note 6: This valve is interlocked with RHR punıp pressure and valve SI-5A(B).

TABLE 3.2-6

COMPONENT LEVEL SUPPORT REQUIREMENTS
SERVICE WATER SYSTEM

Active Components	Electrical	Support System Coolant (Water)	Air	Other Coolant	Position Normal	Position SI Signal	Position Recirc.
SW Pump A1	4160V Bus 5 125V DC BRA-104	(Note 1)	-	-	Standby (Note 2)	Start/Run	Run
SW Pump A2	4160V Bus 5 125V DC BRA-104	(Note 1)	-	-	Running	Run	Run
SW Pump B1	4160V Bus 6 125V DC BRB-104	(Note 1)	-	-	Running	Run	Run
SW Pump B2	4160V Bus 6 125V DC BRB-104	(Note 1)	-	-	Running	Run	Run
STRAINER A1	MCC 52D	-	-	-	Standby (Note 2)	Start/Run	Run
STRAINER A2	MCC 52D	-	-	-	Running	Run	Run
STRAINER B1	MCC 62D	-	-	-	Running	Run	Run
STRAINER B2	MCC 62D	-	-	-	Running	Run	Run
Trav. Screen A1	MCC 52D	(Note 5)	(Note 5)	-	*	*	*

TABLE 3.2-6

COMPONENT LEVEL SUPPORT REQUIREMENTS (Continued)
SERVICE WATER SYSTEM

Active Components	Electrical	Support System Coolant (Water)	Air	Other Coolant	Position Normal	SI Signal	Recirc.
Trav. Screen A2	MCC 35C	(Note 5)	(Note 5)	-	*	*	*
Trav. Screen B1	MCC 45C	(Note 5)	(Note 5)	-	*	*	*
Trav. Screen B2	MCC 62D	(Note 5)	(Note 5)	-	*	*	*
SW-10A	MCC 52A	-	-	-	Open	-	Open
SW-10B	MCC 62A	-	-	-	Open	-	Open
SW-301A	125V DC BRA-104	-	To Close	-	Closed	Open	Open (Note 3)
SW-301B	125V DC BRB-104	-	To Close	-	Closed	Open	Open (Note 3)
SW-30A2	120V BRA-127	(Note 4)	To Open		*	*	*
SW-30B1	120V BRB-127	(Note 4)	To Open		*	*	*
SW-30B2	120V BRB-127	(Note 4)	To Open		*	*	*
SW-202A1	120V BRA-127	(Note 4)	To Open		*	*	*
SW-202A2	120V BRA-127	(Note 4)	To Open		*	*	*

TABLE 3.2-6

COMPONENT LEVEL SUPPORT REQUIREMENTS (Continued)
SERVICE WATER SYSTEM

Active Components	Electrical	Support System Coolant (Water)	Air	Other Coolant	Position Normal	SI Signal	Recirc.
SW-202B1	120V BRB-127	(Note 4)	To Open		*	*	*
SW-202B2	120V BRB-127	(Note 4)	To Open		*	*	*

*Intermittent operation based on system control demand for all sequences.

Note 1: Seal Water Supply
Note 2: Assumption for FT Modeling
Note 3: If Diesel Generator Operating
Note 4: Air and Service Water During Backwash Cycle
Note 5: Service Water During Backwash Cycle.

TABLE 3.2-7

COMPONENT LEVEL SUPPORT REQUIREMENTS
COMPONENT COOLING WATER

Active Components	Electrical	Support System Coolant (Water)	Air	Other Coolant	Position Normal	Position SI Signal	Position Recirc.
CCW Pump A	480V Bus 51 125V DC BRA-104	-	-	-	Running	Run	Run
CCW Pump B	480V Bus 61 125V DC BRB-104	-	-	-	Standby (Note 1)	Start/Run	Run
CC-6A	MCC 52B	-	-	-	Open	Open	Open
CC-6B	MCC 62B	-	-	-	Open	Open	Open
SW-1300A	MCC 52B	-	-	-	Closed	Open	Open
SW-1300B	MCC 62E	-	-	-	Closed	Open	Open

Note 1: Assumption for FT Modeling

TABLE 3.2-8

COMPONENT LEVEL SUPPORT REQUIREMENTS AUXILIARY FEEDWATER

Active Components	Electrical	Support System Coolant (Water)	Air	Other Coolant	Position Normal	SI Signal
AFW Pump A	4160V Bus 5 125V DC BRA-104	(Note 1)	-	(Note 2)	Standby	Start/Run
AFW Pump B	4160V Bus 6 125V DC BRB-104	(Note 1)	-	(Note 2)	Standby	Start/Run
AFW Pump C	Main Steam 125V DC BRA-104	(Note 1)	-	(Note 2)	Standby	Start/Run
ALOP C	125V DC BRA-104	-	-	-	Standby	Start/Run
SW-601A	MCC 52C	-	-	-	Closed	Note 3
SW-601B	MCC 62C	-	-	-	Closed	Note 3
SW-502	125V DC BRA-104	-	-	-	Closed	Note 3
AFW-2A	120V AC BRA-115 125V DC BRA-104	- -	To Close	-	Open	-
AFW-2B	120V AC BRA-115	-	To Close	-	Open	-
AFW-10A	125V DC BRA-104	-	-	-	Open	-

TABLE 3.2-8

COMPONENT LEVEL SUPPORT REQUIREMENTS (Continued)
AUXILIARY FEEDWATER

Active Components	Electrical	Support System Coolant (Water)	Air	Other Coolant	Position Normal	Position SI Signal
AFW-10B	125V DC BRB-104	-	-	-	Open	-
MS-100A	MCC 52E	-	-	-	Open	-
MS-100B	MCC 62J	-	-	-	Open	-
MS-102	125V DC BRA-104	-	-	-	Closed	Note 4
AFW-111A	120V AC BRA-127	-	-	-	Closed	Note 5
AFW-111B	120V AC BRA-127	-	-	-	Closed	Note 5
AFW-111C	120V AC BRA-127	-	-	-	Closed	Note 5

Note 1: Cooled by CST Water Being Pumped
Note 2: Cooled by SW When CST Depleted
Note 3: Operator Opens Valve when CST Depleted
Note 4: Valve Opens on Bus 1 & 2 UV, SG Lo-Lo Level and AMSAC Actuation
Note 5: Valve Opens on Associated Pump Start

TABLE 3.2-9

COMPONENT LEVEL SUPPORT REQUIREMENTS
MAIN FEEDWATER

Active Components	Electrical	Support System Coolant (Water)	Air	Other Coolant	Position Normal	Position SI Signal
FW Pump A	4160V Bus 1 125V DC BRC-103	Service Water	Note 1	-	Running	Off
FW Pump B	4160V Bus 2 125V DC BRD-103	Service Water	Note 1	-	Running	Off
Cond. Pump A	4160V Bus 3 125V DC BRC-103	Service Water	-	-	Running	Running
Cond. Pump B	4160V Bus 4 125V DC BRD-103	Service Water	-	-	Running	Running
FWP A ALOP	MCC 32G	-	-	-	Standby	Note 2
FWP B ALOP	MCC 42G	-	-	-	Standby	Note 2
FW-2A	MCC 32G	-	-	-	Open	Note 3
FW-2B	MCC 42G	-	-	-	Open	Note 3
FW-7A	125V DC BRB-104/ BRA-104	-	To Open	-	Open	Closed

TABLE 3.2-9

COMPONENT LEVEL SUPPORT REQUIREMENTS (Continued)
MAIN FEEDWATER

Active Components	Electrical	Support System Coolant (Water)	Air	Other Coolant	Position Normal	Position SI Signal
FW-7B	125V DC BRB-104/ BRA-104	-	To Open	-	Open	Closed
FW-10A	125V DC BRB-104/ BRA-104	-	To Open	-	Closed	Closed
FW-10B	125V DC BRB-104/ BRA-104	-	To Open	-	Closed	Closed
FW-12A	MCC 52E	-	-	-	Open	Closed
FW-12B	MCC 62J	-	-	-	Open	Closed

Note 1: Air used for seal water reg. valves and pump recirc. valves
Note 2: Pump starts as lo pressure decreases after pump trip
Note 3: Valve closes when pump is tripped

TABLE 3.2-10

COMPONENT LEVEL SUPPORT REQUIREMENTS
RESIDUAL HEAT REMOVAL

Active Components	Electrical	Support System Coolant (Water)	Air	Other Coolant	Position Normal	Position SI Signal
RHR Pump A	4160V Bus 5 125V DC BRA-104	Component Cooling	-	-	Standby	Start/Run
RHR Pump B	4160V Bus 6 125V DC BRB-104	Component Cooling	-	-	Standby	Start/Run
RHR-1A	MCC 52B Ext.	-	-	-	Closed	-
RHR-1B	MCC 62B Ext.	-	-	-	Closed	-
RHR-2A	MCC 52B Ext.	-	-	-	Closed (Note 1)	-
RHR-2B	MCC 62B Ext.	-	-	-	Closed (Note 1)	-
RHR-8A	NV Inst. Power	-	To Close	-	Open	-
RHR-8B	NV Inst. Power	-	To Close	-	Open	-
RHR-101	NV Inst. Power	-	To Open	-	Closed	-
CC-400A	MCC 1-52B	-	-	-	Closed	-
CC-400B	MCC 1-62E	-	-	-	Closed	-

TABLE 3.2-10

COMPONENT LEVEL SUPPORT REQUIREMENTS (Continued)
RESIDUAL HEAT REMOVAL

Active Components	Electrical	Support System Coolant (Water)	Air	Other Coolant	Position Normal	Position SI Signal
RHR Fan Coil A	MCC 52E	-	-	Service Water	Standby	Start/Run
RHR Fan Coil B	MCC 62E	-	-	Service Water	Standby	Start/Run
RHR-11	MCC 52B	-	-	-	Closed (Note 1)	-

Note 1: Breaker Off and Locked

TABLE 3.2-11

COMPONENT LEVEL SUPPORT REQUIREMENTS
CHEMICAL AND VOLUME CONTROL

Active Components	Electrical	Support System Coolant (Water)	Air	Other Coolant	Position Normal	SI Signal
Charging Pump A	MCC 52E	-	(Note 1)	-	(Note 2)	Start/Run
Charging Pump B	MCC 62E	-	(Note 1)	-	(Note 2)	Start/Run
Charging Pump C	Bus 52	-	-	-	(Note 2)	Start/Run
Boric Acid Pump A	MCC 52B	-	-	-	Run/Standby	Start/Run
Boric Acid Pump B	MCC 62E	-	-	-	Run/Standby	Start/Run
MOV CVC-440	MCC 52E	-	-	-	Closed	Closed

Note 1: To operate above min. speed
Note 2: Two of three pumps running with one in auto

TABLE 3.2-12

COMPONENT LEVEL SUPPORT REQUIREMENTS
CONTAINMENT SPRAY SYSTEM

Active Components	Electrical	Support System Coolant (Water)	Air	Other Coolant	Position Normal	Position P Signal	Position Recirc.
ICS Pump A	480V Bus 51 125V DC BRA-104	Component Cooling	-	-	Standby	Start/Run	Run (Note 2)
ICS Pump B	480V Bus 61 125V DC BRB-104	Component Cooling	-	-	Standby	Start/Run	Run (Note 2)
ICS-2A	MCC 52E	-	-	-	Open	-	Closed
ICS-2B	MCC 62E	-	-	-	Open	-	Closed
ICS-5A	MCC 52E	-	-	-	Closed	Open	Open
ICS-5B	MCC 62E	-	-	-	Closed	Open	Open
ICS-6A	MCC 52E	-	-	-	Closed	Open	Open
ICS-6B	MCC 62E	-	-	-	Closed	Open	Open
ICS-201	125V DC BRA-104	-	To Open	-	Open	(Note 1)	Closed
ICS-202	125V DC BRB-104	-	To Open	-	Open	(Note 1)	Closed
RHR-400A	MCC 52E	-	-	-	Closed	-	Open

TABLE 3.2-12

COMPONENT LEVEL SUPPORT REQUIREMENTS (Continued)
CONTAINMENT SPRAY SYSTEM

		Support System			Position		
Active Components	Electrical	Coolant (Water)	Air	Other Coolant	Normal	P Signal	Recirc.
RHR-400B	MCC 62H	-	-	-	Closed	-	Open

Note 1: Valve closes on containment isolation actuation.
Note 2: RHR System provides the water source in the recirc mode.

TABLE 3.2-13

COMPONENT LEVEL SUPPORT REQUIREMENTS
INSTRUMENT AIR SYSTEM

Active Components	Electrical	Support System Coolant (Water)	Air	Other Coolant	Position Normal	Position SI Signal
Air Compressor A	MCC 5262	Service Water	-	-	(Note 1)	Start/Run
Air Compressor B	MCC 62A	Service Water	-	-	(Note 1)	Start/Run
Air Compressor C	MCC 52A	Service Water	-	-	(Note 1)	Start/Run
Air Compressor D	MCC 32B	Service Water	-	-	Standby	Standby
Air Compressor E	MCC 42B	Service Water	-	-	Standby	Standby
Air Compressor F	Bus 35	-	-	-	Running	Running
Turb. Bldg. Bsmt. Fan Coil A	MCC-52A	Service Water	-	-	Running	Running (Note 2)
Turb. Bldg. Bsmt. Fan Coil B	MCC-62E	Service Water	-	-	Running	Running (Note 2)
AOV IA-101		-	To Close	-	Open	Open
AOV IA-121		-	To Close	-	Closed	Closed
AOV SA-200		-	To Open	-	Open	Open
AOV SA-400		-	To Open	-	Open	Open

Note 1: One of three compressors is in running/standby mode.
Note 2: Provides area cooling for MCC-5262.

TAB .2-14

COMPONENT LEVEL SUPPORT REQUIREMENTS
CONTAINMENT AIR COOLING SYSTEM

		Support System			Position	
Active Components	**Electrical**	**Coolant (Water)**	**Air**	**Other Coolant**	**Normal**	**SI Signal**
Fan Coil A	480V Bus 51 125V DC BRA-104	Service Water	-	-	Running	Run
Fan Coil B	480V Bus 51 125V DC BRA-104	Service Water	-	-	Standby (Note 1)	Start/Run
Fan Coil C	480V Bus 61 125V DC BRB-104	Service Water	-	-	Running	Run
Fan Coil D	480V Bus 61 125V DC BRB-104	Service Water	-	-	Running	Run
SW-903A	MCC 52E	-	-	-	Open	Open
SW-903B	MCC 52E	-	-	-	Closed (Note 1)	Open
SW-903C	MCC 62E	-	-	-	Open	Open
SW-903D	MCC 62E	-	-	-	Open	Open
RBV-150A	125V DC BRA-104	-	To Close	-	Closed	Open
RBV-150B	125V DC BRA-104	-	To Close	-	Closed	Open
RBV-150C	125V DC BRB-104	-	To Close	-	Closed	Open
RBV-150D	125V DC BRB-104	-	To Close	-	Closed	Open

Note 1: Assumption for FT Modeling

TABLE 3.2-15

COMPONENT LEVEL SUPPORT REQUIREMENTS MISCELLANEOUS SYSTEM COMPONENTS

Active Components	Electrical	Support System Coolant (Water)	Air	Other Coolant	Position Normal	Position SI Signal
Aux. Basement Fan Coil A	MCC 52E	Service Water	-	-	Running	Start/Run
Aux. Basement Fan Coil B	MCC 62E	Service Water	-	-	Running	Start/Run
Aux. Basement Fan Coil C	MCC 52E	Service Water	-	-	Running	Start/Run
Aux. Basement Fan Coil D	MCC 62E	Service Water	-	-	Running	Start/Run
AOV PR-2A	BRB-104	-	To Open	-	Closed	Closed
AOV PR-2B	BRA-104	-	To Open	-	Closed	Closed
MOV PR-1A	MCC 52B	-	-	-	Open	Open
MOV PR-1B	MCC 62B	-	-	-	Open	Open
AOV SD-3A	BRA-105	-	To Open	-	Closed	Closed
AOV SD-3B	BRB-105	-	To Open	-	Closed	Closed
AOV SD-11A1	BRD-103	-	To Open	-	Closed	Closed
AOV SD-11B1	BRD-103	-	To Open	-	Closed	Closed
AOV MS-1A	BRA-104/BRB-104	-	To Open	-	Open	Open
AOV MS-1B	BRA-104/BRB-104	-	To Open	-	Open	Open
BREAKER 13301	BRC-103	-	-	-	Closed	Closed
BREAKER 14301	BRD-103	-	-	-	Closed	Closed

3.3.1/2 Data Analysis

Plant data was collected and analyzed to support the Kewaunee PRA. The IPE study used this plant specific data along with generic data as necessary to estimate probabilities and other statistical information required to quantify system fault tree and event tree accident sequence models. The main steps in performing data collection and analysis for the plant were (1) collecting the information from plant records and documents, (2) interpreting the information to count the various parameters of concern (e.g., failures, demands, operating hours, initiating events, instances of test or maintenance and its duration), (3) estimating failure rates and test and maintenance frequencies and average duration from the data, and (4) calculating probabilities from these failure rates and test and maintenance parameters.

The final products of the data collection and analysis effort were demand based failure probabilities, operating time based failure rates, test and maintenance unavailabilities, and common cause related unavailabilities as applicable to a specific component/subsystem/system of concern. Additionally, transient initiating event data was collected during this effort for subsequent classification and analysis in accordance with the initiating event analysis as described in section 3.1.1 of this report. Values obtained for the demand based failure probabilities and operating time based failure rates are intended to represent maximum likelihood estimates.

A. Methodology

Before failure probabilities and rates are derived, the basic events for which data is required must be identified. This is an iterative step and a complete list of events was not finalized until the completion of the systems analysis and initiating event analysis. Common cause mechanisms for failure of certain groups of components are considered in the process of collecting data.

All event probabilities in the Kewaunee PRA fault tree models are estimated from plant data. The scope of the data effort included a sampling of all active components in key fluid and electric systems (e.g., pumps, diesel generators, valves that must change position, fans, air compressors, circuit breakers that must change position, etc.) and some "passive" components (e.g., batteries, battery chargers, transformers, inverters, buses, etc.). Failure probability estimates for instrumentation and control components are not derived from plant data, because failures of these components are not consistently reported in plant records unless they lead to the failure of another component in a fluid or electric supply system.

1. Database Selection

Except as otherwise noted, the NUREG/CR-4550 database is selected to evaluate component failure probabilities required as input to fault trees for quantification of the unavailabilities of plant systems.

Each data point calculated in this section is either:

a. Generic
b. Plant-Specific
c. Bayesian Updated (update of generic data by plant-specific data)

2. Hardware Failure Probabilities

Data analysis provides the needed event probabilities and other required statistical information for fault tree and event tree quantification. These include demand based failure probabilities and operating time based failure rates. Values obtained for the demand based failure probabilities and operating time based failure rates are intended to represent maximum likelihood estimates.

For calculating demand based failure probabilities and operating time based failure rates, data for similar components (e.g., all motor-operated valves, air operated valves) are pooled, thereby providing a single failure probability or rate to be applied to all components in the data pool. The purpose of data pooling is to base each estimate on more data. This reduces the effect on the probability estimates of random differences in the failure histories of similar components as well as the effect of potential data inaccuracies or biases.

The data that was collected represents fifteen years of plant operation. The only adjustments that were made in this scope is in cases in which plant improvements or modification had a positive effect on component availability. In these cases the more recent data was used.

In the data collection process, the following served as major sources:

- Maintenance Work Requests (MWR)
- Incident Reports (IR)
- Licensee Event Reports (LER)
- Diesel Generator Reliability Program
- Nuclear Plant Reliability Data System (NPRDS)
- Auxiliary Feedwater PRA Study Database
- Plant Operating Procedures
- Plant Maintenance Procedures
- Plant Surveillance Procedures

3. Test and Maintenance Data Source

This section addresses those test and maintenance actions that cause a component/subsystem/system to be unavailable when required. Testing actions refer to periodic operations or inspections of components/subsystem/systems to verify that they are operable. These acts are performed to satisfy requirements contained in the Kewaunee Technical Specifications. Also, two general categories of maintenance actions must be considered. One is preventive maintenance. This maintenance is scheduled to occur periodically and is intended to ensure that a component operates at peak efficiency. Actions such as oil changes, bearing replacement, filter replacement, etc. are examples of this type of maintenance. Corrective (and hence unscheduled) maintenance is the second type of maintenance action of importance. These actions involve repair or replacement of a component due to an incipient failure, failure during operation, or degraded performance as detected during normal operation or periodic testing. Corrective maintenance actions generally require a longer time to complete than preventive actions.

To account for a component being unavailable due to test or maintenance, the action must take place during power operation

The unavailability of components due to testing, preventive maintenance or corrective maintenance was determined by calculating the frequency of the maintenance activity and the average unavailability duration per maintenance activity. The average unavailability associated with each activity is approximated by the product of the frequency that the activity occurs and the average duration of the activity.

4. Common Cause Model Data Source

After common cause events have been determined and placed in the system models according to the steps outlined in the Fault Tree Guidelines, probability estimates are assigned to each event for fault tree quantification and cutset generation. This requires selection of a common cause probability model, a data analysis to derive parameter estimates for the model, and the evaluation of event probability according to the model and the data.

In the Kewaunee PRA study, the Multiple Greek Letter Method was used and is discussed in Section 3.3.4 of this report. Actual calculations were performed and are documented in the various system notebooks.

5. Human Reliability Data Source

The human reliability analysis (HRA) for the Kewaunee PRA is based on the THERP (Technique for Human Error Rate Prediction) methodology described in Reference 44. The HRA consists of delineating the procedural steps that are necessary for successfully completing the task for a given event, modeling the task in failure configuration, and deducing the probability that the operating crew fails to complete the task. Therefore, failure to complete any (or a combination) of the selected steps for a task results in failure of that task. Details of the HRA approach are presented in section 3.3.3 of this report.

6. Method for Performing Updates

As far as possible, the plant-specific operating experience with respect to a component, i.e., its failures and maintenance outages, has been taken into account. This was done with the aid of the Bayes' theorem which superimposes the plant-specific operating experience on the prior probability distribution to derive a distribution that is biased by the specific experience at the plant. This is called the posterior distribution and represents the data used in this study. Such Bayesian updates are performed when specific component data is available for component failure data and for the component test and maintenance unavailabilities. The Bayesian update calculations are done by using the BAYES3 code of the GRAFTER Code System.

B. Master Data File

In order to make it easier to locate various fault categories or types of data and to provide a logical structure for the file, the database in Table 3.3.1-1 is organized in the following way:

Category	Type of Data in this Category	Fault Numbers (From-To)
A	Logic Switches (Zero and One)	1-2
B	Dummy Probability for Sub-Basic Event	3-5
C	Initiating Event Frequencies	6-33
D	Failure Probabilities for Support System Modules	34-50
E	Fluid System Random Faults	51-275
F	Fluid System and Instrument System Test and Maintenance Unavailabilities	276-375
G	Electrical System Random Faults	376-419
H	Fault Tree and Event Tree Scalars	420-474
I	Electrical System Test and Maintenance Unavailabilities	475-524
J	Common Cause Faults	525-675
K	Human Errors	676-725
L	Other Miscellaneous Faults	726-775
M	Instrument System Random Faults	776-825

This data bank has been organized in line with the fault tree guidelines. This section describes the concept of component tree modules, which have been developed to assure consistency between fault trees and to provide a method of accounting for multiple modes of component failures as well as single modes of failure. Component tree modules completely describe all modes of failure that should be considered in the development and analysis of a component failure.

TABLE 3.3.1-1
MASTER DATA FILE FOR PRA FAULT TREE AND CORE MELT ANALYSIS

	COMP	SYSTEM	FAILURE MODE	FAILRATE	VARIANCE	UNIT	SOURCE
1	XX	ALL	LOGICAL ONE	1.000E+00	0.000E+00	D	N001
2	XX	ALL	LOGICAL ZERO	0.000E+00	0.000E+00	D	N001
3	XX	ALL	DUMMY PROBABILITY FOR SUB-BASIC EVENTS	1.000E-01	0.000E+00	D	N002
4	XX	ALL	DUMMY PROBABILITY FOR SUB-BASIC EVENTS	1.000E-01	0.000E+00	D	N002
5	XX	ALL	DUMMY PROBABILITY FOR IAS SUB-BASIC EVENTS	1.000E-04	0.000E+00	D	N002
11	IF	ASA	LARGE LOCA IE FREQUENCY	5.000E-04	0.000E+00	D	N003
12	IF	ASA	MEDIUM LOCA IE FREQUENCY	2.360E-03	0.000E+00	D	N003
13	IF	ASA	SMALL LOCA IE FREQUENCY	5.120E-03	0.000E+00	D	N003
14	IF	ASA	REACTOR VESSEL FAILURE IE FREQUENCY	3.000E-07	0.000E+00	D	N003
15	IF	ASA	INTERFACING SYSTEMS LOCA IE FREQUENCY	1.480E-06	0.000E+00	D	N003
16	IF	ASA	STEAM GENERATOR TUBE RUPTURE IE FREQUENCY	6.410E-03	0.000E+00	D	N003
17	IF	ASA	TRANSIENTS WITH MAIN FEED WATER IE FREQUENCY	3.000E+00	0.000E+00	D	N003
18	IF	ASA	TRANSIENTS WITHOUT MFW IE FREQUENCY	1.400E-01	0.000E+00	D	N003
19	IF	ASA	LOSS OF 125VDC EMERGENCY BUS IE FREQUENCY	2.350E-03	0.000E+00	D	N003
20	IF	ASA	LOSS OF OFFSITE POWER IE FREQUENCY	4.360E-02	0.000E+00	D	N003
21	IF	ASA	STATION BLACKOUT IE FREQUENCY	4.350E-04	0.000E+00	D	N003
22	IF	ASA	ATWS WITHOUT MAIN FEEDWATER IE FREQUENCY	3.840E-06	0.000E+00	D	N003
23	IF	ASA	LOSS OF CCW SYSTEM IE FREQUENCY	1.620E-03	0.000E+00	D	N003
24	IF	ASA	LOSS OF SW SYSTEM IE FREQUENCY	1.220E-04	0.000E+00	D	N003
25	IF	ASA	LOSS OF INSTRUMENT AIR IE FREQUENCY	1.070E-04	0.000E+00	D	N003
26	IF	ASA	LARGE STEAM/FEED LINE BREAK IE FREQUENCY	2.500E-03	0.000E+00	D	N003
27	XX	SGB	LARGE STEAM LINE BREAK IE FREQUENCY	1.250E-04	1.200E-07	D	N004
28	IF	IFA	FLOODING IE FREQUENCY - AREA FL1	8.900E-05	0.000E+00	D	N023
29	IF	IFA	FLOODING IE FREQUENCY - AREA FL2	1.100E-04	0.000E+00	D	N023
30	IF	IFA	FLOODING IE FREQUENCY - AREA FL3	5.000E-04	0.000E+00	D	N023
31	IF	IFA	FLOODING IE FREQUENCY - AREA FL4	5.000E-04	0.000E+00	D	N023
32	IF	IFA	FLOODING IE FREQUENCY - AREA FL5	1.500E-04	0.000E+00	D	N023
33	IF	IFA	FLOODING IE FREQUENCY - AREA FL6	1.500E-04	0.000E+00	D	N023
51	MV	ALL	MOV FAILS TO OPEN	3.000E-03	5.500E-05	D	4550
52	MV	ALL	MOV FAILS TO CLOSE	3.000E-03	5.500E-05	D	4550
53	MV	ALL	MOV TRANSFERS CLOSED	1.000E-07	5.600E-15	HR	4550
54	MV	ALL	MOV TRANSFERS OPEN	5.000E-07	1.500E-10	HR	4550
55	MV	ALL	MOV PLUGGED	1.000E-07	5.600E-15	HR	4550
56	MV	ALL	VALVE CATASTROPHIC INTERNAL FAILURE	5.000E-07	6.300E-10	HR	IREP
57	AV	ALL	AOV FAILS TO OPEN	2.000E-03	2.200E-06	D	4550
58	AV	ALL	AOV FAILS TO CLOSE	2.000E-03	2.200E-06	D	4550

TABLE 3.3.1-1
MASTER DATA FILE FOR PRA FAULT TREE AND CORE MELT ANALYSIS

	COMP	SYSTEM	FAILURE MODE	FAILRATE	VARIANCE	UNIT	SOURCE
59	AV	ALL	AOV TRANSFERS CLOSED	1.000E-07	5.600E-15	HR	4550
60	AV	ALL	AOV TRANSFERS OPEN	5.000E-07	1.000E-10	HR	4550
61	AV	ALL	AOV PLUGGED	1.000E-07	5.600E-15	HR	4550
62	SV	ALL	SOV FAILS TO OPEN	2.000E-03	2.200E-06	D	4550
63	SV	ALL	SOV FAILS TO CLOSE	2.000E-03	2.200E-06	D	4550
64	SV	ALL	SOV TRANSFERS CLOSED	1.000E-07	5.600E-15	HR	N015
65	SV	ALL	SOV TRANSFERS OPEN	5.000E-07	1.500E-10	HR	N015
66	SV	ALL	SUV PLUGGED	1.000E-07	5.600E-15	HR	4550
67	HV	ALL	HOV FAILS TO OPEN	2.000E-03	2.200E-06	D	4550
68	HV	ALL	HOV FAILS TO CLOSE	2.000E-03	2.200E-06	D	4550
69	HV	ALL	HOV TRANSFERS CLOSED	1.000E-07	5.600E-15	HR	N015
70	HV	ALL	HOV TRANSFERS OPEN	5.000E-07	1.500E-10	HR	N015
71	HV	ALL	HOV PLUGGED	1.000E-07	5.600E-15	HR	4550
72	CV	ALL	CHECK VALVE FAILS TO OPEN	1.000E-04	5.600E-09	D	4550
73	CV	ALL	CHECK VALVE FAILS TO CLOSE	1.000E-03	5.600E-07	D	4550
74	CV	ALL	CHECK VALVE PLUGGED	1.000E-07	5.600E-15	D	N016
75	XV	ALL	MANUAL VALVE FAILS TO OPEN	1.000E-04	5.600E-09	D	IREP
76	XV	ALL	MANUAL VALVE FAILS TO CLOSE	1.000E-04	5.600E-09	D	IREP
77	RV	ALL	PORV FAILS TO CLOSE	2.000E-03	2.200E-06	D	4550
78	UV	ALL	SAFETY VALVE FAILS TO OPEN	1.000E-05	5.600E-11	D	IREP
79	UV	ALL	SAFETY VALVE FAILS TO CLOSE	1.000E-02	5.600E-05	D	IREP
80	RV	ALL	RELIEF VALVE FAILS TO CLOSE	1.600E-02	2.000E-05	D	4550
81	AV	MS	MSIV FAILS TO CLOSE	2.500E-06	3.800E-11	HR	E500
82	PM	CCW	CCW PUMP FAILS TO START	9.980E-03	1.500E-05	D	N014
83	PM	CCW	CCW PUMP FAILS TO RUN	1.370E-05	1.000E-10	HR	N014
84	PM	SW	SW PUMP FAILS TO START	1.370E-02	3.300E-05	D	N014
85	PM	SW	SW PUMP FAILS TO RUN	1.200E-05	8.100E-11	HR	N014
86	PM	ICS	ICS PUMP FAILS TO START	2.130E-02	7.100E-05	D	N014
87	PM	ICS	ICS PUMP FAILS TO RUN	3.000E-05	5.500E-09	HR	N014
88	PM	RHR	RHR PUMP FAILS TO START	1.410E-03	3.400E-06	D	N014
89	PM	RHR	RHR PUMP FAILS TO RUN	1.070E-05	1.600E-10	HR	N014
90	PM	SI	SI PUMP FAILS TO START	1.460E-03	3.900E-06	D	N014
91	PM	SI	SI PUMP FAILS TO RUN	3.000E-05	5.500E-09	HR	N014
92	PM	AFW	MD AFW PUMP FAILS TO START	1.560E-02	2.600E-05	D	N014
93	PM	AFW	MD AFW PUMP FAILS TO RUN	3.000E-05	5.500E-09	HR	N014
94	PM	FW	FW PUMP FAILS TO START	3.000E-03	5.500E-05	D	N014

TABLE 3.3.1-1
MASTER DATA FILE FOR PRA FAULT TREE AND CORE MELT ANALYSIS

	COMP	SYSTEM	FAILURE MODE	FAILRATE	VARIANCE	UNIT	SOURCE
95	PM	FW	FW PUMP FAILS TO RUN	3.590E-06	1.000E-11	HR	N014
96	PM	CD	CD PUMP FAILS TO START	3.000E-03	5.500E-05	D	N014
97	PM	CD	CD PUMP FAILS TO RUN	1.720E-05	6.800E-11	HR	N014
98	PT	AFW	TD AFW PUMP FAILS TO START	1.960E-02	2.200E-04	D	N014
99	PT	AFW	TD AFW PUMP FAILS TO RUN	5.000E-03	1.500E-04	HR	N014
100	PP	SWS	SERVICE WATER PIPE FAILURE	3.400E-09	0.000E+00	HR	N007
101	PM	ALL	MDP FAILS TO START	3.000E-03	5.500E-05	D	4550
102	PM	ALL	MDP FAILS TO RUN	3.000E-05	5.500E-09	HR	4550
103	PT	ALL	TDP FAILS TO START	3.000E-02	5.500E-03	D	4550
104	PT	ALL	TDP FAILS TO RUN	5.000E-03	1.500E-04	HR	4550
105	CV	ALL	CHECK VALVE LEAKAGE	3.000E-05	5.500E-09	HR	4550
106	CV	ALL	CHECK VALVE CATASTROPHIC INTERNAL FAILURE	5.000E-07	6.300E-10	HR	4550
107	RV	ALL	PORV FAILS TO OPEN ON ACTUATION	2.000E-03	2.200E-06	D	4550
108	RV	ALL	PORV FAILS TO OPEN FOR PRESSURE RELIEF	3.000E-04	5.500E-07	D	4550
109	RV	ALL	RELIEF VALVE FAILS TO OPEN	3.000E-04	5.500E-07	D	IREP
110	RV	ALL	RELIEF VALVE TRANSFERS OPEN	3.900E-06	9.300E-11	HR	IREP
111	RV	RCS	PORV FAILS TO OPEN	2.780E-04	3.200E-07	D	N014
112	RV	RCS	PORV FAILS TO CLOSE	1.850E-03	1.700E-06	D	N014
113	MV	RCS	PORV BLOCK VALVE FAILS TO OPEN	1.650E-02	1.900E-04	D	N014
114	MV	RCS	PORV BLOCK VALVE FAILS TO CLOSE	4.140E-02	5.900E-04	D	N014
115	AV	MS	MSIV FAILS TO CLOSE	1.090E-02	7.200E-04	D	N013
116	AV	MS	MSIV FAILS TO CLOSE	8.470E-03	5.600E-05	D	N014
117	HX	RHR	RHR HEAT EXCHANGER SHELL LEAK	2.310E-06	1.600E-11	HR	N014
118	HX	CCW	CCW HEAT EXCHANGER SHELL LEAK	1.140E-06	1.900E-12	HR	N014
119	HX	AFW	AFW PUMP COOLER SHELL LEAK	3.000E-06	5.500E-11	HR	N014
120	MV	ALL	MOV FAILS TO OPEN	7.500E-03	3.200E-05	D	N014
121	MV	ALL	MOV FAILS TO CLOSE	4.760E-03	1.300E-05	D	N014
122	AV	ALL	AOV FAILS TO OPEN	5.320E-03	1.600E-05	D	N014
123	AV	ALL	AOV FAILS TO CLOSE	3.100E-03	5.400E-06	D	N014
124	PM	CVC	CHARGING PUMP FAILS TO START	7.640E-04	6.900E-07	D	N014
125	PM	CVC	CHARGING PUMP FAILS TO RUN	6.170E-05	2.100E-09	HR	N014
126	SY	SWS	IAS FAILURE PROBABILITY (SWA/SWB)	3.000E-06	0.000E+00	D	N007
127	SY	SWS	IAS FAILURE PROBABILITY (SWAG/BG\SWAP/BP)	3.000E-02	0.000E+00	D	N007
128	SY	EPS	DC BUS FAILURE PROBABILITY	3.000E-05	0.000E+00	D	N005
129	UV	MS	RELIEF/SAFETY VALVE FAILS TO CLOSE (SSV)	5.000E-01	0.000E+00	D	N013
188	PP	RCS	PIPE BREAKS >0.7 INCH DIAMETER	5.000E-01	0.000E+00	D	N022

TABLE 3.3.1-1
MASTER DATA FILE FOR PRA FAULT TREE AND CORE MELT ANALYSIS

	COMP	SYSTEM	FAILURE MODE	FAILRATE	VARIANCE	UNIT	SOURCE
189	PP	IAS	INSTRUMENT AIR PIPE FAILURE	6.970E-08	2.700E-15	HR	N013
190	PP	LT3	UP TO 3 INCH DIAM. RUPTURE OR PLUG PER SECTION	8.500E-09	5.100E-15	HR	1400
191	PP	GT3	MORE THAN 3 INCH DIAM. RUPTURE OR PLUG PER SECTION	8.500E-10	5.100E-17	HR	1400
192	HX	ALL	HEAT EXCHANGER BLOCKAGE	5.700E-06	2.000E-10	HR	4550
193	HX	ALL	HEAT EXCHANGER TUBE LEAK PER TUBE	3.000E-09	5.500E-17	HR	IREP
194	HX	ALL	HEAT EXCHANGER SHELL LEAK	3.000E-06	5.500E-11	HR	IREP
195	FL	ALL	STRAINER/FILTER PLUGGED	3.000E-05	5.500E-09	HR	IREP
196	OR	ALL	ORIFICE: FAILURE TO REMAIN OPEN (PLUGGED)	3.000E-04	5.000E-08	D	IREP
197	OR	ALL	ORIFICE: RUPTURE	3.000E-08	5.500E-15	HR	IREP
204	TK	ALL	TANK RUPTURE	8.000E-10	0.000E+00	HR	1400
205	SY	ALL	MULTIPLIER OF TWO FOR FAILURE COMBOS	2.000E+00	0.000E+00	D	----
206	PM	ALL	AIR COMPRESSOR FAILS TO START	8.000E-02	3.600E-03	D	4550
207	PM	ALL	AIR COMPRESSOR FAILS TO RUN	2.000E-04	2.400E-07	HR	4550
208	SY	EPS	DGABO FAILURE PROBABILITY (DGA)	4.090E-03	0.000E+00	D	N005
209	SY	EPS	DGBBO FAILURE PROBABILITY (DGB)	4.830E-03	0.000E+00	D	N005
276	TM	SI	SI PUMP 1B UNAVAILABLE DUE TO T&M	1.440E-03	4.900E-07	D	N014
277	TM	FCU	MOV SW-903B UNAVAILABLE DUE TO T&M	7.510E-04	3.200E-07	D	N014
278	TM	CCW	CCW PUMPS UNAVAILABLE DUE TO T&M	2.370E-03	3.200E-06	D	N014
279	TM	SW	SW PUMPS UNAVAILABLE DUE TO T&M	2.900E-03	4.700E-06	D	N014
280	TM	CCW	CCW HEAT EXCHANGERS UNAVAILABLE DUE TO T&M	1.730E-04	1.700E-08	D	N014
281	TM	RHR	RHR PUMP 1A UNAVAILABLE DUE TO T&M	1.090E-03	6.700E-07	D	N014
282	TM	ICS	ICS PUMP 1A UNAVAILABLE DUE TO T&M	2.900E-03	4.700E-06	D	N014
283	TM	ICS	ICS PUMP 1B UNAVAILABLE DUE TO T&M	1.910E-03	2.000E-06	D	N014
284	TM	FCU	FAN COIL UNITS UNAVAILABLE DUE TO T&M	8.610E-03	4.200E-05	D	N014
285	TM	AFW	AFW PUMP 1A UNAVAILABLE DUE TO T&M	1.340E-03	1.000E-06	D	N014
286	TM	AFW	AFW PUMP 1B UNAVAILABLE DUE TO T&M	1.800E-03	1.800E-06	D	N014
287	TM	AFW	AFW PUMP 1C UNAVAILABLE DUE TO T&M	3.000E-03	5.000E-06	D	N014
289	TM	RHR	RHR PUMP 1B UNAVAILABLE DUE TO T&M	1.170E-03	7.700E-07	D	N014
293	TM	SI	MOV SI-351B UNAVAILABLE DUE TO T&M	8.880E-04	4.400E-07	D	N014
294	TM	CCW	MOV SW-1300B UNAVAILABLE DUE TO T&M	1.290E-04	9.300E-09	D	N014
295	TM	ICS	MOV ICS-5A UNAVAILABLE DUE TO T&M	6.320E-04	2.200E-07	D	N014
296	TM	ICS	MOV ICS-5B UNAVAILABLE DUE TO T&M	5.680E-04	1.800E-07	D	N014
297	TM	ICS	MOV ICS-6A UNAVAILABLE DUE TO T&M	7.830E-04	3.400E-07	D	N014
298	TM	ICS	MOV ICS-6B UNAVAILABLE DUE TO T&M	8.190E-04	3.800E-07	D	N014
299	TM	ICS	MOV ICS-2A UNAVAILABLE DUE TO T&M	3.410E-04	6.500E-08	D	N014
300	TM	ICS	MOV ICS-2B UNAVAILABLE DUE TO T&M	3.410E-04	6.500E-08	D	N014

TABLE 3.3.1-1
MASTER DATA FILE FOR PRA FAULT TREE AND CORE MELT ANALYSIS

	COMP	SYSTEM	FAILURE MODE	FAILRATE	VARIANCE	UNIT	SOURCE
301	TM	ICS	MOV RHR-400A UNAVILABLE DUE TO T&M	1.170E-03	7.700E-07	D	N014
302	TM	ICS	MOV RHR-400B UNAVILABLE DUE TO T&M	9.600E-04	5.200E-07	D	N014
303	TM	ICS	AOV CI-1001A UNAVILABLE DUE TO T&M	1.450E-04	1.200E-08	D	N014
304	TM	ICS	AOV CI-1001B UNAVILABLE DUE TO T&M	7.660E-05	3.300E-09	D	N014
306	TM	SI	MOV SI-4B UNAVAILABLE DUE TO T&M	1.110E-03	6.900E-07	D	N014
308	TM	SI	MOV SI-5B UNAVAILABLE DUE TO T&M	8.200E-04	3.800E-07	D	N014
310	TM	RHR	MOV RHR-300B UNAVAILABLE DUE TO T&M	1.430E-03	1.200E-06	D	N014
311	TM	AFW	MOV AFW-10A UNAVAILABLE DUE TO T&M	2.730E-05	4.200E-10	D	N014
312	TM	AFW	MOV AFW-10B UNAVAILABLE DUE TO T&M	2.730E-05	4.200E-10	D	N014
313	TM	AFW	MOV MS-102 UNAVAILABLE DUE TO T&M	5.400E-04	1.600E-07	D	N014
314	TM	AFW	MOV SW-502 UNAVAILABLE DUE TO T&M	7.870E-04	3.500E-07	D	N014
315	TM	AFW	MOV SW-601A UNAVAILABLE DUE TO T&M	4.680E-04	1.200E-07	D	N014
316	TM	AFW	MOV SW-601B UNAVAILABLE DUE TO T&M	5.050E-04	1.400E-07	D	N014
317	TM	CAC	MOV SW-903B UNAVAILABLE DUE TO T&M	7.510E-04	3.200E-07	D	N014
318	TM	ALL	AIR COMPRESSOR UNAVAILABLE DUE TO T&M	2.000E-03	2.400E-05	D	4550
319	TM	RPS	RPS LOGIC MAINTENANCE UNAVAILABILITY	4.650E-04	0.000E+00	D	N021
320	TM	RPS	SAFECUARDS LOGIC MAINTENANCE UNAVAILABILITY	2.480E-04	0.000E+00	D	N021
321	TM	RPS	REACTOR TRIP BREAKER MAINTENANCE UHAVAILABILITY	8.310E-07	0.000E+00	D	N021
322	TM	RPS	SAFEGUARDS LOGIC TESTING UNAVAILABILITY	2.690E-03	0.000E+00	D	N021
323	TM	RPS	RPS LOGIC TESTING UNAVAILABILITY	2.690E-03	0.000E+00	D	N021
324	TM	RPS	REACTOR TRIP BREAKER TESTING UNAVAILABILITY	2.680E-03	0.000E+00	D	N021
325	TM	RPS	SEQUENCER MAINTENANCE UNAVAILABILITY	4.000E-04	0.000E+00	D	N021
326	TM	RPS	SEQUENCER TEST UNAVAILABILITY	2.690E-03	0.000E+00	D	N021
376	DG	ALL	DG FAILS TO START	3.000E-02	5.000E-04	D	4550
377	DG	ALL	DG FAILS TO RUN	2.000E-03	2.400E-05	HR	4550
378	BS	ALL	DC BUS FAILURE	1.000E-07	1.600E-14	HR	4550
379	BY	ALL	BATTERY FAILURE	1.000E-06	5.600E-13	HR	4550
380	BC	ALL	CHARGER FAILURE	1.000E-06	5.600E-13	HR	4550
381	IV	ALL	INVERTER FAILURE	1.000E-04	5.600E-09	HR	IREP
382	BC	EDC	CHARGER FAILURE	1.000E-06	5.600E-13	HR	N014
383	CB	ALL	CIRCUIT BREAKER FAILS TO TRANSFER (OPEN/CLOSED)	3.000E-03	5.000E-06	D	4550
384	CB	ALL	CIRCUIT BREAKER SPURIOUS OPENING	1.000E-06	6.000E-12	HR	4550
386	BS	ALL	AC BUS FAILURE	1.000E-07	1.600E-14	HR	4550
387	TR	ALL	POWER TRANSFORMER FAILURE	1.000E-06	5.600E-13	HR	IREP
388	BS	IAS	BUS 62 FAILURE	8.200E-05	0.000E+00	D	N013
389	CB	EHV	BUS 5 FEEDER BKRS FAIL TO OPEN (BUS5P)	4.800E-02	0.000E+00	D	N005

TABLE 3.3.1-1
MASTER DATA FILE FOR PRA FAULT TREE AND CORE MELT ANALYSIS

	COMP	SYSTEM	FAILURE MODE	FAILRATE	VARIANCE	UNIT	SOURCE
390	CB	EHV	BUS 6 FEEDER BKRS FAIL TO OPEN (BUS6P)	4.200E-02	0.000E+00	D	N005
391	CB	ALL	CIRCUIT BREAKER FAILS TO OPEN	3.000E-03	5.000E-06	D	4550
392	TR	EHV	AUXILIARY TRANSFORMER FAILURE	1.790E-06	1.300E-12	HR	N014
394	SY	ELV	BUS (32\42\35\45) SUB FAILURE	3.000E-06	0.000E+00	D	N005
395	AS	AFW	VARIABLE FOR AFW-2A CONTROL FAILURE	3.000E-06	5.500E-11	D	N019
397	DG	DGM	DG FAILS TO START	7.720E-03	1.300E-05	D	N014
398	DG	DGM	DG FAILS TO RUN	3.420E-03	6.400E-06	HR	N014
399	IV	EDC	INVERTER FAILURE	5.480E-05	8.800E-10	HR	N014
400	BY	EDC	BATTERY FAILURE	1.000E-06	5.600E-13	HR	N014
402	CN	ALL	LIMIT SWITCH FAILS TO OPERATE	1.000E-04	5.600E-09	D	IREP
403	CN	ALL	PRESSURE SWITCH FAILS TO OPERATE	1.000E-04	5.600E-09	D	IREP
405	AS	AFW	VARIABLE FOR HOTWELL LEVEL CONTROL FAILURE	3.000E-06	5.500E-11	D	N019
408	TL	ALL	LEVEL TRANSMITTER FAILURE	1.000E-06	5.600E-13	HR	4550
409	CN	ALL	RELAY CONTACTS FAIL TO CLOSE	3.000E-04	5.500E-07	D	IREP
410	CN	ALL	TEMPERATURE SWITCH FAILS TO OPERATE	1.000E-04	5.600E-09	D	4550
411	TR	ELV	SERVICE TRANSFORMER FAILURE	8.070E-07	2.800E-13	HR	N014
[illegible]2	TR	EDC	INSTRUMENT TRANSFORMER FAILURE	8.950E-07	3.900E-13	HR	N014
413	AS	MS	SG PORV CONTROL FAILURE	3.000E-06	5.500E-11	D	N019
414	AS	MS	STEAM DUMP CONTROL FAILURE	3.000E-06	5.500E-11	D	N019
420	XX	XX	LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.190E-04	0.000E+00	D	N005
421	XX	XX	BREAK IS UNISOLABLE (UPSTREAM OF MSIV)	2.700E-01	0.000E+00	D	N013
422	XX	XX	POWER NOT RESTORED IN 2 HOURS	2.650E-01	0.000E+00	D	N022
423	XX	XX	POWER NOT RESTORED IN 11 HOURS	2.000E-02	0.000E+00	D	N022
424	XX	XX	POWER NOT RESTORED IN 9 HOURS	4.100E-02	0.000E+00	D	N022
426	XX	XX	REACTOR TRIP BREAKERS FAIL	4.740E-01	0.000E+00	D	N022
427	XX	XX	REACTOR TRIP SIGNAL FAILS	4.460E-01	0.000E+00	D	N022
429	XX	XX	CORE NOT COVERED IN 11 HOURS	7.070E-02	0.000E+00	D	N022
430	XX	XX	CORE NOT COVERED IN 2 HOURS	2.830E-02	0.000E+00	D	N022
431	XX	XX	CORE NOT COVERED IN 9 HOURS	7.620E-02	0.000E+00	D	N022
432	XX	XX	SECONDARY SYSTEM INTEGRITY FAILS	9.840E-01	0.000E+00	D	N022
433	XX	XX	SECONDARY SYSTEM INTEGRITY SUCCESSFUL	1.600E-02	0.000E+00	D	N022
434	IF	XX	BREAKER FAILURE (DBIE)	5.950E-04	2.500E-07	D	N003
435	IF	XX	COMPRESSOR FAILURE (IAIE)	8.260E-01	0.000E+00	D	N003
[illegible]2	XX	XX	RHR PUMP SEAL FAILS	9.950E-01	0.000E+00	D	N022
443	XX	XX	RHR PIPE BREAK	5.000E-03	0.000E+00	D	N022
444	XX	XX	BELOW 10% POWER	4.200E-02	0.000E+00	D	N022

TABLE 3.3.1-1
MASTER DATA FILE FOR PRA FAULT TREE AND CORE MELT ANALYSIS

	COMP	SYSTEM	FAILURE MODE	FAILRATE	VARIANCE	UNIT	SOURCE
445	XX	XX	POWER NOT RESTORED FROM 2-24 HOURS	2.550E-01	0.000E+00	D	N022
446	XX	XX	POWER NOT RESTORED FROM 9-24 HOURS	3.100E-02	0.000E+00	D	N022
447	XX	XX	POWER NOT RESTORED FROM 11-24 HOURS	1.000E-02	0.000E+00	D	N022
448	XX	XX	POWER NOT RESTORED IN 24 HOURS	1.000E-02	0.000E+00	D	N022
449	XX	XX	MECHANICAL FAILURE OF MORE THAN ONE RCCA	7.700E-02	0.000E+00	D	N022
475	TM	DGM	DG 1A UNAVAILABLE DUE TO T&M	1.160E-03	1.300E-06	D	N014
476	TM	DGM	DG 1B UNAVAILABLE DUE TO T&M	1.750E-04	3.060E-08	D	N014
525	GM	CCW	CCW COMMON CAUSE (CCW\CCWP)	2.940E-04	0.000E+00	D	N006
526	CM	CCW	CCW COMMON CAUSE (CCWP)	3.300E-04	0.000E+00	D	N006
527	CM	MFW	MFW COMMON CAUSE (OMO)	1.240E-03	0.000E+00	D	N020
528	CM	SW	SW COMMON CAUSE (SWA\SWB)	6.220E-05	0.000E+00	D	N007
529	CM	SW	SW COMMON GAUSE (SWAP\SWBP)	4.650E-04	0.000E+00	D	N007
530	CM	RHR	LPSI COMMON GAUSE (LI1\LI2\LPI)	4.210E-04	0.000E+00	D	N011
531	CM	DG	EPS COMMON CAUSE (DGA\DGB)	1.970E-03	0.000E+00	D	N005
532	CM	RHR	RHR COMMON CAUSE (RHR\RHRA/B\LR1/2/3/4)	9.790E-04	0.000E+00	D	N011
533	CM	MSI	MSI COMMON CAUSE (ISO)	7.060E-04	0.000E+00	D	N013
77	CM	MSI	MSI\MFI COMMON CAUSE (IS1)	7.060E-04	0.000E+00	D	N013
535	CM	SW	SW COMMON CAUSE (SWAG/SWBG)	7.300E-04	0.000E+00	D	N007
536	CM	SI	SIS COMMON CAUSE (HPI\HPID\HIO/1/2/3/4)	7.560E-04	0.000E+00	D	N012
537	CM	SI	SIS COMMON CAUSE (HPR\HRO/1/2)	8.140E-04	0.000E+00	D	N012
538	CM	MS	MSIVS FAIL DUE TO COMMON CAUSE	1.090E-03	0.000E+00	D	N013
539	CM	AFW	AFW COMMON CAUSE (AF6)	1.060E-04	0.000E+00	D	N010
540	CM	MFW	MFW COMMON CAUSE (OM2/3)	1.040E-04	0.000E+00	D	N020
541	CM	MFW	MFW COMMON CAUSE (OM1)	1.130E-03	0.000E+00	D	N020
542	CM	AFW	AFW COMMON CAUSE (AF0/3/4/5/G)	2.120E-04	0.000E+00	D	N010
543	CM	MS	MSIVS FAIL (OC) DUE TO COMMON CAUSE	9.600E-04	0.000E+00	D	N004
544	CM	MS	MSIVS FAIL (IC) DUE TO COMMON CAUSE	9.600E-07	0.000E+00	D	N004
545	CM	ICS	ICS COMMON CAUSE (CSI/ICS/CSD)	1.570E-03	0.000E+00	D	N009
546	CM	ICS	ICS COMMON CAUSE (CSR)/ICS/CSD)	1.750E-03	0.000E+00	D	N009
547	CM	CVC	CVCS COMMON CAUSE (LTS)	1.180E-04	0.000E+00	D	N013
548	CM	CAC	CAC COMMON CAUSE (FCH/FCHP/FCD)	5.950E-04	0.000E+00	D	N008
549	CM	AFW	AFW COMMON CAUSE (AF1)	4.100E-04	0.000E+00	D	N010
550	CM	MS	MS COMMON CAUSE (OCD)	2.630E-04	0.000E+00	D	N013
551	CM	CVC	CVCS COMMON CAUSE (CHG)	2.160E-06	0.000E+00	D	N013
552	CM	AFW	AFW COMMON CAUSE (AF2)	2.590E-04	0.000E+00	D	N010
553	CM	RCS	RCS COMMON CAUSE (PPR)	1.820E-04	0.000E+00	D	N013

TABLE 3.3.1-1
MASTER DATA FILE FOR PRA FAULT TREE AND CORE MELT ANALYSIS

	CCMP	SYSTEM	FAILURE MODE	FAILRATE	VARIANCE	UNIT	SOURCE
554	CM	MS	MS COMMON CAUSE (OP1\OP2)	4.520E-05	0.000E+00	D	N013
555	CM	SW	SWS COMMON CAUSE (SWIE)	1.210E-04	0.000E+00	D	N003
556	CM	IAS	IA COMMON CAUSE (IAS/IAIE)	1.730E-08	0.000E+00	D	N003
557	CM	ACA	ACA COMMON CAUSE (ABBC)	2.660E-04	0.000E+00	D	N013
558	CM	RPS	COMMON CAUSE FAILURE OF 2 REACTOR TRIP BREAKERS	1.110E-05	0.000E+00	D	N021
559	CM	RPS	CCF OF 2 COMPARATORS (M=3)	7.770E-08	0.000E+00	HR	N021
560	CM	RPS	CCF OF 3 COMPARATORS (M=3)	1.200E-07	0.000E+00	HR	N021
561	CM	RPS	CCF OF 2 COMPARATORS (M=4)	5.180E-08	0.000E+00	HR	N021
562	CM	RPS	CCF OF 3 COMPARATORS (M=4)	1.220E-08	0.000E+00	HR	N021
563	CM	RPS	CCF OF 4 COMPARATORS (M=4)	3.980E-08	0.000E+00	HR	N021
564	CM	RPS	CCF OF 2 LEAD/LAG CONTROLLERS (M=4)	1.390E-08	0.000E+00	HR	N021
565	CM	RPS	CCF OF 3 LEAD/LAG CONTROLLERS (M=4)	3.290E-09	0.000E+00	HR	N021
566	CM	RPS	CCF OF 4 LEAD/LAG CONTROLLERS (M=4)	1.070E-08	0.000E+00	HR	N021
567	CM	RPS	CCF OF 2 RELAYS - MECHANICALLY BOUND (M=4)	7.140E-09	0.000E+00	HR	N021
568	CM	RPS	CCF OF 3 RELAYS - MECHANICALLY BOUND (M=4)	1.680E-09	0.000E+00	HR	N021
569	CM	RPS	CCF OF 4 RELAYS - MECHANICALLY BOUND (M=4)	5.490E-09	0.000E+00	HR	N021
5[illegible]0	CM	RPS	CCF OF 2 RELAYS-RELAY FAILS DURING OPERATION (M=4)	9.110E-09	0.000E+00	HR	N021
571	CM	RPS	CCF OF 3 RELAYS-RELAY FAILS DURING OPERATION (M=4)	2.150E-09	0.000E+00	HR	N021
572	CM	RPS	CCF OF 4 RELAYS-RELAY FAILS DURING OPERATION (M=4)	7.000E-09	0.000E+00	HR	N021
573	CM	RPS	CCF OF RX TRIP TRAINS A&B (NONDIVERSE, NO MANUAL C IRCUITRY)	6.790E-05	0.000E+00	D	N021
574	CM	RPS	CCF OF RX TRIP TRAINS A&B (NONDIVERSE, WITH MANUAL CIRCUITRY)	1.100E-05	0.000E+00	D	N021
575	CM	RPS	CCF OF RX TRIP TRAINS A&B (DIVERSE, NO MANUAL CIRC UITRY)	1.300E-05	0.000E+00	D	N021
576	CM	RPS	CCF OF RX TRIP TRAINS A&B (DIVERSE, WITH MANUAL CI RCUITRY)	1.100E-05	0.000E+00	D	N021
577	CM	RPS	CCF OF ESFAS TRAINS A&B (NO MANUAL CIRCUITRY)	1.510E-04	0.000E+00	D	N021
578	CM	RPS	CCF OF ESFAS TRAINS A&B (WITH MANUAL CIRCUITRY)	1.350E-05	0.000E+00	D	N021
579	CM	RPS	CCF OF 2 SOLID STATE TIME DELAY RELAYS	9.360E-08	0.000E+00	HR	N021
580	CM	ANS	CCF OF 2 OF 4 AMSAC S/G LEVEL CIRCUITS	6.450E-05	0.000E+00	D	N021
581	CM	RPS	CCF OF 2 RELAYS - RELAY FAILS DURING OPERATION (M=2)	4.080E-08	0.000E+00	HR	N021
582	GM	RPS	CCF OF SEQUENCER AUTO INHIBIT COMPONENTS	1.110E-04	0.000E+00	D	N021
583	CM	RPS	CCF OF CS TRAINS A AND B (NO MANUAL CIRCUITRY)	2.830E-04	0.000E+00	D	N021
584	CM	RPS	CCF OF AFW PUMP SLAVE RELAYS	3.680E-08	0.000E+00	HR	N021
5[illegible]5	CM	RPS	CCF OF SEQUENCER ON BO	1.760E-04	0.000E+00	D	N021
586	CM	RPS	CCF OF RHR PUMP PRESSURE ACTUATION CIRCUITS FOR RH R-300A/B	8.900E-05	0.000E+00	D	N021

TABLE 3.3.1-1
MASTER DATA FILE FOR PRA FAULT TREE AND CORE MELT ANALYSIS

	COMP	SYSTEM	FAILURE MODE	FAILRATE	VARIANCE	UNIT	SOURCE
587	CM	RPS	CCF OF CS TRAINS A&B,WITH MANUAL CIRCUITRY	1.510E-06	0.000E+00	D	N021
588	CM	RPS	CCF OF BAT TO RWST AUTO SWITCHOVER CIRCUITRY	1.630E-03	0.000E+00	D	N021
589	CM	RPS	CCF OF SLAVE RELAYS FOR MS-102 AUTO OPEN CIRCUITRY	5.200E-08	0.000E+00	HR	N021
590	CM	CI	CI COMMON CAUSE - PEN 5 (CI)	2.640E-04	0.000E+00	D	N018
591	CM	CI	CI COMMON CAUSE - PEN 26 (CI)	9.600E-07	0.000E+00	D	N018
592	CM	CI	CI COMMON GAUSE - PEN 14 (CI)	1.870E-04	0.000E+00	D	N018
593	CM	CI	CI COMMON CAUSE - PEN 28N (CI)	2.680E-05	0.000E+00	D	N018
594	CM	CI	CI COMMON CAUSE - PEN 13N (CI)	6.000E-05	0.000E+00	D	N018
595	CM	CI	CI COMMON CAUSE - PEN 13E (CI)	6.000E-05	0.000E+00	D	N018
596	CM	CI	CI COMMON CAUSE - PEN 29N (CI)	2.640E-07	0.000E+00	D	N018
597	CM	CI	CI COMMON CAUSE - PEN 29E (CI)	2.640E-07	0.000E+00	D	N018
598	CM	RCS	B&F COMMON CAUSE (OB1/OB2/OB3/OB5)	1.820E-04	0.000E+00	D	N013
599	CM	RCS	B&F COMMON CAUSE (OB4)	4.770E-04	0.000E+00	D	N013
600	CM	RCS	C&D COMMON CAUSE (EC3)	4.520E-05	0.000E+00	D	N013
601	CM	RCS	C&D COMMON CAUSE (ES1)	9.920E-04	0.000E+00	D	N013
602	CM	EPS	COMMON CAUSE (TIB)	2.400E-06	0.000E+00	D	----
603	CM	IAS	COMMON CAUSE (IASP)	2.440E-03	0.000E+00	D	N013
604	CM	RCS	C&D COMMON CAUSE (OS1/OS2/EC4)	8.710E-05	0.000E+00	D	N013
605	CM	CI	CI COMMON CAUSE - PEN 10 (CI/CID)	6.000E-05	0.000E+00	D	N018
606	CM	CI	CI COMMON CAUSE - PEN 12 (CI/CID)	2.680E-05	0.000E+00	D	N018
607	CM	CI	CI COMMON CAUSE - PEN 48 (CI/CID)	6.000E-05	0.000E+00	D	N018
608	CM	RCS	COMMON CAUSE (OSD)	4.240E-08	0.000E+00	D	N013
609	CM	RPS	COMMON CAUSE (AFM)	1.040E-05	0.000E+00	D	N021
676	HE	ALL	HUMAN ERROR OF VALVE MISPOSITION AFTER T&M	1.000E-05	0.000E+00	D	----
677	HE	ALL	DUMMY VARIABLE FOR HUMAN ERROR	1.000E-02	0.000E+00	D	----
678	HE	HRA	HUMAN ERROR (23IXV-ICS7AB-HE)	4.030E-05	0.000E+00	D	N017
679	HE	HRA	HUMAN ERROR (23R-RWST-RHR-HE)	5.860E-04	0.000E+00	D	N017
680	HE	HRA	HUMAN ERROR (36--OSD------HE)	7.960E-04	0.000E+00	D	N017
681	HE	HRA	HUMAN ERROR (33R-1TRN-REC-HE)	3.530E-04	0.000E+00	D	N017
682	HE	HRA	HUMAN ERROR (36-OB1-------HE)	9.810E-05	0.000E+00	D	N017
683	HE	HRA	HUMAN ERROR (33IXV--SI7AB-HE)	4.030E-05	0.000E+00	D	N017
684	HE	HRA	HUMAN ERROR (05B-CST-SWS--HE)	7.400E-04	0.000E+00	D	N017
685	HE	HRA	HUMAN ERROR (05BAV-MU3A---HE)	3.680E-02	0.000E+00	D	N017
686	HE	HRA	HUMAN ERROR (05B-MIAFW----HE)	5.450E-05	0.000E+00	D	N017
688	HE	HRA	HUMAN ERROR (31-LO-SW1300-HE)	2.200E-04	0.000E+00	D	N017
689	HE	HRA	HUMAN ERROR (36--EC3------HE)	3.990E-04	0.000E+00	D	N017

TABLE 3.3.1-1
MASTER DATA FILE FOR PRA FAULT TREE AND CORE MELT ANALYSIS

	COMP	SYSTEM	FAILURE MODE	FAILRATE	VARIANCE	UNIT	SOURCE
690	HE	HRA	HUMAN ERROR (36--ES1------HE)	4.990E-04	0.000E+00	D	N017
691	HE	HRA	HUMAN ERROR (06--ISO------HE)	9.400E-04	0.000E+00	D	N017
692	HE	HRA	HUMAN ERROR (36-OB4-------HE)	9.810E-05	0.000E+00	D	N017
693	HE	HRA	HUMAN ERROR (36-OB2-OB5---HE)	3.880E-04	0.000E+00	D	N017
694	HE	HRA	HUMAN ERROR (36-OB3-------HE)	5.900E-04	0.000E+00	D	N017
695	HE	HRA	HUMAN ERROR (05A--OM0-----HE)	1.930E-03	0.000E+00	D	N017
696	HE	HRA	HUMAN ERROR (05A--OM1-----HE)	3.300E-03	0.000E+00	D	N017
697	HE	HRA	HUMAN ERROR (05A--OM2-OM4-HE)	1.190E-04	0.000E+00	D	N017
698	HE	HRA	HUMAN ERROR (05A--OM3-----HE)	1.920E-02	0.000E+00	D	N017
699	HE	HRA	HUMAN ERROR (36--OP1------HE)	1.700E-03	0.000E+00	D	N017
700	HE	HRA	HUMAN ERROR (36--OP2------HE)	8.820E-04	0.000E+00	D	N017
701	HE	HRA	HUMAN ERROR (40--ORT------HE)	2.650E-02	0.000E+00	D	N017
702	HE	HRA	HUMAN ERROR (36--OS1------HE)	9.800E-03	0.000E+00	D	N017
703	HE	HRA	HUMAN ERROR (36--OS2------HE)	5.000E-02	0.000E+00	D	N017
704	HE	HRA	HUMAN ERROR (47--MRT------HE)	2.030E-03	0.000E+00	D	N017
705	HE	HRA	HUMAN ERROR (42-DMS-------HE)	4.340E-05	0.000E+00	D	N017
707	HE	HRA	HUMAN ERROR (35--LTS------HE)	9.110E-05	0.000E+00	D	N017
708	HE	HRA	HUMAN ERROR (36--OCD------HE)	3.350E-03	0.000E+00	D	N017
709	HE	HRA	HUMAN ERROR (36--ORI------HE)	9.960E-05	0.000E+00	D	N017
710	HE	HRA	HUMAN ERROR (36-RCS-DEP---HE)	5.770E-04	0.000E+00	D	N017
711	HE	HRA	HUMAN ERROR (56-CI-CAT-B--HE)	1.850E-04	0.000E+00	D	N017
712	HE	HRA	HUMAN ERROR (34I---LI2A---HE)	4.230E-04	0.000E+00	D	N017
713	HE	HRA	HUMAN ERROR (34I---LI2----HE)	6.110E-05	0.000E+00	D	N017
714	HE	HRA	HUMAN ERROR (23I-MAN-ICS--HE)	2.580E-03	0.000E+00	D	N017
715	HE	HRA	HUMAN ERROR (02----SWT----HE)	4.350E-05	0.000E+00	D	N017
716	HE	HRA	HUMAN ERROR (33-OIP-------HE)	1.400E-02	0.000E+00	D	N017
717	HE	HRA	HUMAN ERROR (33-MAN-SI-IN-HE)	2.550E-03	0.000E+00	D	N017
718	HE	HRA	HUMAN ERROR (33-OSR-------HE)	3.320E-02	0.000E+00	D	N017
719	HE	HRA	HUMAN ERROR (33R-2TRN-REC-HE)	4.920E-05	0.000E+00	D	N017
720	HE	HRA	HUMAN ERROR (36--EC4------HE)	5.000E-02	0.000E+00	D	N017
721	HE	HRA	HUMAN ERROR (36-RXCP-STOP-HE)	2.330E-03	0.000E+00	D	N017
726	DM	ALL	DAMPER FAILS TO OPERATE	3.000E-03	5.500E-05	D	4550
727	FN	ALL	AIR COOLERS FAIL TO RUN	1.000E-05	5.600E-11	HR	4550
728	FN	ALL	AIR COOLERS FAIL TO START	3.000E-04	5.000E-08	D	4550
729	FN	RBV	CONTAINMENT FAN COIL UNIT FAILS TO START	7.090E-04	2.300E-07	D	N014
730	FN	RBV	CONTAINMENT FAN COIL UNIT FAILS TO RUN	6.300E-06	9.900E-12	HR	N014

TABLE 3.3.1-1
MASTER DATA FILE FOR PRA FAULT TREE AND CORE MELT ANALYSIS

	COMP	SYSTEM	FAILURE MODE	FAILRATE	VARIANCE	UNIT	SOURCE
731	SY	CI	CATEGORY A PENETRATION FAILURE	1.200E-04	0.000E+00	D	N018
776	SY	RPS	1-TRAIN REPRESENTATIVE ESFAS SIGNAL FAILS (WITH MA NUAL CIRCUITRY)	3.380E-03	0.000E+00	D	N021
777	SY	RPS	2-TRAIN REPRESEHTATIVE ESFAS SIGNAL FAILS (WITH MA NUAL CIRCUITRY)	4.590E-06	0.000E+00	D	N021
778	SY	RPS	1-TRAIN REPRESENTATIVE ESFAS SIGNAL FAILS (HO MANU AL CIRCUITRY)	3.840E-03	0.000E+00	D	N021
779	SY	RPS	2-TRAIN REPRESENTATIVE ESFAS SIGNAL FAILS (NO MANU AL CIRCUITRY)	2.880E-05	0.000E+00	D	N021
780	SY	RPS	1-TRAIN REPRESENTATIVE CS SIGNAL FAILS (NO MANUAL CIRCUITRY)	3.310E-03	0.000E+00	D	N021
781	SY	RPS	2-TRAIN REPRESENTATIVE CS SIGNAL FAILS (NO MANUAL CIRCUITRY)	5.560E-05	0.000E+00	D	N021
782	CN	RPS	RELAY CONTACTS FAIL TO OPEN/CLOSE	8.500E-06	0.000E+00	D	TOPS
783	RE	RPS	RELAY FAILS DURING OPERATION	5.100E-07	0.000E+00	HR	TOPS
784	CN	RPS	RELAY CONTACTS SPURIOUSLY OPEN	8.700E-08	0.000E+00	HR	TOPS
785	RE	RPS	RELAY MECHANICALLY BOUND	4.000E-07	0.000E+00	HR	TOPS
786	DC	RPS	LOOP POWER SUPPLY FAILS	5.800E-06	0.000E+00	HR	TOPS
787	SW	RPS	SWITCH (PUSHBUTTON-TOGGLE) FAILS	3.600E-09	0.000E+00	HR	TOPS
[illegible]8	TP	RPS	PRESSURE SENSOR FAILS	2.800E-06	0.000E+00	HR	TOPS
789	AD	RPS	COMPARATOR FAILS	2.900E-06	0.000E+00	HR	TOPS
790	TL	RPS	LEVEL SENSOR FAILS	4.900E-06	0.000E+00	HR	TOPS
791	AM	RPS	LEAD/LAG AMPLIFIER FAILS	7.800E-07	0.000E+00	HR	TOPS
792	CB	RPS	REACTOR TRIP BREAKER FAILS TO OPEN	6.890E-05	0.000E+00	D	TOPS
793	CB	RPS	BYPASS BREAKER FAILS TO OPEN	3.490E-04	0.000E+00	D	TOPS
794	RE	RPS	NON-SOLID STATE TIME DELAY RELAY FAILS	6.700E-07	0.000E+00	HR	E500
795	SD	RPS	SOLID STATE TIME DELAY RELAY FAILS (100-199 VDC)	1.170E-06	0.000E+00	HR	E500
796	FU	RPS	FUSE FAILS	2.300E-07	0.000E+00	HR	TOPS
797	SD	AMS	AMSAC PROCESSOR (PLC) FAILS	1.300E-06	0.000E+00	HR	E500
798	SD	AMS	AMSAC POWER SUPPLY FAILS	1.610E-06	0.000E+00	HR	E500
799	SD	AMS	AMSAC AC ISOLATED OUTPUT CARD FAILS	5.600E-07	0.000E+00	HR	E500
800	SD	AMS	AMSAC ISOLATED ANALOG INPUT CARD FAILS	6.800E-07	0.000E+00	HR	E500
801	AM	AMS	AMSAC I/I ISOLATION AMPLIFIER FAILS	3.820E-07	0.000E+00	HR	E500
802	FA	RPS	MECHANICAL FAILURE OF MORE THAN 1 RCCAS	1.800E-06	0.000E+00	D	N021

TES

1. NOTES / REFERENCES:

N001 = PROBABILITIES OF 1 AND 0 ARE PLACED IN THE DATA BANK TO BE USED AS NEEDED

N002 = FOR SUPPORT SYSTEM SUB TREES THAT APPEAR AS BASIC EVENTS, THIS TEMPORARY DUMMY PROBABILITY IS ASSIGNED.

N003 = CALCULATED IN IE NOTEBOOK,EF=3 FOR VARIENCES.

N004 = CALCULATED IN SGB FAULT TREE FILE.

N005 = CALCULATED IN EPS NOTEBOOK.

N006 = CALCULATED IN CCW NOTEBOOK.

N007 = CALCULATED IN SWS NOTEBOOK.

N008 = CALCULATED IN CAC NOTEBOOK.

N009 = CALCULATED IN ICS NOTEBOOK.

N010 = CALCULATED IN AFW NOTEBOOK.

N011 = CALCULATED IN LPSI NOTEBOOK.

N012 = CALCULATED IN HPSI NOTEBOOK.

N013 = CALCULATED IN MISC SYSTEM NOTEBOOK.

N014 = CALCULATED IN DATA ANALYSIS NOTEBOOK.

N015 = PROBABILITY FOR AOV FROM 4550 USED HERE.

N016 = PROBABILITY FOR MANUAL VALVE FROM 4550 USED HERE.

N017 = CALCULATED IN HRA NOTEBOOK.

N018 = CALCULATED IN CI NOTEBOOK.

N019 = PROBABILITY FOR INSTRUMENTATION FROM 4550 USED HERE.

N020 = CALCULATED IN MFW NOTEBOOK.

N021 = CALCULATED IN RPS NOTEBOOK.

N022 = CALCULATED IN COREMELT NOTEBOOK.

N023 = CALCULATED IN FLOODING NOTEBOOK.

RZZ = VERIFY DATA SOURCE AND VALUES.

4550 = REFERS TO NUREG/CR-4550 REV 1.

IREP = REFERS TO NUREG/CR-2728.

1400 = REFERS TO WASH - 1400, NUREG 75/014.

E500 = REFERS TO IEEE 500.

TOPS = REFERS TO WCAP-10271,"EVALUATION OF SURVEILLANCE FREQUENCIES AND OUT OF SERVICE TIMES FOR THE REACTOR PROTECTION SYSTEM". AND SUPPLEMENTS 2&3, WESTINGHOUSE PROPIETARY CLASS 2.

TABLE 3.3.1-1
MASTER DATA FILE FOR PRA FAULT TREE AND CORE MELT ANALYSIS

ABBREVIATIONS USED:

XX = ITEM RELATED NOT TO A COMPONENT

ALL = ITEM APPLIES TO ALL SYSTEMS

IE = INITIATING EVENT

D(2) = DEMAND FAILURE

HR(1) = HOURLY FAILURE RATE (LAMBDA), NEEDS TO BE MULTIPLIED BY A DELTA-T

SG = STEAM GENERATOR

SUB- = FIRST FOUR DIGITS OF THE BASIC EVENT ID OF A SUBTREE

ATWS = ANTICIPATED TRANSIENTS WITHOUT SCRAM

LT3 = LESS THAN OR EQUAL TO 3 INCH DIAMETER PIPE

GT3 = GREATER THAN 3 INCH DIAMETER PIPE

RZZ- = DATA LINE REQUIRES RESOLUTION

3. COMMENTS:

E DATA BANK ALSO CONTAINS GENERIC DATA FOR COMPONENTS; AND DUMMY VALUES FOR HUMAN ERROR

THE VARIANCES ARE CALCULATED AS EXPLAINED IN SECTION 4.3

3.3.3 Human Failure Data

The human reliability analysis (HRA) for the Kewaunee PRA is based on the THERP (Technique for Human Error Rate Prediction) methodology described in Reference 44.

In applying the THERP methodology, modifications to THERP base human error probabilities (BHEPs) can be made logically to account for changes in nuclear power plants operating philosophy, that came about after THERP basic human error probabilities (BHEPs) were developed. Such changes surrounding the development of generic Westinghouse Emergency Response Guidelines are outlined as follows:

a) development of PWR plant specific symptom-based procedures
b) training on the use of these symptom-based procedures
c) usage of these symptom-based procedures in actual plant emergencies

It was decided that modifications should be made to the base HEPs for commission errors, diagnosis (alarm response) errors, and control room recovery errors to account for the advantages or enhancements in using the symptom-based procedures, given the type of the event and timing considerations. The method of adjusting the THERP BHEPs is discussed in section 3.3.3.1.

The HRA consists of delineating the procedural steps necessary for successfully completing the task for a given event, modeling the task in failure configuration, and deducing the probability that the operating crew will fail to complete the task. Therefore, failure to complete any (or a combination) of the selected steps for a task results in failure of that task. The HRA subtasks are described in section 3.3.3.2. The results of the HRA quantification are provided in Table 3.3.3-2.

3.3.3.1 Approach

Top events identified in the event tree analysis that contain specific operator actions for the event's success criterion, are selected for inclusion in the HRA. Operator actions identified by the fault tree analysts pertaining to system alignment are also included in this analysis. The current Kewaunee emergency operating procedures, and normal or abnormal operating procedures are used as bases and sources for selecting the subtasks of the various human actions. In that regard, each subtask is selected on the basis that failure of the operator to perform that specific step significantly impacts or results in failure of a safety system. Therefore, most steps that are follow-throughs from or to some operator actions and that do not involve some physical operator activity are screened out. This typically involves steps for checking system parameters while doing other major (physical) steps. The assumption used here is that if the plant response is normal or as expected for the accident scenario, then such system parameter responses are attained. Moreover, such checking activities have diagnosis implications and it becomes quite cumbersome to model the numerous amount of these steps. However, such checking steps are taken into account during the quantification activity in which they are represented as performance shaping factors (PSFs) to the modeled tasks, as applicable. It should be noted that during quantification, the subtasks are ORed (summed) or ANDed (multiplied), as necessary, in order to deduce the estimated human error probability (HEP) of each particular task.

In general, subtasks are selected from steps along the main success path in the procedures, designated by "EXPECTED RESPONSE". This approach was chosen since the procedures contain a very large number of subtask steps, many of which are branched to alternate success paths (designated by "RESPONSE NOT OBTAINED"), and it is not practical to model such alternate paths. Moreover, it is conservative not to model the alternate success paths because the failure probabilities of such additional paths would be multiplied by the random component failure probabilities along the main path, thus lowering the derived HEP for the task.

A. Major Assumptions

1. Operators are highly skilled in performing the necessary tasks; each having more than 6 months experience. In most cases, normal (low to medium level) stress is applied due to the level of experience, the nature of the event and lack of being unduly challenged in performing the proceduralized tasks. It is believed that the operator will experience high stress during a steam generator tube rupture, LOCA, loss of all AC power accident and other specialized situations discussed in the appropriate section. During quantification, the three different stress levels are applied to the respective nodes as performance shaping factors, which are multiplied by the nominal HEPs of the subtasks.

2. Control room indication is provided for equipment status, with visual and audible alarm indications of equipment failures or parameter deviations.

3. Visual and audible alarms demand (or serve as prompts for) initial operator response. Loss of component cooling water and loss of service water events are diagnosed within the respective abnormal operating procedures. For any other abnormal plant condition resulting in a reactor trip or the need for reactor trip, the operators' activities begin with the proceduralized steps in E-0 within which diagnosis of the event is conducted. In other words, the operators are not led from the alarm indications directly to diagnosis of the event without going through the procedure. Therefore, it is of utmost importance that the operators respond to the initial alarm(s) and then use the applicable procedures to arrive at the specific diagnosis activities.

   THERP defines diagnosis as having three components, namely detection + diagnosis + decision. The THERP definition is applicable to knowledge-based responses, in which the operators go through more thought-process (deciphering) in order to diagnose an event. The new generic procedures are based on the philosophy of symptomatic responses to an emergency operating situation, and therefore, reduce the diagnosis of an event to responding to cues such as alarms, annunciators, indicators (detection); thus avoiding the cognitive aspects (diagnosis + decision). Therefore, it is advisable not to use Table 20-3 of the THERP Handbook[44] or similar models for actions governed by symptom-based procedures in which the operators are trained; such activities are termed rule-based actions.

4. It is assumed that each operator is responsible for completing specific tasks. However, in addition to the control room supervisor, primary systems operator and balance of plant operator, there is the shift supervisor and shift technical advisor (STA) on the operating team. The primary systems operator and balance of plant (BOP) operator are conversant with the operations and controls in the entire control room; each is assigned one position for a shift, but can be rotated to the other position on a different shift. It is believed that an awareness checking is carried out by one operator and shift supervisor or STA during an abnormal event, that could recover errors made by the other operator; this checking is classified in this analysis as unproceduralized checking or recovery, and is applied in general to all tasks.

   The formulation of unproceduralized checking is as follows:

   a) the operator on the control board may recover his own error; or the operator who is reading the procedure may recover an error made by the operator on the control board.

      (This recovery is assigned 8.0E-02, described by THERP as "one-of-a-kind checking with alert factors").

   b) by assuming that the STA arrives at the control room within 10 minutes after an initiating event, credit is taken for the status tree monitoring activities which could recover failures. A multiplicative factor of 0.1 is assigned for this recovery by the STA.

   The basis for the 0.1 assigned for the STA is as follows: the recovery function of the STA is more or less independent of the function of the control room operators. However, we evaluate the STA's recovery as having a low dependency on recovery by the control room operators, described above. The low dependency calculation is obtained by applying the equation from THERP, Table 20-17: $(1 + 19n)/20$; where $n = 8.0E-02$ (which happens to have the same value as recovery described in (a) above). This recovery is equated to 1.26E-01 which is rounded to 1.0E-01.

   Therefore, unproceduralized recovery is represented by 8.0E-02 x 0.1; (i.e., item (a) multiplied by item (b)). Unproceduralized recovery is addressed further under item 9.

5. For local actions, a checking recovery of 1.6E-01 is applied. This BHEP was selected from the THERP BHEP for "checking routine tasks". Although the THERP definition for checking is not exactly the same as the definition meant for this kind of recovery, the error rate of 1.6E-01 is suitable to represent this recovery. This local recovery is based on the existence of radio communication between control room and auxiliary operators, and also, in some cases, on the existence of control room indication of the status of equipment being locally manipulated.

   The recovery factor applied for STA discussed in item 4 is not applied to local actions.

6. In order to apply the benefit in using the symptomatic procedures and also take into account that there are 3 operators responding to the alarms, it is logical to reduce the THERP BHEPs for "failure to respond to alarms". In that regard, a moderate dependency is applied to the "alarm response failure" BHEP which, in most cases, falls between the values of 8.0E-03 and 2.7E-04; this dependency value is rounded to 1.0E-01.

   Therefore, for an initiating event with a time window greater than 5 minutes and having the appropriate slack time, the diagnosis failure of responding to alarms is evaluated by applying the THERP BHEP multiplied by 0.1.

   It should be mentioned that the subtasks for diagnosis of an event also include the performance of essential proceduralized verification steps. If such steps are assessed as being critical to the success of the overall task, they are modeled as omission and/or commission errors.

7. Based on engineering judgement, no PSF that reduces the nominal HEP is given to operator experience for events that require completion of diagnosis and corrective action within 5 minutes. For instance the unproceduralized checking discussed in item 4 and the symptomatic procedure credit in item 6 are not applicable to these events. It is assumed that such events are fairly complex.

8. If the operators have more time than the average amount of time it takes to complete an action, then it is assumed that the operator's performance in diagnosis and action is not time dependent, since the operator has to follow the applicable procedure(s), and operator does not have a physical time clock running during an abnormal operating condition. However, recovery within that event may be time dependent.

9. If the available time window is greater than the average time for completing an action, then slack time is available; slack time being "the time window" minus "the actual time".

It is believed that slack time provides opportunities for recovery. This may be proceduralized recovery as well as unproceduralized recovery.

To limit the credit taken for slack time, if the time window is less than 5 minutes, unproceduralized recovery is not modeled. Certainly, this assumption could be relaxed if specific circumstances do allow recovery. For instance, if an operator is starting a pump locally and can hear the noise of the running pump, then the lack of noise of the running pump can be considered as a detection source of a recovery action, regardless of the duration of the available time window.

The following criteria are used in time application:

a) If the average crew time is greater than the available time window, the operator action is considered as being failed; HEP of 1.0E-0 is applied.

b) If the average crew time is equal to the available time window or, if the slack time is less than 5 minutes, then the HEP is calculated based on failure of subtasks of the action; no credit is taken for recovery.

c) If the average crew time is less than the available time by 5 minutes or more, then unproceduralized recovery and proceduralized recovery (if provided) are considered. (Note: In most cases, only unproceduralized recovery is modeled. Proceduralized recovery is sometimes judged to be non-essential since failure to do the recovery step does not fail the action).

d) If slack time is between 5 minutes and 60 minutes, then unproceduralized recovery is applied.

e) If slack time is between 60 minutes and 3 hours, an additional recovery factor is applied by using a moderate dependency on the THERP BHEP of 8.0E-02; this gives a factor of 0.21. Therefore, for an event having a slack time of 90 minutes, recovery is: "[8.0E-02 x 0.1 x 0.21] x (stress factor and other PSFs)".

f) If slack time is greater than 3 hours, an additional recovery factor is applied by using a high dependency on the THERP BHEP of 8.0E-02; this gives a factor of 0.54. Therefore, for an event having, a slack time of 4 hours, recovery is: "[8.0E-02 x 0.1 x 0.21 x 0.54] x (stress factor and other PSFs)".

The additional recovery factor outlined in (f), is avoided for conservatism and the recovery stated in (e) above is applied for events having slack time greater than 3 hours.

10. Commission errors are evaluated in the HRA based on the concept of the operator doing something different from what is intended. Errors such as selecting the wrong control for the equipment, selecting wrong component display, or misreading a plant parameter fall into this classification. The THERP BHEP for the most common or credible cases of commission errors is about 1.3E-03.

From talk-throughs and control room visits, it was realized that controls are properly labeled, mimic lines are clearly drawn, and violation of stereotypes does not exist. Most importantly, symptom-based procedures have explicit equipment identifications, and the actual equipment or instrument numbers are communicated back and forth between the operator who is reading the procedure and the operator who carrying out the action on the control board. Therefore, it is believed that commission errors are less than the THERP BHEPs.

There is no guideline to modify THERP BHEPs to account for such factors. Therefore, based on engineering judgement, a factor of 0.1 is applied to the BHEP for commission errors to account for the benefits in using the new type of procedures, along with the operating philosophy that includes constant communication feedback. The 0.1 adjustment factor is consistent with modifications made in typical HRA using the THERP methodology.

In general, the 0.1 multiplicative factor for commission errors is used for any event except the early actions in an anticipated transient without scram (ATWS) event, which have a time window of approximately 2 minutes. These ATWS cases are categorized as skill-based actions; operator responses are somewhat second nature. In other words, the 0.1 factor is applied only to rule-based activities where the operators are following the symptom-based procedures.

Although there may be several possible commission errors for the same task, one commission error is chosen for each task - the one most credible for the particular tasks. This method is not believed to have an impact on the results.

It should also be mentioned that the impact of committing a specific error is not evaluated in the HRA. The HRA evaluates the possible types of reason why an action is not done; (i.e. omission, commission, etc. resulting in failure of the event).

11. In analyzing the operator actions used as basic events in the fault trees, a general assumption is made. In accordance with THERP, failure to use written procedures under normal/abnormal conditions is applied. However, failure based on the assumption that the operator uses the procedures under such conditions is also evaluated. In that regard, the analysis of such activities evaluates the failure probability based on the assumption that the operator may or may not use the procedure. In other words, the failure probability of the activity is the summation

of the failure probabilities of the actions when the operator uses the procedure and when he does not use the procedure.

12. The dependency evaluation covers positive dependency between events whereby failure on the first task increases the probability of failure on the second task. The evaluation does not cover negative dependency which implies that failure on the first task reduces the probability of failure on the second task; application of negative dependency produces results that may not be realistic.

    Dependencies are evaluated by the equations provided by THERP[44] Tables 20-17 and 20-18.

    The dependency modeling is addressed as conditional probabilities based on the following set of criteria:

    a) Dependencies in manipulating 2 or more of the same type of component, by the same operator in the same procedure step are modeled as follows:

        1. Failure to operate 2 of 2 controls is modeled with the second action having a low dependency of the first action. The model reflects BHEP x 0.05. In most cases, however, a moderate dependency, which results in BHEP x 0.15, is applied.

           If the operator manipulates both controls together, then complete dependency is assumed; that is, if one control is missed, the other is missed also.

        2. Failure to operate 3 of 3 controls is modeled with the second action having a low dependency of the first action, and the third action having a moderate dependency on the previous actions. The model reflects BHEP x 0.05 x 0.15.

        3. Failure to operate N of N controls (N > or = 4) is modeled with the second action having a low dependency of the first action, the third action having a moderate dependency on the previous actions, and fourth and subsequent actions each having a high dependency on previous actions. The model reflects BHEP x 0.05 x 0.15 x 0.5 x ... x 0.5. In general, one high dependency value (0.5) is assigned for all fourth and subsequent actions. Therefore, the joint conditional probability, for N > that 4, is evaluated by BHEP x 0.05 x 0.15 x 0.5.

        4. Failure to operate M of N controls (2 < M < N) is modeled by applying the appropriate dependency level (shown in 1, 2 or 3

above) based on the value of M. The binomial coefficient of "M out of N" is used in this evaluation. For example, failure to operate 2 of 4 controls will reflect BHEP x 0.15 x 6.

NOTE: See the notes 1 and 2 of Table 3.3.3-1 for usage of THERP conditional probability equations.

b) In general, zero dependency is assumed between subtasks. Given the symptom-based procedures being used, operators are following the steps as directed. Dependency between subtasks is believed to be valid if proper procedural guidance is unavailable or during knowledge-based responses. The events analyzed in the HRA are categorized as rule-based activities whereby procedural guidance is followed.

On the other hand, where groups of subtasks are provided as different options for accomplishing the same goal, dependency modeling is conducted similar to that described in (a) above. An example of this is the actions in performing depressurization using the condenser steam dumps or using the atmospheric relief valves; the second option is modeled as being dependent on the first.

c) Dependencies between different events are evaluated, based on factors such as time window, slack time, complexity of tasks, and type of procedural guidance available.

For this type of conditional probability evaluation, a more extensive or intricate analysis is performed using the charts provided in Figures 3.3.3-2 through 3.3.3-4. The answers to questions in this table are agreed upon by the cognizant system analysts and HRA analyst.

The starting point in Figures 3.3.3-2 through 3.3.3-4 is to determine the stress level of the first (or preceding) task. If low stress level is used for the first task, then Figure 3.3.3-2 is used; if moderate stress Figure 3.3.3-3 is used and if high stress, Figure 3.3.3-4 is used.

The exercise continues with the aim of determining factors specific to the second task such as time window, slack time, complexity of the tasks, and the type of procedural guidance. The end result is the deducing of the dependency level for the second task.

Conditional probabilities are then documented in Table 3.3.3-1 which summarizes and captures the factors considered in Figures 3.3.3-1 through 3.3.3-4.

13. Omission errors are evaluated in the HRA, based on the concept of the operator skipping the steps that are essential to the success of the task. THERP has BHEPs for omission when procedures with checkoff are used, and a higher set of BHEPs when procedures without checkoff are used.

    Operator talk-throughs have revealed that operators do mark the pages of the procedure as they go through them. Although emergency procedures do not have checkoff provision boxes, given that more explicit symptoms-based procedures are used and operators do check off steps, the BHEPs defined in THERP for "omission when procedures with checkoff provisions" are used.

    Omission errors can be made if the operator reading the procedures skips a step, or if the control board operator skips the step. On that basis, no adjustment is made to the BHEP for error of omission. In that respect, omission errors are treated differently from commission errors.

14. No departure from the THERP methodology is taken in modeling operator failure to restore equipment after testing or maintenance. It is assumed that an independent checking is performed during restoration activities; hence, a recovery is applied to these tasks using the THERP BHEP of 1.6E-01; same value as that described previously for local recovery.

B. Summary of Applicable Procedure Steps

The events identified in the event trees and applicable steps of the procedures for the operator actions are:

Operator Action	Procedure [steps]
Cool down and depressurize RCS with ECA-3.1/3.2 (EC3)	EOP E-3 [21]; EOP ECA-3.1 [10 to 21]
Cool down and depressurize RCS for charging flow (ES1)	EOP E-[17]; EOP ES-1.2 [2 to 24]
Isolate ruptured SG by closing MSIV (ISO)	EOP E-0 [22]; EOP E-3 [2 & 3]
Assure long-term shutdown after an ATWS (LTS)	EOP FR-S.1 [1 & 5]
Manually trip reactor (MRT)	EOP FR-S.1 [1]; EOP E-0 [1]
Initiate bleed and feed with SI already actuated (OB1)	EOP FR-H.1 [9 to 12]
Initiate bleed and feed without SI already actuated (OB2)	EOP FR-H.1 [9 to 12]
Initiate bleed and feed with the loss of a DC bus (OB3)	EOP FR-H.1 [9 to 12]
Initiate bleed and feed after a steam line break (OB4)	EOP FR-H.1 [9 to 12]
Cool down the RCS after restoration of power (OCD)	EOP ECA-0.0 [16]

Operator Action	Procedure [steps]
Establish main feedwater during a LOCA (OM0)	EOP E-0 [16]; EOP FR-H1 [1 to 4]
Establish main feedwater with one steam generator available (OM1)	EOP E-0 [16]; EOP FR-H1 [1 to 4]
Establish main feedwater with feedwater pumps already running (OM2)	EOP E-0 [16]; EOP FR-H1 [1 to 4]
Locally establish main feedwater (OM3)	EOP E-0 [16 & 18]; EOP FR-H1 [1 to 4]
Cool down and depressurize RCS during a medium LOCA (OP1)	EOP E-1 [17]; EOP ES-1.2 [4 & 5]
Cool down and depressurize RCS during a small LOCA (OP2)	EOP E-1 [17]; EOP ES-1.2 [4 & 5]
Restore RCS inventory (ORI)	EOP ECA-0.2 [2 to 5];
Trip reactor after manual trip has failed (ORT)	EOP FR-S1 [1]; EOP E-0 [1]
Cool down and depressurize RCS <u>before</u> SG overfills (OS1)	EOP E-3 [4 to 21]
Cool down and depressurize RCS <u>after</u> SG overfills (OS2)	EOP E-3 [4 to 21]
Establish on-site power (OSP)	EOP ES-0.0 [5]
Restore SW to the turbine building (02----SWT----HE)	E-SW-02 [3.2.2]
Switch AFW from CST to SW (05B-CST-SWS-HE)	EOP E-0 QRF [4], E-1 QRF [7], E-2 QRF [5]
Manually initiate auxiliary feedwater (05B-MIAFW----HE)	EOP E-0 [7]
Isolate MU-3A (05BAV-MU3A-HE)	E-AS-01 [4.16.1]
Realign valve ICS-7A or ICS-7B after test (23IXV-ICS7AB-HE)	SP-23-100 [6.3.14 & 6.4.14]
Initiate ICS recirculation (23R-RWST-RHR-HE)	EOP ES-1.3 [7]
Locally open SW-1300A or B (31-LO-SW1300-HE)	EOP E-0 [4]
Manually initiate safety injection (33-MAN-SI-IN-HE)	EOP E-0 [4 & 8]
Realign valve SI-7A or SI-7B after test (33IMV--SI7AB-HE)	SP-33-098 [6.34.3 & 6.35.3]
Initiate low pressure injection (34I---LI2----HE)	EOP ES-1.2 [11]
Stop the reactor coolant pumps (36-RXCP-STOP-HE)	EOP E-0 [16]; EOP FR-H1 [1 to 4]
Isolate category B containment penetrations (56-CI-CAT-B--HE)	EOP E-0 [4 & 6]

The events identified in the flooding study are based on the Alarm Response Sheets.

C. Quantification

The model quantifies a HEP in terms of three phases:

Phase 1: Cognitive phase (detection/diagnosis/decision making) with Qd.
Phase 2: Action phase with Qa.
Phase 3: Recovery phase with Qr.

The model uses the following equation to estimate a HEP (Q):
Q = Qd*Qdr + [(1-Qd)+Qd*(1-Qdr)]*Qa*Qar, Equation 1

where the symbols are used as defined above; e.g. Qdr = cognitive phase failure probability. The above equation is illustrated in terms of an event tree in Figure 3.3.3-1.

Thus, the model will estimate the following quantities to calculate HEP:

1. Qd
2. Qdr
3. Qa
4. Qar

Equation 1 is usually simplified in the following manner:
Q = Qd*Qdr + Qa*Qar.

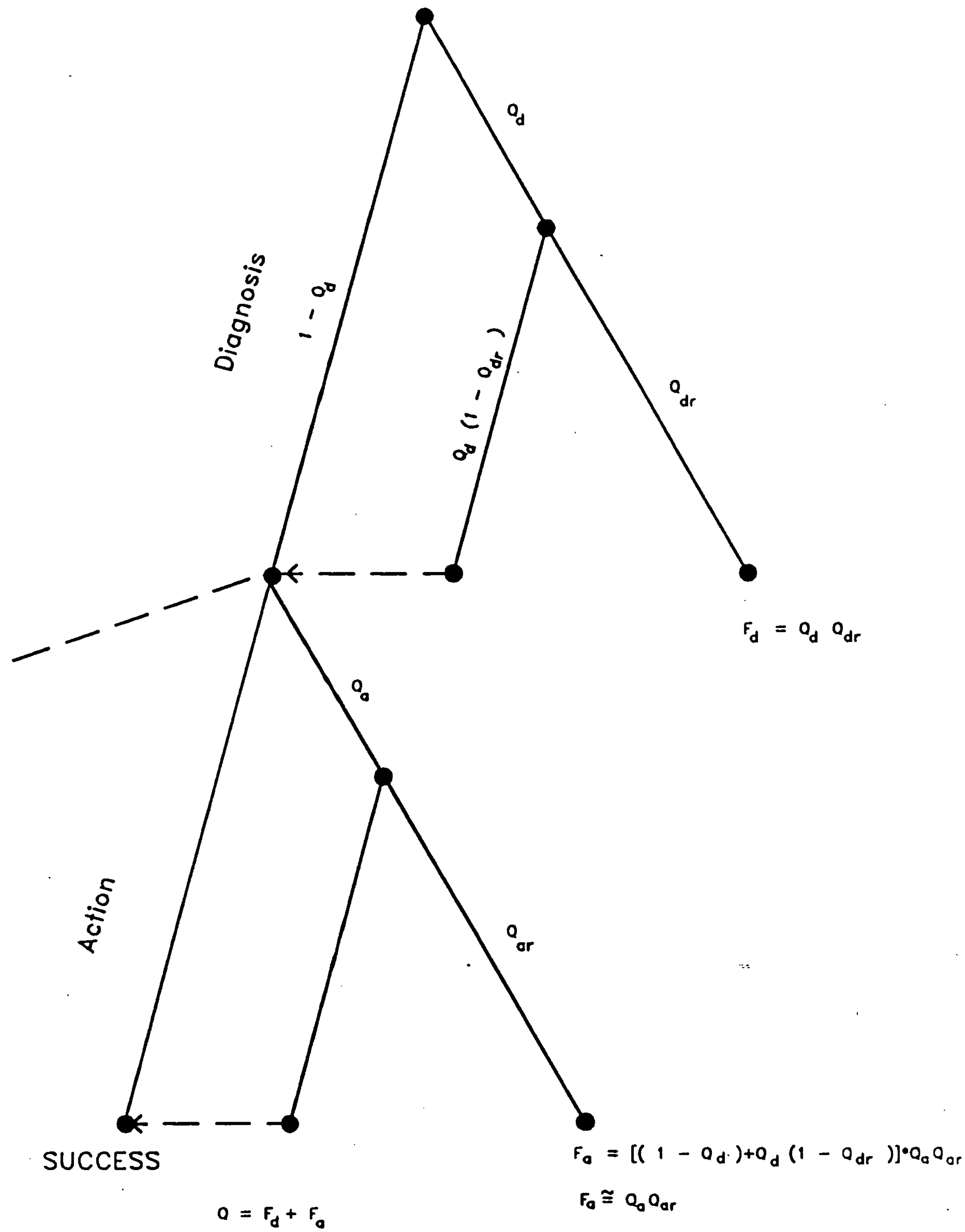


FIGURE 3.3.3-1
EVENT TREE FOR HUMAN ERROR MODELING

FIGURE 3.3.3-2 - **DEPENDENCY LEVEL EVALUATION**

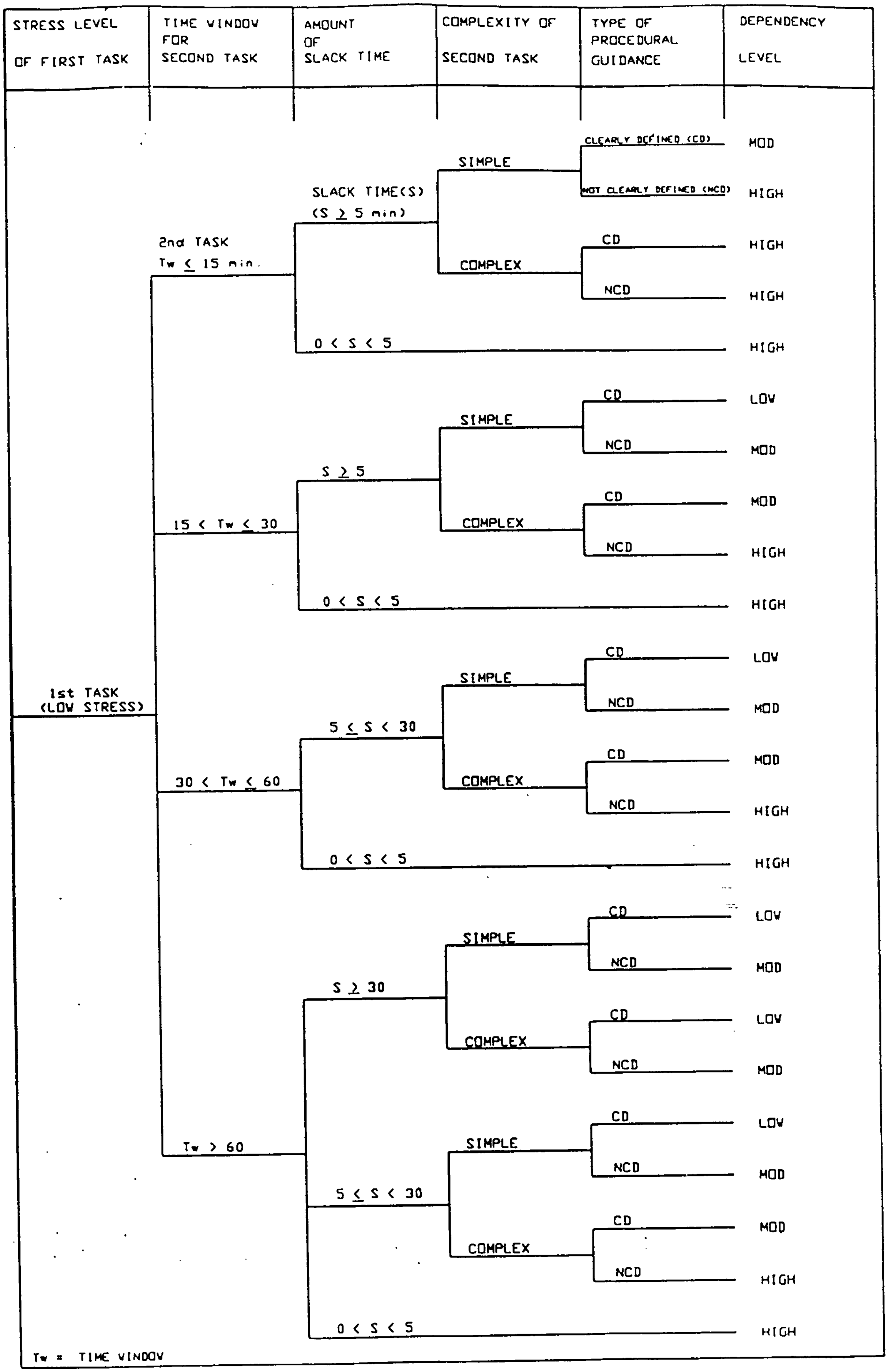


FIGURE 3.3.3-3 - **DEPENDENCY LEVEL EVALUATION**

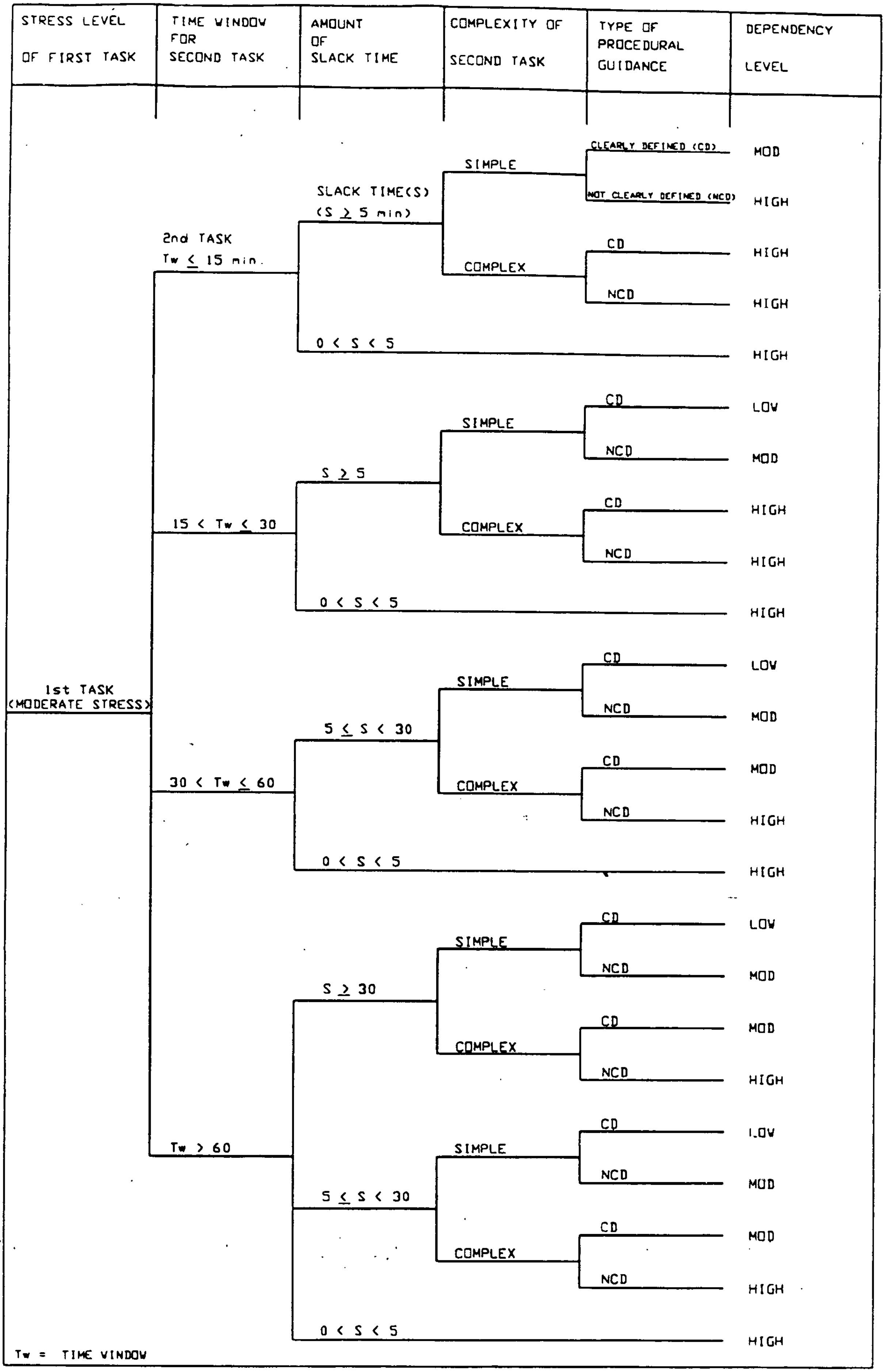


FIGURE 3.3.3-4 **- DEPENDENCY LEVEL EVALUATION**

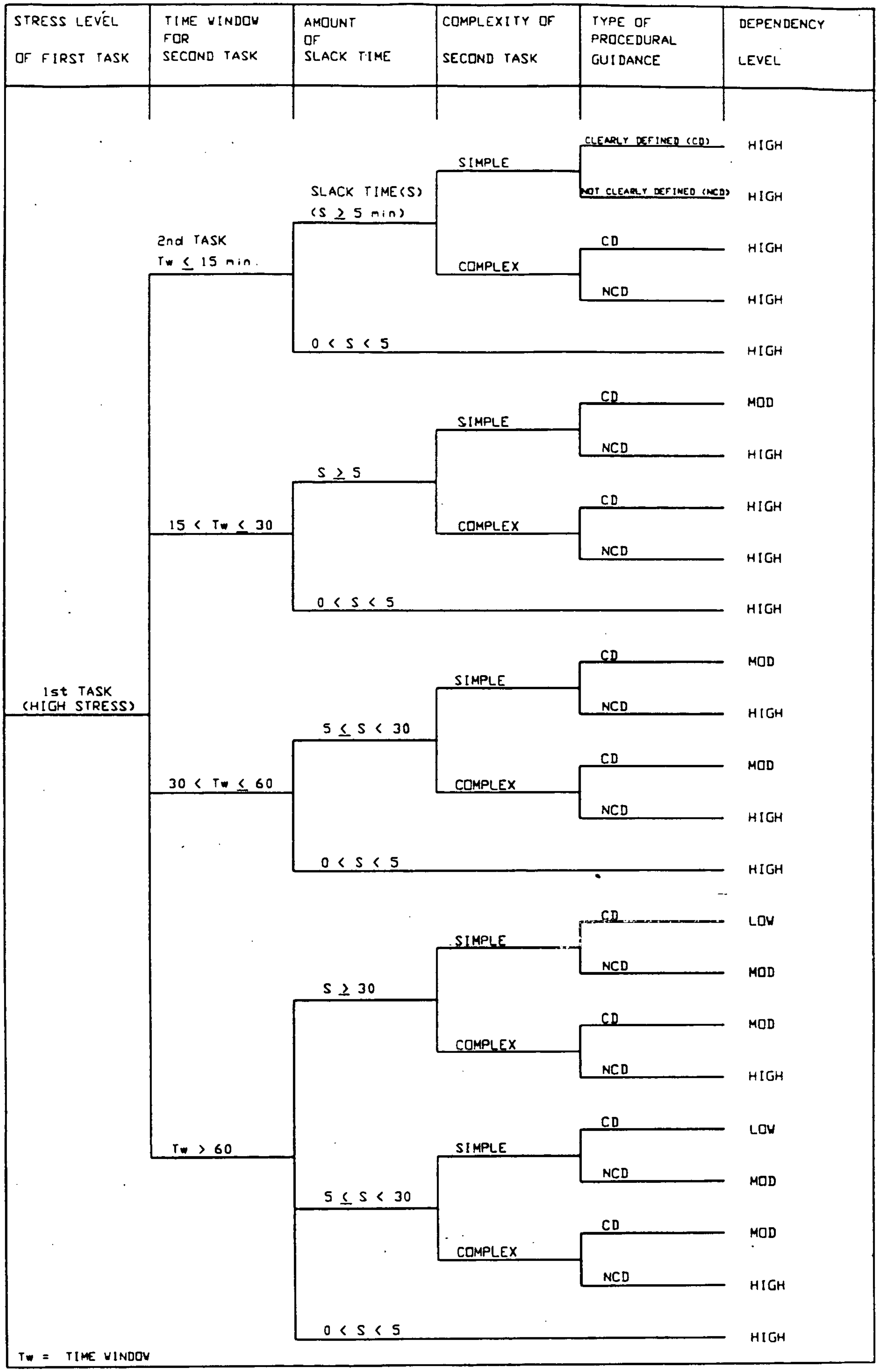


TABLE 3.3.3-1

CONDITIONAL PROBABILITY EVALUATION SUMMARY

Preceding Event		Dependent Event Characteristics							
Name	Stress	Name	Times		Tasks	Procedure	Dependency Level	Uncond. Prob.	Cond. Prob.
			Available	Actual	Simple or Complex	Clear or Unclear			
0S1	High	OS2	3.5 hours	0.5 hours	Complex	Clear	Low*	3.99E-3	5.00E-2
0S1	High	EC4	6.0 hours	0.5 hours	Complex	Clear	Low*	9.80E-3	5.00E-2

*Since the slack time is very large ($\geq$ 3 hours) low dependency is used even though medium dependency is indicated by the figure.

TABLE 3.3.3-1

CONDITIONAL PROBABILITY EVALUATION SUMMARY (Continued)

Notes

1. If Event Unconditional Failure Probability is less than or equal to 1.0E-02, then apply the conditional failure probability as follows:

   a) 0.05 for low dependency;
   b) 0.15 for moderate dependency; and
   c) 0.5 for high dependency.

2. If Event Unconditional Failure Probability is greater than 1.0E-02, then evaluate conditional failure probability by the applicable equation as follows:

   a) (1+19N)/20 for low dependency;
   b) (1+6N)/7 for moderate dependency; and
   c) (1+N)/2 for high dependency;

   Where, N is the unconditional failure probability of the dependent event.

3. Definition of terms are provided as follows:

   a) Time window - Available time to perform the required tasks before system failure occurs;

   b) Actual time - Estimated time to perform the required tasks;

   c) Slack time - "Time window" minus "Actual time";

   d) Simple task - Activities consisting of less than 10 steps, and not involving any specific operator interaction or dependency;

   e) Complex task - Activities consisting of 10 or more steps, and/or involving more than normal operator dependency;

   f) Clearly defined procedure - Steps are such that operators do not have to shuffle between procedures, and/or steps are not confusing or ambiguous.

3.3.3.2 Operator Action Descriptions

CHB - OPERATOR ACTION TO START CHARGING PUMPS POWERED BY THE TSC DIESEL

In a station blackout (SBO) situation, operators attempt to start two charging pumps powered by the technical support center (TSC) diesel generator. Since these procedure steps have not yet been written, a value of 1.0E-2 based on engineering judgement is used. This value represents a small but not unlikely probability. This is used because written procedures will be available, but it is a high stress action.

EC3 - OPERATOR ACTION TO COOL DOWN AND DEPRESSURIZE THE RCS WITH ECA-3.1/3.2 AFTER A FAILURE TO ISOLATE

In ECA-3.1, SGTR With Loss of Reactor Coolant - Subcooled Recovery Desired, the operators initiate a cooldown to cold shutdown at a maximum rate of 100 F/hr using the intact steam generator (SG). Once residual heat removal (RHR) entry conditions are established, the RHR System can be placed in service to continue the cooldown to cold shutdown. As the cooldown progresses, the high pressure safety injection (SI) pumps are sequentially stopped. The reactor coolant system (RCS) is then depressurized to minimize the break flow. EC3 models this action after a failure to isolate the ruptured steam generator (ISO).

EC4 - OPERATOR ACTION TO COOL DOWN AND DEPRESSURIZE THE RCS WITH ECA-3.1/3.2 WITH SG SAFETY VALVES STUCK OPEN

In ECA-3.1, SGTR With Loss of Reactor Coolant - Subcooled Recovery Desired, the operators initiate a cooldown to cold shutdown at a maximum rate of 100 F/hr using the intact SG. Once RHR entry conditions are established, the RHR System can be placed in service to continue the cooldown to cold shutdown. As the cooldown progresses, the SI pumps are sequentially stopped. The RCS is then depressurized to minimize the break flow. EC4 models the same actions as EC3, but after a failure to cooldown and depressurize the RCS (OS1), and a failure of steam generator secondary side integrity (SSV).

ES1 - OPERATOR ACTION TO COOL DOWN AND DEPRESSURIZE THE RCS FOR CHARGING FLOW

In EOP ES-1.2, Post LOCA Cooldown and Depressurization, the operator initiates a cooldown at a maximum rate of 100°F/hr using the intact SG. Once the RHR entry conditions are established (RCS pressure less than 425 psig, coldest RCS wide range temperature less than 380°F), the RHR system can be placed in service to continue the cooldown to cold shutdown

(200°F). As the cooldown progresses, the SI pumps are sequentially stopped (based on specified subcooling and RCS inventory criteria) with the charging pumps able to supply the required makeup. The RCS would also be depressurized (using pressurizer spray or a PORV) to increase inventory and to minimize the break flow. Using EOP ES-1.2, it may be possible for very small break cases to depressurize the RCS to near atmospheric pressure and thereby terminate or substantially reduce the break flow. By doing so, the charging flow can be reduced and switchover to high pressure recirculation can be avoided.

ISO - OPERATE ACTION TO ISOLATE THE RUPTURED STEAM GENERATOR BY CLOSING MSIV

During the above accident scenario, the operator is required to isolate the ruptured SG in order to reduce the offsite radiation dose following the steam generator tube rupture (SGR) event. The primary tasks controlling this event are to diagnose the SGR event, identify the ruptured SG and isolate the ruptured SG. The operator is required to isolate the SG by closing its main steam isolation valve (MSIV).

LTS - OPERATOR ACTION TO ASSURE LONG TERM SHUTDOWN AFTER AN ATWS

After an anticipated transient without SCRAM or main feedwater (AWS) has occurred and operators cannot trip the reactor by means of the manual pushbuttons (MRT) or by opening source breakers to the rod drive motor generator sets (ORT) they can still shut down the reactor by manually driving rods into the core or borating to cold shutdown.

MRT - OPERATOR ACTION TO MANUALLY TRIP THE REACTOR

Operators recognize that the plant is in an AWS situation by observing activation of the reactor protection system (RPS) with no indication of shutdown of the reactor. The activation of the RPS would be indicated by the lighting of red annunciators on annunciator panels located on mechanical vertical panel B. Indications of the reactor not being shut down are reactor trip breakers not open, neutron flux not decreasing, and control rod bottom lights not lit. Operators attempt to manually trip the reactor.

OB1 - OPERATOR ACTION TO INITIATE BLEED AND FEED WITH SI ALREADY ACTUATED

The operator recognizes the need for bleed and feed. Bleed and feed is initiated based on loss of secondary cooling and wide range SG level in the non-ruptured SG below 15% or pressurizer pressure increases above 2335 psig. Because the nature of the initiating event, SI is automatically actuated, and the operators simply have to establish a RCS bleed path by opening 1 of 2 pressurizer PORVs in order to establish bleed and feed.

Because of the complexity of this event, successful completion of this operator action is dependent on interpretation of the emergency operating procedures and also on plant parameters during the event (e.g. SG levels).

OB2 - OPERATOR ACTION TO INITIATE BLEED AND FEED WITHOUT SI ALREADY ACTUATED

OB2 is the initiation of bleed and feed for those initiating events where SI is not expected to automatically actuate. The operator recognizes the need for bleed and feed, and bleed and feed is initiated based on loss of secondary cooling and wide range steam generator level in either steam generator below 15% or pressurizer pressure increases above 2335 psig.

Because the nature of the initiating events, SI is not automatically actuated. Therefore, in order to establish bleed and feed, the operators have to manually initiate SI and establish a RCS bleed path by opening 1 of 2 pressurizer PORVs.

OB3 - OPERATOR ACTION TO INITIATE BLEED AND FEED WITH THE LOSS OF A DC BUS

OB3 is the initiation of bleed and feed for the loss of 125V DC bus (TDC) event, where SI is not expected to automatically actuate. The Operator recognizes the need for bleed and feed, and initiate it based on loss of secondary cooling and wide range SG level in either SG below 15% or pressurizer pressure increases above 2335 psig.

Because the nature of the initiating events, SI is not be automatically actuated. Therefore, in order to establish bleed and feed, the operators manually initiate SI and establish a RCS bleed path by opening the one available PORV.

OB4 - OPERATOR ACTION TO INITIATE BLEED AND FEED AFTER A STEAM LINE BREAK

The Operators recognizes the need for bleed and feed. Bleed and feed is initiated based on loss of secondary cooling and wide range SG level in the intact SG below 15% or pressurizer pressure increases above 2335 psig. Because the nature of the initiating event, SI is automatically actuated, and the operators simply have to establish a RCS bleed path by opening 1 of 2 pressurizer PORVs in order to establish bleed and feed.

Because of the complexity of this event, successful completion of this operator action is dependent on interpretation of the emergency operating procedures and also on plant parameters during the event (e.g. SG levels).

OB5 - OPERATOR ACTION TO INITIATE BLEED AND FEED DURING A LOSS OF OFFSITE POWER

OB5 is the initiation of bleed and feed for a loss of offsite power (LSP). In this case, SI is not expected to automatically actuate. The operator recognizes the need for bleed and feed, and initiate it based on loss of secondary cooling and wide range SG level in either SG below 15% or pressurizer pressure increases above 2335 psig.

Because the nature of the initiating events, SI is not automatically actuated. Therefore, in order to establish bleed and feed, the operators would have to manually initiate safety injection and establish a RCS bleed path by opening 1 of 2 pressurizer PORVs

OCD - OPERATOR ACTION TO COOL DOWN THE RCS DURING A STATION BLACKOUT

Given the above accident scenario, the emergency operating procedures instruct the operator to depressurize the intact SGs to 300 psig by locally dumping steam at the maximum rate using the SG PORVs. By establishing auxiliary feedwater (AFW) to and using the associated SG PORV for one or both of the SGs, it should be possible to cool the RCS to around 410°F within a one hour time period. By reducing the temperature to around 410°F or less, the reactor coolant pump (RXCP) seal leak rate is significantly reduced. In addition, for the more highly voided "problem" scenarios, the cooldown to 410°F results in an RCS depressurization that causes most of the contents of the accumulators to be injected.

OIP-OPERATOR ACTION TO ISOLATE RHR PUMPS

Operators identify that the RHR pump seals have failed and isolate the pumps by locally closing valves RHR 4A and 4B, thus isolating the leak.

OM0 - OPERATOR ACTION TO ESTABLISH MAIN FEEDWATER DURING A LOCA

For a loss of secondary cooling during a medium or small loss of coolant accident (LOCA), the operators are instructed to establish secondary cooling from the main feedwater MFW system. First, the operators stop the RXCPs in order to limit the heat input into the primary system. This is modeled as 36-RXCP-STOP-HE. The operators are then instructed to establish MFW flow by checking to ensure the condensate system is in service, checking to ensure the feedwater isolation valves are open, and establishing feedwater flow with normal operating procedure N-FW-5A.

OM1 - OPERATOR ACTION TO ESTABLISH MAIN FEEDWATER WITH ONE STEAM GENERATOR AVAILABLE

For a loss of secondary cooling during a SGR or steam line break (SLB), the operators are instructed to establish secondary cooling from the MFW system. First, the operators stop the RXCPs in order to limit the heat input into the primary system. This is modeled as 36-RXCP-STOP-HE. The operators are then instructed to establish MFW flow by checking to ensure the condensate system is in service, checking to ensure the feedwater isolation valves are open, and establishing feedwater flow.

If secondary cooling is established, the operators need to provide feedwater to the intact steam generator.

OM2 - OPERATOR ACTION TO ESTABLISH MAIN FEEDWATER WITH FEEDWATER PUMPS ALREADY RUNNING

For a loss of secondary cooling during a transient with main feedwater available or loss of component cooling, if AFW flow is unavailable, the operators are instructed to establish secondary cooling from the MFW. First, the operators stop the RXCPs in order to limit the heat input into the primary system. This is modeled as 36-RXCP-STOP-HE. The operators are then instructed to establish MFW flow by checking to ensure the condensate system is in service, checking to ensure the feedwater isolation valves are open, and establishing feedwater flow with normal operating procedure N-FW-5A. It is assumed that the MFW pumps remain running because SI is not initiated for these initiating events. Establishing MFW involves only opening the feedwater bypass valves FW-10A(B).

OM3 - OPERATOR ACTION TO LOCALLY ESTABLISH MAIN FEEDWATER

The loss of station and instrument air (INA) results in the loss of control room control for both trains of steam dumps, MFW control valves, and SG PORVs. The operator actions of OM3, therefore, assume that there is no control room control of equipment lost.

For a loss of secondary cooling during an INA event, if AFW flow is unavailable, the operators are instructed to establish secondary cooling from the MFW system. First, the operators stop the RXCPs in order to limit the heat input into the primary system. This is modeled as 36-RXCP-STOP-HE. The operators are then instructed to establish MFW flow. It is assumed that the MFW pumps remain operating because an SI signal has not been actuated. An SI signal causes the MFW pumps to trip and a feedwater isolation signal to be generated.

OM4 - OPERATOR ACTION TO ESTABLISH MAIN FEEDWATER DURING A LOSS OF A DC BUS

For a loss of secondary cooling during a loss of a 125V DC bus, if AFW flow is unavailable, the operators are instructed to establish secondary cooling from the MFW system. First, the operators stop the RXCPs in order to limit the heat input into the primary system. This is modeled as 36-RXCP-STOP-HE. The operators are then instructed to establish MFW flow by checking to ensure the condensate system is in service, checking to ensure the feedwater isolation valves are open, and establishing feedwater flow with normal operating procedure N-FW-5A. It is assumed that the MFW pumps remain running because SI is not initiated for these initiating events. Establishing MFW involves only opening the feedwater bypass valves FW-10A(B).

OP1 - OPERATOR ACTION TO COOL DOWN AND DEPRESSURIZE THE RCS DURING A MEDIUM LOCA

Operators cool down the RCS by dumping steam from the intact SG to maintain a maximum 100°F/hr cooldown rate. Once the break is uncovered, the RCS rapidly depressurizes. The cooldown is continued until conditions are met for placing the RHR system in service.

OP2 - OPERATOR ACTION TO COOL DOWN AND DEPRESSURIZE THE RCS DURING A SMALL LOCA

Operators cool down the RCS by dumping steam from the intact SG to maintain a maximum 100°F/hr cooldown rate. Once the break is uncovered, the RCS will rapidly depressurizes. The cooldown is continued until conditions are met for placing the RHR system in service.

ORI - OPERATOR ACTION TO RESTORE RCS INVENTORY

Given a station blackout, the operator is required to restore safeguard systems when power is restored. SI to restore RCS inventory is required and decay heat removal must be established. It is assumed that AFW is available and core uncovery has not occurred. SI with 1 out of 2 SI pumps is therefore required to restore the RCS inventory.

ORT - OPERATOR ACTION TO TRIP THE REACTOR AFTER MANUAL TRIP HAS FAILED

Operators recognize that the plant has not tripped even after passing manual pushbuttons. The activation of the RPS would be indicated by the lighting of red annunciators on annunciator panels located on mechanical vertical panel B. Indication of the reactor not being shut down is reactor trip breakers not open. Operators perform immediate actions of FR-S.1, Response to Nuclear Power Generation/ATWS, which are to manually drive the control rods into the core, open the supply breakers to the buses supplying the control rod drive motor-generator (MG) sets, and locally trip the reactor trip breakers. For this node, it is assumed that manually pressing the reactor trip pushbuttons did not trip the reactor, and the actions of de-energizing the control rod MG sets are required to drop the control rods into the core and stop the nuclear reaction. The physical failure of all control rods inserting into the core (e.g. structural failure of the control rod system) is not assumed.

OS1 - OPERATOR ACTION TO COOL DOWN AND DEPRESSURIZE THE RCS AND STOP THE LEAK BEFORE THE RUPTURED STEAM GENERATOR OVERFILLS

The RCS cooldown is performed using the intact SG. The operators dump steam at a maximum controllable rate using the steam dump system or atmospheric PORV until a required core exit thermocouple temperature is reached. The RCS is then depressurized using pressurizer spray or, alternatively, using pressurizer PORVs or auxiliary spray. Ideally, the depressurization is stopped when the RCS pressure is less than the ruptured SG pressure and pressurizer level is greater than 5%. The depressurization is also stopped if pressurizer level reaches 74% or RCS subcooling drops below 30°F. After RCS depressurization, SI flow is terminated to limit the RCS pressure and stop the primary to secondary leak.

OS2 - OPERATOR ACTION TO COOL DOWN AND DEPRESSURIZE THE RCS AND STOP THE LEAK AFTER THE RUPTURED SG OVERFILLS

Operators fail to stop the leak in the ruptured SG before it overfills: After the SG overfills, however, the operators perform actions to cooldown and depressurize the RCS and stop the leak. The actions are similar to those of OS1, except for the fact that the ruptured SG has overfilled. The high level actions are initial RCS cooldown using the intact SG, RCS depressurization using pressurizer sprays (or PORVs or auxiliary spray), and SI termination. In this operator action, it is assumed that it is obvious to the operators which SG is ruptured and that diagnosis and identification of the ruptured SG are not required for success of the actions.

OSD - OPERATOR ACTION TO STOP THE RCS DEPRESSURIZATION BY CLOSING THE PORV

Nodes OS1 and OS2 require a depressurization of the RCS. If the pressurizer PORVs are used for this purpose, there is a possibility that they do not close, in which case RCS inventory is lost through the PORVs. This action consists of closing the open PORV.

OSP - OPERATOR ACTION TO ESTABLISH ON-SITE POWER

If both emergency diesel generators fail to automatically start upon a LSP, the operators are instructed to manually start the emergency diesel generators in order to power the safeguards buses.

OSR-OPERATOR ACTION TO MINIMIZE ECCS FLOWS

Operators identify that recirculation capability has been lost and therefore transfer to ECA-1.1, Loss of Emergency Coolant Recirculation. They then determine the minimum SI flow required to remove decay heat, and throttle valve SI-7A or 7B (depending on which SI pump is running) to obtain that flow rate.

PPR - OPERATOR ACTION TO OPEN THE CLOSED PORV BLOCK VALVE

If an AWS occurs and the operators succeed in manually tripping the reactor (i.e. MRT and ORT fail), it is still necessary to relieve the resulting pressure increase in the Reactor Coolant System. This is normally done automatically. In the unlikely situation in which one PORV fails to open and the other PORV's block valve is closed, an operator action is necessary to open the closed block valve.

02----SWT----HE - OPERATOR ACTION TO RESTORE SW TO THE TURBINE BUILDING

If one service water (SW) pump is lost, that train can only supply loads for that train in the auxiliary building. The turbine building loads must be transferred to the train with two operable pumps. This action models that switchover.

05B-CST-SWS--HE - OPERATOR ACTION TO SWITCH AFW FROM CST TO SERVICE WATER

If the suction path to the AFW pumps from the condensate storage tanks (CST) becomes unavailable, operators are instructed to first try to establish main feedwater or condensate flow, and then to attempt to switch AFW suction from the CST to SW. This action represents the switchover of AFW suction.

05B-MIAFW----HE - OPERATOR ACTION TO MANUALLY INITIATE AUXILIARY FEEDWATER

This operator action considers situations in which the automatic signal to start AFW fails and the operators must start at least one AFW pump.

05BAV-MU3A---HE - OPERATOR ACTION TO ISOLATE VALVE MU-3A

If instrument air is lost to valve MU-3A, it is necessary to close manual valve MU-2A to prevent the CSTs from draining into the condenser. This is a local action to be performed by the nuclear equipment operator.

23I-MAN-ICS--HE - OPERATOR ACTION TO MANUALLY INITIATE ICS

If the automatic internal containment spray (ICS) signal fails, operators are instructed to manually initiate ICS. This is done by pressing the manual ICS pushbuttons or, if that doesn't work, opening ICS pump discharge valves, ICS-5A and ICS-5B, and starting the ICS pumps. Since the action is initiated at 23 psig containment pressure, well below the containment failure pressure, ample time is available to perform it.

23IXV-ICS7AB-HE - OPERATOR ACTION TO OPEN MANUAL VALVE ICS-7A OR ICS-7B AFTER TESTING

Surveillance Procedure SP-23-100, Containment Spray Pump and Valve Test - IST, requires operators to remove first one train, then the other train, of ICS from service. In order to test the ICS pumps without actually having flow to containment, manual valves ICS-7A and ICS-7B are closed when the corresponding train is tested. If a valve is not reopened, then the corresponding train is not capable of performing its function. This action models the reopening of the valves.

23R-RWST-RHR-HE - OPERATOR ACTION TO INITIATE ICS RECIRCULATION

If the operators are performing Emergency Operating Procedure ES-1.3, Transfer to Containment Sump Recirculation, and ICS pumps are still needed, the operators are instructed to transfer the suction of the ICS pumps from the RWST to the discharge of the RHR pumps. This action models that transfer. It is estimated that a time window of 48 minutes exists between RWST low level and RWST low-low level. Therefore, an estimated slack time of 20 minutes is credited to the operators to recover from switchover errors. Therefore, a medium dependency is applied for operator recovery during this slack time.

31-LO-SW1300-HE - OPERATOR ACTION TO LOCALLY OPEN SW-1300A OR SW-1300B

During normal operations, SW is supplied to the component cooling water (CCW) heat exchangers through SW-1306A and SW-1306B. When a SI signal occurs, the heat load on the CCW System increased dramatically. Each heat exchanger is therefore supplied with an additional SW valve (SW-1300A and SW-1300B), which receives a SI signal to open. In the unlikely event that the valve does not open, the Nuclear Auxiliary Operator is dispatched to locally open the valves.

Since the added SW capacity is only needed in the sump recirculation mode of operation, the operator has at least 30 minutes to open the valves, and more in most cases. Therefore, a medium dependency is applied for operator recovery during this slack time.

33-MAN-SI-IN-HE - OPERATOR ACTION TO MANUALLY INITIATE SAFETY INJECTION

If the safety injection (SI) sequencer fails, it is necessary to manually start the SI pumps and position the suction valves in order to provide flow to the RCS.

33IXV--SI7AB-HE - OPERATOR ACTION TO OPEN MANUAL VALVE SI-7A OR SI-7B AFTER TESTING

Surveillance Procedure SP-33-098, SI Pump and Valve Test - IST, requires operators to remove first one train, then the other train, of SI from service. In order to test the SI pumps without challenging the check valves in the SI lines, manual valves SI-7A and SI-7B are closed when the corresponding train is tested. If a valve is not reopened, then the corresponding train is not capable of performing its function. This action models the reopening of the valves.

33R-1TRN-REC-HE - OPERATOR ACTION TO ALIGN ONE SI TRAIN FOR RECIRCULATION

Operators recognize the need for sump recirculation based on 37% RWST level. They first establish flow to the residual heat exchangers from the CCW system. The operators then determine the number of operable SI/RHR trains. This action is used for the TDC event. BRA-104 is assumed lost. Therefore, only train B is available, so operators are instructed to maintain injection until the RWST level reaches 10%. SI pump B is then started to provide injection flow to the core while RHR train B is aligned for recirculation. Once the RHR train is aligned for recirculation, the operators determine whether high pressure recirculation should be established. If the RCS pressure is above 150 psig, then SI pump B is stopped and aligned to take suction from the RHR recirculation train. The SI pump is then started and flow is established.

33R-2TRN-REC-HE - OPERATOR ACTION TO ALIGN 1 OF 2 SI TRAINS FOR RECIRCULATION

Operators recognize the need for sump recirculation based on 37% RWST level. They first establish flow to the residual heat exchangers from the CCW system. The operators then determine the number of operable SI/RHR trains. If two operable trains are available, train B is lined up for recirculation while train A maintains injection lineup. If only one train is available, the operators are instructed to maintain injection until the RWST level reaches 10%. The SI pump on the operable SI/RHR train is then started to provide injection flow to the core while the RHR train is being aligned for recirculation. Once the RHR recirculation train is aligned, the RHR pump is started and flow is established. The operators then determine whether high pressure recirculation is required by checking to see if RCS pressure is above 150 psig. If it is not, then low pressure recirculation is continued. If it is above 150 psig, then high pressure recirculation is established.

34I---LI2----HE - OPERATOR ACTION TO MANUALLY INITIATE LOW PRESSURE INJECTION

After RCS cooldown is performed, it is necessary to start one residual RHR train in the injection mode. The only necessary operator action is to start one RHR pump.

34I---LI2A---HE - OPERATOR ACTION TO STOP THE RHR PUMPS

When a SI signal occurs, RHR pumps automatically start. At this point, unless there is a large break LOCA, the RCS pressure is higher than the shutoff head of the RHR pumps. The only flow through the pumps, then, is the recirculation flow back to the RWST. If this minimum flow continues more than 30 minutes, the pump can be damaged. Therefore, an operator action is necessary to stop the pumps so they can be used later on in the event.

36-RXCP-STOP-HE - OPERATOR ACTION TO STOP REACTOR COOLANT PUMPS

When AFW is not available after a reactor trip, it is necessary to provide alternate forms of cooling (MFW, condensate, bleed and feed). Each of these takes time to establish. During this time, there is heat input into the RCS via decay heat and RXCPs but no heat sink so the SGs are therefore losing inventory. Stopping the RXCPs greatly reduces the heat input into the RCS and thus greatly increases the length of time before the SGs boil dry. It is assumed, therefore, that if the operators fail to stop the RXCP's, there will not be enough time to establish alternate modes of cooling and core melt occurs.

40--BUS52----HE - OPERATOR ACTION TO ALIGN THE TSC DIESEL TO BUS 52

In a SBO situation, operators attempt to restore power to bus 52 with the TSC diesel generator. Since these procedure steps have not yet been written, a value of 1.0E-2 based on engineering judgement is used. This value represents a small but not unlikely probability. This is used because written procedures will be available, but it is a high stress action.

56-Cl-CAT-B--HE - OPERATOR ACTION TO ISOLATE CATEGORY B CONTAINMENT PENETRATIONS

When a containment isolation signal is received, certain penetrations (designated category B penetrations) are given a signal to close. If this signal fails for some reason, operators must manually close the valves.

CRDDET/CRDISO - OPERATOR ACTION TO DETECT AND ISOLATE FLOOD THE CONTROL ROD DRIVE ROOM

Given a flood in the control rod drive room, 480 volts buses 33 and 43 are damaged. This should alert operators to the problem. Otherwise, routine tours would detect the flood within 2 hours. Given that the flood is detected, the operator must then close both service water isolation valves, remotely. It is estimated that the maximum time available for detection and isolation is 130 minutes.

RELAYDET/RELAYISO - OPERATOR ACTION TO DETECT AND ISOLATE FLOOD IN THE RELAY ROOM

Given a flood in the relay room, routine tours would detect the flood and several alarms are expected to annunciate in the control room within 2 hours as flood water rises and cause loss of function to supported equipment. It is assumed, for this analysis, that the operator is required to respond to two alarms for diagnosis of the event. It is further assumed that the postulated flood is the result of a pipe break with a flow rate of approximately 28 gpm.

Therefore, the operator is expected to detect the flood by responding to both alarms within 2 hours. Given that the flood is detected, the operator must then perform local actions to close the isolation valve for the 1½ inch potable water line that goes through room 135. It is estimated that the maximum time available for detection and isolation is 2 hours.

TURBDET/TURBISO - OPERATOR ACTION TO DETECT AND ISOLATE FLOOD IN THE TURBINE BUILDING BASEMENT

Given a flood in the turbine building basement, several alarms come in. The first alarm is probably high turbine sump level, but other alarms, such as air compressor alarms, come in as well.

The operator is expected to detect the flood by responding to the sump alarm within 5 minutes. Given that the flood is detected, the operator must then shut off both circulating water pumps remotely. It is estimated that the maximum time available for detection and isolation is 15 minutes.

For this analysis, it is assumed that the operators are required to respond to 3 alarms annunciating simultaneously.

TABLE 3.3.3-2

QUANTIFICATION RESULTS SUMMARY

Event Tree Top Events	Identifiers	HEP
Start charging pumps powered by TSC diesel	CHB	1.00E-2
Cooldown and depressurize the RCS after failure to isolate	EC3	3.99E-3
Cooldown and depressurize the RCS with safeties stuck open	EC4	5.00E-2
Cooldown and depressurize the RCS for charging flow	ES1	4.99E-4
Isolate ruptured SG by closing MSIV	ISO	9.40E-4
Assure long term shutdown after ATWS	LTS	9.11E-5
Manually trip the reactor	MRT	2.03E-3
Initiate bleed and feed with SI actuated	OB1	9.81E-5
Initiate bleed and feed without SI actuated	OB2	3.88E-4
Initiate bleed and feed with the loss of a DC bus	OB3	5.90E-4
Initiate bleed and feed after a steam line break	OB4	9.81E-5
Initiate bleed and feed during a loss of offsite power	OB5	3.88E-4
Cool down the RCS during a station blackout	OCD	3.35E-3
Isolate RHR pumps	OIP	1.40E-2
Establish main feedwater during a LOCA	OM0	1.95E-3
Establish main feedwater with one SG available	OM1	3.30E-3
Establish main feedwater with MFW pumps running	OM2	1.19E-4
Locally establish main feedwater	OM3	1.92E-4
Establish main feedwater with the loss of a DC bus	OM4	1.19E-4
Cool down and depressurize the RCS during a medium LOCA	OP1	1.70E-3
Cool down and depressurize the RCS during a small LOCA	OP2	8.82E-4
Restore RCS inventory	ORI	9.96E-5
Trip the reactor after manual trip has failed	ORT	2.65E-2

TABLE 3.3.3-2

QUANTIFICATION RESULTS SUMMARY (Continued)

Cool down and depressurize RCS before SG overfills	OS1	9.80E-3
Cool down and depressurize RCS after SG overfills	OS2	5.00E-2
Stop the RCS depressurization	OSD	7.96E-4
Establish onsite power	OSP	4.34E-5
Minimize ECCS flows	OSR	3.32E-2
Open the closed PORV block valve	PPR	5.77E-4
Fault Tree Basic Events		
Restore SW to the turbine building	02----SWT----HE	4.39E-5
Switch AFW form CST to SW	05B-CST-SWS--HE	3.40E-4
Manually initiate AFW	05B-MIAIW----HE	5.45E-4
Isolate valve MU-3A	05BAV-MU3A---HE	3.68E-2
Manually initiate ICS	23I-MAN-ICS--HE	2.58E-3
Open manual valve ICS-7A or ICS-7B	23IXV-ICS7AB-HE	4.03E-5
Initiate ICS recirculation	23R-RWST-RHR-HE	5.86E-4
Locally open SW-1300A or SW-1300B	31-LO-SW1300-HE	2.20E-4
Manually initiate safety injection	33-MAN-SI-IN-HE	2.55E-3
Open manual valve SI-7A or SI-7B	33IXV-SI7AB-HE	4.03E-5
Align one SI train for recirculation	33R-1TRN-REC-HE	3.53E-4
Align 1 of 2 SI trains for recirculation	33R-2TRN-REC-HE	4.92E-5
Manually initiate LPI	341---LI2----HE	6.11E-5
Stop the RHR pumps	34I---LI2A---HE	4.23E-4
Stop reactor coolant pumps	36-RXCP-STOP-HE	2.33E-3
Align TSC diesel to bus 52	40--BUS52----HE	1.00E-2
Isolate category B penetrations	56-CI-CAT-B--HE	1.85E-4
Flooding Detection/Isolation		
Detect a flood in the CRD room	CRDDET	1.43E-6
Isolate a flood in the CRD room	CRDISO	3.33E-3

TABLE 3.3.3-2

QUANTIFICATION RESULTS SUMMARY (Continued)

Detect a flood in the relay room	RELAYDET	7.13E-7
Isolate a flood in the relay room	RELAYISO	4.90E-4
Detect a flood in the turbine building basement	TURBDET	1.30E-5
Isolate a flood in the turbine building basement	TURBISO	1.06E-4

3.3.4 Common Cause Failure Data

The purpose of this section is to provide a description of the methodology for treatment of common cause failures in fault tree models. These types of dependent failures are analyzed as part of a systems analysis and represent the dominant contributors to system failure. Several methods are available for the conduct of a quantitative common cause failure analysis to assess the contribution of these root causes. A summary discussion of the method used in the Kewaunee PRA study is provided below:

The Multiple Greek Letter (MGL) method model is an extension of the beta-factor model. In this method, other parameters in addition to the beta-factor are introduced to distinguish among common cause events effecting larger numbers of components in a higher order redundant system.

The MGL parameters consist of a set of failure fractions used to quantify the conditional probabilities of all the possible ways a common cause failure of a component can be shared with other components in the same group, given component failure has occurred. For a system of "m" redundant components and for each given failure mode, "m" different parameters are defined.

For example, the first three parameters of the MGL model are:

β = conditional probability that the common cause of a component failure is shared by one or more additional components

γ = conditional probability that the common cause of a component failure that is shared by one or more additional components is shared by two or more components additional to the first

δ = conditional probability that the common cause of a component failure that is shared by two or more additional components is shared by three or more components in addition to the first

The general equation that expresses the probability of multiple component failures due to common cause, Q_k, in terms of the MGL parameters is:

$$Q_k = \frac{1}{\binom{m-1}{k-1}} \left(\Pi_{i=1}^{K} \rho_i\right) \left(1 - \rho_{k+1}\right) Q_t$$

$$\rho_i = 1,\ \rho_2 = \beta_1,\ \rho_3 = \Upsilon,\ \rho_4 = \delta, \ldots . \rho_m + 1 = 0$$

where "m" is the number of components in the common cause group, and "k" is the number of specific components that fail such that $1 \leq k \leq m$. The binomial term

$$\binom{m-1}{k-1} = \frac{(m-1)!}{(m-k)!\,(K-1)!}$$

represents the number of different ways that a specific component can fail with (k-1) other components in a group of m similar components.

A. Procedure for Common Cause Analysis

Once the fault tree for a system has been developed to the appropriate level of detail, the common cause failure analysis can begin. The steps involved in the common cause failure analysis are summarized below.

- Identification of Common Cause Component Groups
- Placement of Common Cause in the Fault Tree
- Calculation of Common Cause Probabilities

1. Identification of Common Cause Component Groups

In this step, important common cause component groups are to be identified for inclusion in the system fault tree. The component groups for which common cause events may be defined include:

- Pumps
- Motor-Operated Valves
- Air-Operated Valves
- Check Valves
- Safety/Relief Valves
- Fan Coolers
- Diesel Generators
- Batteries
- Battery Chargers
- Reactor Trip Breakers
- Circuit Breakers

Common cause events for other component groups in a system may be defined if it appears that such an event would be an important contributor to system unavailability and if the components in the group can be linked to conceivable common cause failures such as those defined previously (design/manufacturing/construction inadequacy/abnormal environmental stress/etc.).

The above conditions are used to account for factors effecting component interdependence and to readily identify the presence of identical redundant components.

A review of plant data as well as design information to identify other dependent failure modes as well as subtle failures described in Reference 9 were also performed.

2. Placement of Common Cause in Fault Trees

Once the groups of components that have been judged to be susceptible to common cause failures have been identified, the fault tree is modified to include basic events representing the failure due to common cause. The common cause failure event may be shown at either of two modeling levels: (1) it may be shown as a top level event that is logically "OR"ed with random fault logic shown at the system level, or (2) it may be shown as a sub level event that is logically "OR"ed with random fault logic shown at the component level.

3. Calculation of Common Cause Probabilities

Once all important common cause failures were identified, the MGL method is used to quantify the common cause probabilities. A check of the plant specific data for common cause events is made prior to using the generic factors, thereby ensuring the generic factors are conservative.

The following procedure is used to calculate common cause probabilities.

a. For each set of multiple failures in a cutset that are identified, the cutset common cause failure probability is calculated by using equations given in Table 3.3.4-1 and the factors listed in Table 3.3.4-2.

   Q_i=failure of i similar components by common cause, i=2, 3, 4.

   Note that Q_i is a function of M, where M is the total number of similar components in the system being modeled by a fault tree. Also note that some cutsets contain an independent failure multiplying the common cause candidate failures. In such a case, Q_i must be multiplied by the probability of the additional independent failure.

b. Due to cutoff probability in obtaining cutsets, some cutsets containing higher order common cause candidates may be dropped and may not appear in the cutset list. To address such cutsets, the cutsets are examined to assure that all higher order cutsets with common cause candidates are present; if not, such cutsets are added to the list.

c. Any other special common cause contributors (if any) that are observed in activities such as plant walkdowns are added.

d. Recovery factors, if applicable, are introduced.

e. All common cause probabilities are summed and either entered as a single basic event in the appropriate box immediately below the top event; or an OR gate is placed below the common cause box in the fault tree and groups of common cause failures are defined to present a more detailed representation.

f. The common cause event is added to the master data file with the appropriate identifier in accordance with the data analysis section of the PRA.

Refer to Table 3.3.1-1 in section 3.3.1/2 for a summary of the common calculations for the Kewaunee PRA study.

g. The fault tree input file is edited to add the common cause event with the master data file identification number. The fault tree in question is then requantified so that the common cause contribution to total system failure probability is reflected.

B. Assumptions

The major assumptions in the model used here are summarized below:

1. Generic β, γ, and δ values in Table 3.3.4-2 are from Table C-1 of Reference 42.

2. It was assumed that the probabilities in the master data bank are QT (total probability including independent and common cause failures) for the MGL method.

3. For calculational ease, Q1 of the MGL method (probability of independent failures) was assumed to be the same as QT this is conservative since [Q1 = (1 - beta) QT].

TABLE 3.3.4-1

COMMON CAUSE EQUATIONS

M	Q^1	Q^2	Q^3	Q^4	Q^5	Q^6	Q^7	Q^8	Q^9	Q^{10}
1	1	-	-							
2	1-b	b	-							
3	1-b	1/2 b(1-c)	bc	-						
4	1-b	1/3 b(1-c)	1/3 bc(1-d)	bcd	-					
5	1-b	1/4 b(1-c)	1/6 bc(1-d)	0	bcd	-				
6	1-b	1/5 b(1-c)	1/10 bc(1-d)	0	0	bcd	-			
7	1-b	1/6 b(1-c)	1/15 bc(1-d)	0	0	0	bcd	-		
8	1-b	1/7 b(1-c)	1/21 bc(1-d)	0	0	0	0	bcd	-	
9	1-b	1/8 b(1-c)	1/28 bc(1-d)	0	0	0	0	0	bcd	-
10	1-b	1/9 b(1-c)	1/36 bc(1-d)	0	0	0	0	0	0	bcd

M = Total number of similar components susceptible to common cause failure in the system.

Q2 = Failure probability of two similar components due to common cause failure.

Q3 = Failure probability of three similar components due to common cause failure.

Q4 = Failure probability of four similar components due to common cause failure.

Q = Total failure probability of a given component type. ASSUME Q equals Q1, the failure probability of a component due to independent events. Also assume that Q is the probability value from the master data bank.

TABL .4-2

GENERIC MGL PARAMETER ESTIMATES[b]

Component	b (beta)	c (gamma)	d (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[c]	--d--	--d--
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	--d--
Auxiliary Feedwater	0.021	0.20	0.52
Service water and Component Cooling Water	0.032	0.63	0.84
Chillers	0.11[c]	--d--	--d--
Fans	0.13[c]	--d--	--d--
All	0.08[a]	0.33[a]	0.52[a]

[a] Average of all component failures.

[b] These MGL factors were calculated using the data from EPRI NP-3967 (Reference 40) except where noted differently.

[c] Generic estimates for beta based on data from NUREG/CR-4780 (Reference 41).

[d] Value of factor is not calculated. A value equal to the valve for the average of all component failures (All) given in this table should be used for the generic MGL screening method, that is, c (gamma) = 0.33 and d (delta) = 0.52.

3.3.5/6/7 Quantification Process

The core damage sequence integration and quantification process is based on the solution of a linked core damage sequence model that includes sequence logic, top logic models, frontline system models and support system models. The sequence logic models are developed from the event tree models which are produced as part of the accident sequence analysis. The quantification is performed in three stages; initial fault tree quantification, fault tree linking and core melt quantification.

A. Inputs and Codes

The following is a description of the various inputs and computer codes used in the quantification process.

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

   - Data Analysis Results
   - Human Reliability Analysis Results
   - System Fault Tree Models
   - Initiating Event Analysis Results
   - Accident Sequence Analysis Results
   - Batch Programs

2. The following computer codes developed by the Westinghouse Electric Corporation were used in the quantification process.

   - GRAFTER2 - Fault tree logic models
   - SIMON5 - Fault tree initial quantification
     WESCUT
     WESLGE
   - WLINK - Fault tree linking and accident sequence quantification
   - COMPLNK - Importance analysis
   - WALT - Sensitivity analysis

B. Process Description

The first step in the quantification process is to perform an initial quantification of all the system logic models. These models represent system hardware failures, operator actions, test and maintenance unavailabilities, common cause failures and support system dependencies. Support systems are not modeled explicitly in the various system fault trees. Each support system is included in a dependent system fault tree as a single undeveloped event with a unique identification code. This approach greatly reduces the size of the system fault trees, which results in a manageable number of cutsets for the dependent system. A failure rate (1E-1) is assigned to these undeveloped events to retain information on these interfaces that are used in fault tree linking process while ensuring that no important cutsets are lost.

The next step in the process is the final quantification of the fault trees used for core melt quantification. Prior to accident sequence quantification, each system fault tree is quantified via linking with all support systems explicitly included in the model. The linking model is used to link the cutsets of a system with its support system cutsets. The linking can be done before core damage quantification, when final system quantifications are occurring. This structure is best suited for evaluating plant design changes, licensing changes, and procedural changes. If a change is proposed that affects a support system, the separate structure enables quantification of the change in terms of component, cutset, and system unavailability before the effect of the change on core damage frequency is assessed. Some changes in frontline systems may also be adequately evaluated at the system level without the burden of the detailed support system subtrees.

The results of this quantification serve three general purposes:

- To provide a more realistic estimate of system unavailability

- To enhance understanding of system design and operation. With inclusion of support system modeling, dependencies between and among the systems and the various support systems are detected. For example, the cutsets may indicate that the failure of a given AC bus may fail a pump in the system as well as the instrument air supply.

- To provide a means for interpreting accident sequence cutsets. Because accident sequences are defined by an initiating event and failure of one or more key plant systems, each accident-sequence cutset represents the failure of the systems defined by the sequence. Therefore, in interpreting the results of an accident sequence quantification, it is incumbent on the analyst to determine which components in each cutset result in the failure of each system in the accident sequence.

The final step of the process is the accident sequence quantification. The plant accident sequences that result in core damage are defined in the set of event trees and are quantified using the fault tree linking approach. The fault tree linking model uses as input a tabular description of the event tree sequences involving system failure and success. Events involving system failure are input to the model as fault tree minimal cutsets, while other events such as accident initiation, system success, and operator action are input to the model as scalar quantities represented as single probabilities.

The system fault tree minimal cutsets are composed of basic component faults that are determined during the process of fault tree linking. Quantification of the accident sequences occurs in a step by step logical manner. The model initially processes the sequences by initiating event and produces output files. These initiator output files consist of accident sequence cutsets and sequence frequencies for each initiating event. Following this, the sets of accident sequence cutsets for each initiator are reduced and combined into one minimal set of total plant core damage accident sequences cutsets. Additional output includes the dominant core damage sequences and component importances.

3.3.8 Internal Flooding Analysis

An analysis was performed to determine potential accident sequences that could lead to core damage as a result of internal flooding and spraying at Kewaunee as part of the overall plant Level 1 PRA. Equipment may be damaged and fail as a result of internal flooding or spraying. This may prompt a reactor trip, and subsequent demand for the damaged components. The impact of these potential failures is assessed qualitatively, and where necessary, is analyzed quantitatively using the Level 1 PRA models with modifications due to internal flooding effects. The objective of this analysis is to determine the contribution to core damage that arises as a result of internal flooding or spraying.

3.3.8.1 Methodology

A flooding event has the potential to initiate a plant trip, and disable equipment required for a safe shutdown, possibly endangering core integrity. Summarized below are tasks employed to determine if such a scenario exists at Kewaunee:

- Data and Information Collection: Flooding Studies performed by architect engineers and by WPSC in response to NRC requests and INPO SOER 85-5 were reviewed. Equipment layout and potential flooding impacts were reviewed.

- Identified Flood-Induced Initiating Events: Possible events that could be initiated by a flooding event were identified.

- Identified Location of Critical Components and Flood Sources: Components required for mitigation of a flood-induced event were defined and located. System interdependencies and consequences of a possible flood were also defined.

- Screening Assessment to Locate Critical Flood Areas: Previous assessments that calculated flood levels were examined to determine if a postulated flood could initiate a plant trip and endanger safe shutdown components. Flooding events that could not induce a plant trip and endanger safe shutdown components were screened from further analysis.

- Defined Flood Protection in Each Critical Area: Both human and automatic flood detection and isolation were defined for each area where a flood would induce a trip and endanger safe shutdown components.

- Walkdown: A plant walkdown to confirm these findings was performed. Walkdown findings were examined and flooding vulnerabilities identified.

- Determining the Flooding Initiating Event Frequency: For those areas where flood vulnerabilities exist, the flooding initiating event frequency was calculated, accounting for flood initiation, automatic and human detection, and automatic and human isolation.

- Determining Flooding Consequences in Each Flood Area: Each component effected by the postulated flood or its propagation was identified, and its fault tree identifier defined if it was required in the respective flooding accident sequence. Effects from varying flood levels and their time dependence were defined. This was repeated for each area vulnerable to flood effects.

- Determining Flooding-induced Core Damage Frequency: The applicable accident sequence from the internal events analysis was then modified to reflect flood-induced failures in each flood area. Components not affected by the flooding event retained their random failure values, while the failure probabilities of flood-effected components were set to 1.0. The appropriate accident sequence was then quantified to determine that area's flood-induced contribution to core damage. This was repeated for each flood area.

3.3.8.2 Analysis

A. Assumptions

1. Doors with a gap beneath them of less than 1/2" are considered adequately sealed against possible propagation. No flood propagation was considered under doors with a gap of less than 1/2".

2. Doors opening toward a projected flood are assumed to remain intact when subjected to flood forces. Doors opening away from a flood are assumed to fail when water levels reach 3 ft. Diesel generator room doors, which swing out of the room, (doors 1 and 2) are assumed to fail when subjected to floods of 2 ft.

3. Walls and trench barriers are assumed to remain intact throughout a flooding event.

4. Pipes are assumed to leak as described in the NRC Standard Review Plan associated with flooding[23] and are treated as such in the Sargent and Lundy Flooding Analysis and the architect engineer calculations. Expansion joints and flex connections are assumed to catastrophically fail and are treated as such.

5. Insulated pipes are assumed to drip only, and not spray if a pinhole leak develops. This is assumed to be true for low and medium energy lines, but not true for high energy lines. A leak from a high energy line is assumed to be of sufficient energy to penetrate the surrounding insulation and protective sheet metal sheath. Encapsulated high energy lines are assumed to have a very low failure

rate, and the probability of a leak is considered inconsequential. Bare pipes are assumed to be spray sources for a 10 ft. radius from the pipe.

6. Spray damage is not assessed for those rooms already susceptible to flooding damage. Flooding damage is assumed to bound spray damage.

7. The frequency of concurrent spray and flooding events from different sources is assumed to be inconsequential. It was assumed that submersion of equipment definitely fails components, whereas water spray may or may not actually wet the component, causing failures.

8. Environmentally qualified (EQ) components are assumed to be operable to their safe positions when exposed to flood and spray conditions. For example, the solenoid operators for feedwater valves in the feedwater valve room will, by design, operate to their safe positions during HELB events.

9. The terms room, zone and area are interchanged and are considered to mean the same thing.

10. Lines not normally pressurized or charged such as drain lines and dry fire protection piping are not considered as credible flood or spray sources.

11. Water spray impingement or submersion of equipment that is not protected is assumed to result in equipment failure.

12. The plant is assumed to be at power or in hot shutdown mode during a flooding event. As specified in NUREG-1335[4], analysis of refueling and cold shutdown modes of operation is beyond the scope of this study.

13. When considering flooding as a result of a high energy line break (HELB), of the effects of increased humidity on equipment performance in the room due to a HELB is not included in this analysis and is beyond the scope of this study.

14. As specified in NUREG-1335[4], the mission time used for this analysis is 24 hours.

15. Limiting conditions for operation or Technical Specification violations are not considered as plant trip sources for this analysis. Only automatic reactor trips or immediate (within 2 hours of event) manual trips are considered.

16. Flooding in containment is not considered in this analysis. It is assumed that design basis analyses encompasses all floods in containment.

17. Rupture of seismic class I tanks (e.g. concrete reinforced refueling water storage tank) is not considered credible in this analysis.

18. Door failures consider door swing as well as construction integrity.

B. Identification of Flood-Induced Initiating Events

Flood-induced initiating events were postulated and reviewed to ascertain if they were possible at Kewaunee. The following flood-induced initiating events were postulated:

- Loss of Component Cooling Water
- Loss of Feedwater
- Loss of Condensate
- Loss of Condenser Vacuum
- Turbine Trip/Reactor Trip
- Loss of Service Water
- Loss of Offsite Power
- Station Blackout
- Loss of Instrument Air

After a review of these initiating events, it was concluded that flood-induced loss of offsite power, and station blackout are not credible flood-induced events at Kewaunee. This is true for the following reasons. The location of offsite power entry and subsequent distribution within the plant is not vulnerable to internal flooding. However, all other events listed above could feasibly be initiated by a flood at Kewaunee.

A loss of feedwater could occur by rising flood water from a turbine building basement flood rising to the feedwater pump motors, or propagating to room 16B, disabling the feedwater pump power supplies. A loss of condenser vacuum could occur from a rupture of a circulating water (CW) expansion joint (EJ). A turbine trip would follow. A loss of component cooling water could occur by flood water disabling the 480 V buses (buses 1-51 and 1-61) in rooms 5B and 5B-1, which supply the component cooling water (CCW) pumps. A loss of service water could occur by flood water disabling the 4160 V buses (buses 5 and 6 in rooms 2B and 3B) that supply the service water pumps. A loss of condensate could occur from the condensate pumps motors being damaged by flood water in the turbine building basement. A loss of instrument air could occur if both vital and non-vital compressors, which are located throughout the turbine building basement, were disabled.

The responses required for mitigation of these events can be represented by accident sequences that are already modeled in the PRA. The transients with and without main feedwater (TRA and TRS) are applicable. The applicability of the chosen accident sequence is discussed in the analysis of each flooding scenario.

C. Screening Assessment

For this flooding analysis, those events that lead to core melt, but not those events that result in economic losses are considered. To examine flood accidents that may lead to core melt, those systems and components required for a safe shutdown in the event of a trip are considered. For the initial screening of the many components that exist in the plant, a maximum flood in a room that disables everything in that room is postulated. Two questions are then posed:

1. Are there any safety critical components or systems within the room? Safety critical components or systems are defined as those required to perform a critical safety function. This includes safe shutdown systems, power supplies, instrumentation and control systems, as well as support systems such as cooling water for components.

2. Is a reactor trip initiated as a result of the postulated maximum flood?

The following scenarios are considered in this analysis:

1. Rooms that initiate a reactor trip upon a maximum flood, and contain safety critical components are included for further analysis. This is because the initiation of a reactor trip places a demand on the safe shutdown components. If a safe shutdown component is disabled, then the safe shutdown of the reactor may be jeopardized, and this scenario warrants further examination, since it may lead to core melt.

2. If the consequence of a postulated flood is the damage of safety critical components, but no reactor trip, then there is no flood-related demand upon the safe shutdown components. This scenario is not examined further in the analysis. It does not necessarily lead to a plant damage state, but may lead to economic losses or a limiting condition for operation of the plant.

3. If a maximum flood neither damages safety critical components nor initiates a reactor trip, then core melt initiated by flood need not be considered. There is no demand on the safe shutdown components initiated by a reactor trip, nor would those components be jeopardized. In this case the scenario will not be examined further.

4. If the consequence of a maximum flood is to initiate a reactor trip without disabling any safety critical components, then the plant is probably able to shut down safely if there are no coincident random failures of safe shutdown components. For instance, small floods in the turbine building may initiate a trip, but may not damage any components required for a safe shutdown. In this case, the reactor is able to safely shut down, if there were no coincident random

failures. Note that this case is equivalent to the TRA event modeled as part of the internal events PRA. However, this scenario is investigated further, considering the random failures of selected safe shutdown components in conjunction with a flood-induced transient.

In preparation for the walkdown, the screening process described above was followed to minimize the areas requiring investigation during the walkdown. This screening process requires that a reactor trip be initiated as a result of a flooding event to warrant further consideration. A summary of the areas follow:

Room Number	Description
1A	Circulating Water Pump Room
2B	Diesel Generator (DG) A Room
3B	Diesel Generator (DG) B Room
5B	480 V Swgr Bus 51 and 52 Room
5B-1	480 V Swgr Bus 61 and 62 Room
6B	Turbine Building (Condenser) - Basement Floor
16B	4160 V Swgr Bus 1 and 2 Room
121	Turbine Bldg Mezz Floor
129	Battery Room A
130	Battery Room B
135	Relay Room
162	West Feedwater Valve Room
226	Control Room
231A	Lower East Valve Room
233	Control Rod Drive (CRD) Equipment Room
243A	West Valve Room

In each of these rooms a reactor trip is initiated if equipment in the room is damaged by flooding. In these zones, the potential disabling of effected safe shutdown components from a flooding or spray event, and flood propagation potential was examined. The auxiliary feedwater pump rooms were also inspected during the walkdown.

In preparation for the walkdown, calculations for flooding in each vulnerable room were reviewed to determine the maximum postulated flood height, drainage, and dominant flood sources. The findings and assumptions used in the flooding calculations were verified during the walkdown. The calculations in the phases I and II flooding studies performed by Sargent and Lundy and WPSC were augmented with calculations from Fluor Daniel. Flooding rates and isolation times were reviewed. The consequences of flooding events were assessed, and results from previous studies were modified as necessary.

For instance, flood heights in the turbine building and surrounding areas, and in the diesel generator rooms were reassessed accounting for automatic flood isolation. The postulated flood would disable components required for plant operation and initiate a plant trip.

In addition to the aforementioned calculations, flooding due to drain backflow was also examined. Drain backflow was assessed by examining drain and trench drawings for the turbine building basement and the auxiliary building basement. It was found that flood heights due to drain backflow would be limited by the flow restrictors in the drain and trench lines, and that backflow would take a significant amount of time to occur.

The internal flooding analysis plant walkdown was conducted. The findings from the walkdown are summarized below in section D.

D. Summary of Walkdown Findings

The following areas were found to warrant further consideration in the flooding analysis:

Zone	Description
2B	Diesel Generator A Room
3B	Diesel Generator B Room
6B	Turbine Building Basement floor, flood propagates to:
	9B Turbine Building Basement (area connecting aux. bldg. with turbine building basement)
	10B Elevator B Machine Room
	11B Corridor and Ramps
	17B Waste Tank Area
135	Relay Room
233	CRD Equipment Room

E. Summary Evaluation of Areas Vulnerable to Flood Effects

Of the sixteen zones inspected during the walkdown, six flooding scenarios as listed above in the previous section were found to warrant further investigation. The bounding flooding scenario (most severe and/or most frequent) in each zone was examined. Flooding or spraying effects in each zone are discussed.

A flood in the rooms listed above damages components located in the room, initiating a plant trip and potentially disabling components required for successful mitigation of the trip. Rooms containing components whose failures could induce initiating events were identified. The sequence that best describes the required responses for mitigation of this event is described in the analysis of flooding in each room.

Each internal flooding scenario is modeled with its appropriate accident sequence, with changes to the internal event models as described above. Each sequence is named as follows:

Accident Sequence	Flood Description
FL1	CW inlet expansion joint on one of the condensers fails (winter conditions - only one circulating water pump running). Flood in 6B propagates to turbine building areas 9B, 10B, 11B and 17B
FL2	CW inlet expansion joint on one of the condensers fails (summer conditions - both circulating water pumps running). Flood in 6B propagates to turbine building areas 9B, 10B, 11B and 17B
FL3	Service water (SW) flex connection on DG A fails, producing flood in room 2B
FL4	SW flex connection on DG 1B fails, producing flood in room 3B
FL5	Flood in relay room from 1 1/2" potable water (PW) line failure
FL6	Flood in CRD equipment room from 3" SW line failure

Summary of Flooding Event FL1 Originating in Turbine Building Basement, Room 6B (winter operating conditions)

A flood in the turbine building basement due to failure of a condenser circulating water expansion joint failure does not last long before isolation due to stopping the operating CW pump, which is the source of flooding. However, some components are disabled by the flood water. A listing of effected components follows:

- Non-vital air compressors
    - D (room 6B)
    - E (room 6B)
    - F (room 6B)

- Motor control centers (MCCs) for non-vital air compressors, main and auxiliary transformer auxiliaries, turbine building sump pumps, turbine building basement for coils and condenser inlet motor operated valves (MOVs).

The most limiting event that could arise from a turbine building flood due to the rupture of one of the circulating water expansion joints is a loss of feedwater. This is represented in the PRA by the TRS event, with modifications to represent flood-induced failures.

Summary of Flooding Event FL2 Originating in Turbine Building Basement, Room 6B (summer operating conditions)

A flood in the turbine building basement due to failure of a condenser circulating water expansion joint failure does not last long before manual tripping of the circulating water pumps, which are the source of flooding. However, the projected flood level is 3 feet 4 inches in the turbine building basement and adjacent/open areas, and some components will be disabled by flood waters. A listing of effected components follow:

- Non-vital air compressors
  - D (room 6B)
  - E (room 6B)
  - F (room 6B)

- Condensate pumps

- MCCs for non-vital air compressors and other components

Main feedwater and condensate capabilities would be lost, as well as all non-vital air compressors.

The most limiting event that could arise from a turbine building flood due to the rupture of one of the circulating water expansion joints is a loss of feedwater. This is represented in the PRA by the TRS event, with modifications to represent flood-induced failures.

Summary of Flooding Event FL3 Originating Diesel Generator A Room, Room 2B

The failure of either one of the two SW flex connections on the diesel heat exchanger were determined to be the dominant flood sources in the room, with a flood height of 2 feet assumed before automatic isolation. Flood water disables 4160 V bus 5, which is flush with the floor. Disabling of breakers on bus 5 also disables SW pumps A1 and A2 which terminates the event.

It is assumed that the postulated flood reaches a height of 2 feet before flood isolation occurs. This is based on a 5 minutes isolation time. At a flood height of 2 feet, the doors leading to corridor 1A1 are expected to give way. Flood water then propagates to the screenhouse. The increased area over which the flood water disperses results in a lower flood height. Redundant DG B (room 3B) doors are thus subjected to a lower flood height. These doors swing into the flood, and according to assumption 2 do not give in. Affected components are limited to those in DG A room (room 2B).

A trench runs through room 2B, but has trench barriers between rooms 2B and 5B, 4B, and 1A1. There is a drain leading from the trench to a 4 in. drain line in room 3B, but there is a check valve in this line to prevent backflow. So water propagating to room 2B from elsewhere is not likely, and is not considered in this analysis.

The flood-effected components include the 4160 V switchgear bus 5, and DG A. Components dependent on bus 5 are listed below:

- RHR pump A
- Safety injection pump A
- Auxiliary feedwater pump A
- Service water pumps A1 and A2
- Diesel generator A

Also note that disabling bus 5 disables the normal power for buses 51 and 52.

After identifying flood-disabled components, the representative accident sequence for FL3 was chosen. This accident can be represented by the TRA event, with modifications to represent flood-induced failures.

Summary of Flooding Event FL4 Originating Diesel Generator B Room, Room 3B

The failure of either one of the two SW flex connections on the diesel heat exchanger were determined to be the dominant flood sources in the room, with a flood height of 2 feet assumed before automatic isolation. Flood water would disable 4160 V bus 6, which is flush with the floor. Disabling of breakers on bus 6 also disables SW pumps B1 and B2 which terminates the event.

It is assumed that the postulated flood reaches a height of 2 feet before flood isolation occurs. This is based on a 5 minutes isolation time. At a flood height of 2 feet, the doors leading to corridor 1AI are expected to give way. Flood water then propagates to the screenhouse. The increased area over which the flood water disperses results in a lower flood height. Redundant DG A (room 2B) doors are thus subjected to a lower flood height. These doors swing into the flood, and according to assumption 2 do not give in. Affected components are limited to those in B room (room 3B).

There is a floor drain that connects to a 4 inch line to the turbine building sump, but there is a check valve in this line to prevent backflow. So water propagating from elsewhere is not likely, and is not considered in this analysis.

The flood-affected components include the 4160 V switchgear bus 6, and DG B. Components that are dependent on bus 6 are listed below:

- RHR pump B
- Safety injection pump B
- Auxiliary feedwater pump B
- Service water pumps B1 and B2
- Diesel generator B

Also note that disabling bus 6 disables the normal power for buses 61 and 62.

After identifying flood-disabled components, the representative accident sequence for FL4 was chosen. This accident is represented by the TRA event, with modifications to represent flood-induced failures.

Summary of Flooding Event FL5 Originating in the Relay Room, Room 135

The dominant flood source in the relay room is a 28 gpm 1-1/2 inch PW line. Discussion of detection and isolation of this event follows.

Detection would occur either with hourly patrols in the room, with local PW alarms, or a control room alarm. Detection would take a maximum of 1 hour. Operating procedures are followed in the event of abnormal PW system operation. Isolation is accomplished through closure of either of valves PW3 or PW4, which isolate the main potable water header. It is estimated that isolation would be accomplished in a maximum of 10 minutes. Total detection and isolation time is estimated to be 70 minutes. This corresponds to a flood level of 1.95 inches.

The Foxboro panels, Westinghouse panels, relay panels, fuse panels, meter panels, electro-hydraulic panels, auxiliary relay racks, independent panels, annunciator cabinets, sequence of events recorder cabinets, Foxboro instrumentation racks, BOP instrument racks, seismic panel, digital distribution center panel, terminal cabinets, computer I/O cabinet, and computer "Y" panel cabinet are all in this room. The minimum dead space was found during an April 1990 walkdown performed by WPSC personnel was found to be 2 inches.

There are no drains in this room, so backflow via the drain system is not considered.

The postulated flood level of 1.95 inches is very close to the minimum dead space of 2 inches. For this analysis, it is conservatively assumed that the above listed components are effected. If the above listed components are effected, their failures prompt spurious control room alarms. These alarms prompt dispatching of personnel to relay room from control room, and swift isolation of flood.

The representative accident sequence for FL5 was chosen. This accident is represented by the TRA event, with modifications to represent flood-induced failures.

<u>Summary of Flooding Event FL6 Originating in the CRD Equipment Room, Room 233</u>

The dominant flood source in the CRD equipment room is an insulated 3 inches service water line, producing a flood of 3 inches This zone contains 480 V switchgear buses 33 and 43, MCC 62G, MCCs 43A and B, MCCs 33A and B, reactor trip breaker cabinet, rod drive control cabinet, DC holding supply cabinet, rod drive MG sets, and transformers. A flood of 3 inches threatens reactor trip breakers RTA and RTB, and 480 buses 33 and 43, which are flush with the floor, and the pressurizer heater transformer, which is elevated 2 inches. Other components in the room are not affected.

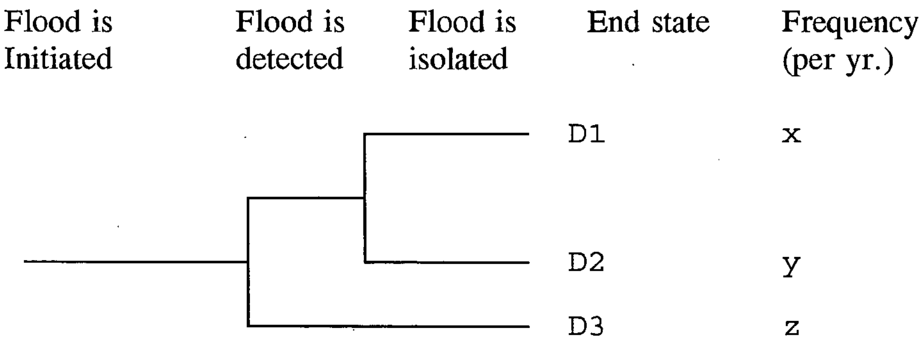
Backflow via the drain system is not considered credible for the CRD equipment room, since it is at the 626 foot elevation.

Detection of a flooding event in this room is expected to take 2 hours, and relies on routine security and Nuclear Auxiliary Operator (NAO) tours. Once detected, the approximate time required for isolation is 10 minutes. Isolation is accomplished by locally closing valve SW-300 manually.

After identifying flood-disabled components listed above, the representative accident sequence for FL6 was chosen. This accident is represented by the TRA event.

3.3.8.3 Quantification of Internal Flooding Initiating Event Frequencies

This section describes the calculation of the initiating event frequency for each of the flooding scenarios FL1 through FL6. The results of the calculation are presented in Table 3.3.8.3-1. This is the frequency at which a flood occurs in that area. Automatic system failures and human errors pertaining to detection and isolation of the flood are also considered in this calculation. These were compiled as shown below to calculate initiating event frequency.



Initiating event frequency = (x + y+ z), dependent on flood level

D1 = no damage or damage state dependent on flood level

D2 = damage state, knowledge of event, flood not isolated

D3 = damage state, no knowledge of event, flood not isolated

For all flooding scenarios in this analysis, flood levels under consideration were found to yield damage states, so D1 was added to D2 and D3 to yield the initiating event frequency.

TABLE 3.3.8.3-1

QUANTIFICATION OF INTERNAL FLOODING INITIATING EVENT FREQUENCIES

ACCIDENT SEQUENCE	FLOOD AREA	FREQUENCIES (Per Year)
FL1	Turbine Building Basement	8.9E-05
FL2	Turbine Building Basement	1.1E-04
FL3	Diesel Generator Room A	5E-04
FL4	Diesel Generator Room B	5E-04
FL5	Relay Room	1.5E-04
FL6	Control Rod Drive Equipment Room	1.5E-04

3.3.8.4 Quantification of Internal Flooding Contribution to Core Damage

Each of the vulnerable room is analyzed to determine the flood-induced contribution to core damage. Each flooding scenario is modeled individually. The initiating event frequency is changed from the internal events value to the initiating event frequency for that flooding sequence. The failure probabilities of flood-effected components are changed from their random failure values to flooding-induced failure values (1.0). Unaffected components retain their random failure values. Flooding-induced contribution to core damage is computed for each flooding sequence. The results of this contribution to core damage is summarized as follows.

Flood Zone	Core Melt Frequency
FL1 - Turbine Building Basement	3.2E-10
FL2 - Turbine Building Basement	4.0E-10
FL3 - Diesel Generator Room A	1.8E-07
FL4 - Diesel Generator Room B	5.8E-08
FL5 - Relay Room	4.0E-10
FL6 - Control Rod Drive Equipment Room	3.1E-10

Based on the results of the quantification described above, there is no credible internal flood/spray scenario that provides a significant contribution to the overall risk for Kewaunee.

The largest contributor to core melt is a flood in DG room A cased by the failure of a SW expansion joint. Even though the resultant core damage frequency is below the reportable limit, this flooding sequence is evaluated further in the quantification of plant damage states.

3.3.8.5 Quantification of Internal Flooding Contribution to Level 2 Resnlts

The following sequences represent over 90% of the core melt frequency due to flooding and include every systemic sequence with a frequency greater than 1.0E-8.

Sequence ID	Event Sequence	RCS Press	LP Recirc	Cont. Sprays	Fan Coil Units	Cont. Isol.	Frequency
FL3-1	SYS-CHG SYS-CCW	H	F	F	A	A	1.33E-07
FL3-2	SYS-CHG SYS-CCW	H	F	F	F	A	4.30E-08

Sequence ID	Event Sequence	RCS Press	LP Recirc	Cont. Sprays	Fan Coil Units	Cont. Isol.	Frequency
FL4-1	SYS-AF3 SYS-OM2 SYS-OB2	H	A	A	A	A	3.96E-09
FL4-2	SYS-AF3 SYS-OM2 SYS-OB2	H	F	F	F	A	1.08E-08
FL4-3	SYS-CHG SYS-CCW	H	F	F	F	A	4.12E-08

Using the methodology described in section 4.3, the internal flooding sequences are binned into containment event tree (CET) end states. The resulting CET end states are analyzed for source term in section 4.4. Therefore, each one of these sequences is bounded by an analyzed sequence in section 4.4. The table below shows the release category, the sequences that contribute to this release category, a description of the release category, the frequency of this release category for an internal flooding event, and conditional probability of this release category given the occurrence of internal flooding event leading to core melt.

Rel. Cat.	Bounded Sequences	Description	Frequency	Conditional Probability[1,2]
A	FL3-2, FL4-2, FL4-3	No containment failure within 48 hour mission time but failure could eventually occur without accident management action; noble gases and less than 0.1% of volatiles released.	9.50E-08	41.0
S	FL3-1, FL4-1	No containment failure (leakage only, successful maintenance of containment integrity; containment not bypassed; isolation successful).	1.37E-07	59.0

Notes:

1. Conditional probability of release category given an internal flooding event leading to core damage.

2. Total core damage frequency from internal flooding = 2.42E-07/year.

Consequently, no internal flooding sequence results in containment failure within the 48 hour Level 2 mission time.

3.4 Results and Screening Process

3.4.1 Application of Generic Letter Screening Criteria

Appendix 2 to Generic Letter 88-20[1] identifies the criteria for reporting potentially important sequences that might lead to core damage or unusually poor containment performance. The criteria applicable to Kewaunee are listed below.

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

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

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

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

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

6. Identification of sequences that, but for low human error rates in recovery actions, would have been above the applicable core damage screening criteria.

3.4.2 Sequences

A. Front-End Analysis Systemic Sequences

There are a total of seventy-one core melt sequences that were quantified for the Level 1 portion of the IPE study. These sequences are presented in Table 3.4.4-4. The first thirteen sequences contribute greater than 85% to the total core damage frequency. The sequence identifiers below are in the form IEV-#, where IEV is the three-letter initiating event description and # is the event tree endpoint number from Tables 3.1.2/3-1 through 3.1.2/3-16. These figures are referred to throughout this section. These sequences are presented again in Table 3.4.2-1 and are described in detail below.

The sequences quantified for the Level 2 portion of the IPE study are presented in section 4.0 of this report.

1. Sequence: SBO-6 and SBO-21

   Sequence Frequency: 1.30E-05
   Contribution to Core Melt: 19.57%

Initiating Event Frequency: 4.35E-04
Conditional Core Melt Frequency: 2.99E-02

Sequence Description:

A station blackout initiating event (loss of offsite AC power and loss of onsite emergency AC power) occurs, leading to a reactor trip. The turbine driven auxiliary feedwater (AFW) pump fails to start, so no secondary heat sink is available. Operators follow emergency operating procedure ECA-0.0, Loss of All AC Power. Procedures to restore power to one charging pump from the technical support center diesel generator are also followed but charging has no bearing on this event. The plant staff works to restore offsite AC power and also at least one of the onsite vital AC buses. However, neither AC power source is restored in 2 hours, by which time the steam generators have boiled dry and the core beings to uncover.

Core melt is postulated due to:

a. Loss of turbine driven AFW pump operation and;

b. Loss of safety injection due to the loss of AC power for greater than 2 hours.

Safety Issues Addressed:

a. Station blackout

b. Loss of auxiliary feedwater

Plant Specific Nature of the Sequence:

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

Modeling Assumptions:

a. No credit is taken for the restoration of charging for seal injection to prevent a RXCP seal LOCA since core melt is assumed to occur due to the loss of secondary cooling.

b. Credit has not been taken for AC power recovery within the time period from the loss of auxiliary feedwater at the initiation of the station blackout to the time of core uncovery (2 hours).

2. Sequence: SLO-4 and SLO-8

Sequence Frequency: 1.21E-05/year
Contribution to Core Melt: 18.21%
Initiating Event Frequency: 5.12E-03
Conditional Core Melt Frequency: 2.36E-03

Sequence Description:

A small LOCA event (a break ranging in size from 0.375 inches to 2 inches equivalent diameter) occurs, leading to a reactor trip and safety injection actuation. The high pressure safety injection pumps automatically start to provide flow to one of two reactor coolant system (RCS) cold legs. The AFW pumps automatically start and provide heat removal through at least one of two steam generators and their associated relief valves. The main feedwater (MFW) system remains available if AFW fails. Operators fail to correctly follow emergency operating procedure ES-1.2, Post LOCA Cooldown and Depressurization, to provide the initial cooldown and depressurization. Containment sump recirculation is attempted as a backup, but it fails too.

Core melt is postulated due to:

a. Failure of cooldown and depressurization to avoid depleting the RWST;

b. Loss of high pressure sump recirculation and;

c. Loss of low pressure sump recirculation.

Safety Issues Addressed:

a. Small LOCA

b. Failure of the long term cooling function following a small LOCA

c. Failure of the recirculation function following a small LOCA

Plant Specific Nature of the Sequence:

This event sequence is a low frequency sequence considered typical for currently operating PWRs. There is no additional failure mode observed for the sequence.

Modeling Assumptions:

a. Credit is taken for the successful operation of AFW or MFW.

3. Sequence: SBO-2 and SBO-11

Sequence Frequency: 8.70E-06/year
Contribution to Core Melt: 13.13%
Initiating Event Frequency: 4.35E-04
Conditional Core Melt Frequency: 2.0E-02

Sequence Description:

A station blackout event (loss of offsite AC power and loss of onsite emergency AC power) occurs, leading to a reactor trip. The turbine driven AFW pump automatically starts and provides heat removal through at least one of the two steam generators and their associated relief valves. Operators follow emergency operating procedure ECA-0.0, Loss of All AC Power, and provide rapid cooldown and depressurization to minimize a potential RXCP seal LOCA due to lack of seal cooling. Operators also follow procedures to restore power to one charging pump from the technical support center diesel generator but charging has no bearing on this event. The plant staff works to restore offsite AC power and also at least one of the vital AC buses. However, neither AC power source is restored in 11 hours.

Core melt is postulated due to:

a. Loss of turbine driven AFW pump operation due to DC battery depletion and;

b. Loss of safety injection due to the loss of AC power for greater than 11 hours.

Safety Issues Addressed:

a. Station blackout

b. Battery depletion after station blackout

Plant Specific Nature of the Sequence:

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

Modeling Assumptions:

a. Credit has not been taken for AC power recovery within the time period from the assumed loss of battery power (8 hours) to the expected time of core uncovery (11 hours).

4. Sequence: MLO-3

Sequence Frequency: 7.42E-06/year
Contribution to Core Melt: 11.19%
Initiating Event Frequency: 2.36E-03
Conditional Core Melt Frequency: 3.14E-03

Sequence Description:

A medium LOCA (a break ranging in size from 2 to 6 inches equivalent diameter) occurs, leading to a reactor trip and safety injection actuation. The high pressure safety injection pumps automatically start to provide flow to one of two RCS cold legs. Operators follow emergency operating procedures to mitigate the event. The long term cooling function provided by high pressure and low pressure sump recirculation fails, however.

Core melt is postulated due to:

a. Loss of high pressure sump recirculation and;

b. Loss of low pressure sump recirculation.

Safety Issues Addressed:

a. Medium LOCA

b. Failure of the recirculation function following a medium LOCA

Plant Specific Nature of the Sequence:

This event sequence is a low frequency sequence considered typical for currently operating PWRs. There is no additional failure mode observed for the sequence.

Modeling Assumptions:

a. Credit has been taken for high pressure injection.

5. Sequence: SGR-9 and 20

Sequence Frequency: 4.31E-06/year
Contribution to Core Melt: 6.51%
Initiating Event Frequency: 6.41E-03
Conditional Core Melt Frequency: 6.72E-04

Sequence Description:

A steam generator tube rupture event occurs, leading to a reactor trip and a safety injection actuation. The high pressure safety injection pumps automatically start to provide flow to one of two RCS cold legs. The AFW pumps automatically start to provide heat removal from the unaffected steam generator and its associated relief valve. MFW remains available and can be initiated if AFW fails. Operators follow emergency operating procedures to isolate the ruptured steam generator. Subsequent actions to cooldown and depressurize the RCS before the steam generator overfills is unsuccessful. Steam generator integrity cannot be maintained due to a safety valve failing open. Further operator action to cool down and depressurize the RCS atmospheric before core damage, fails as well.

Core melt is postulated due to:

a. Failure of operator action to cooldown and depressurize the RCS and terminate safety injection before the steam generator overfills;

b. Affected steam generator safety valves failing to close and;

c. Failure of operator action to cool down and depressurize the RCS to atmospheric.

Safety Issues Addressed:

a. Steam generator tube rupture.

b. Steam generator integrity following a steam generator tube rupture event.

c. Failure of the RCS cooldown and depressurization function following a steam generator tube rupture event.

Plant Specific Nature of the Sequence:

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

Modeling Assumptions:

a. Credit is taken for the success of the secondary cooling function provided by either AFW or MFW.

b. Credit is taken for the isolation of one of two steam generators.

6. Sequence: SBO-10

Sequence Frequency: 4.01E-06/year
Contribution to Core Melt: 6.06%
Initiating Event Frequency: 4.35E-04
Conditional Core Melt Frequency: 9.22E-03

Sequence Description:

A station blackout event (loss of offsite AC power and loss of onsite emergency AC power) occurs, leading to a reactor trip. The turbine driven AFW pump automatically starts and provides heat removal through at least one of two steam generators and their associated relief valves. Operators follow procedure ECA-0.0, Loss of All AC Power, and provide rapid cooldown and depressurization to minimize a potential RXCP seal LOCA due to lack of seal cooling. The operators are unsuccessful in restoring power to one charging pump from the technical support center diesel generator. The plant staff restores AC power to at least one AC vital bus within 11 hours. Core damage has already occurred, however, due to the RXCP seal LOCA.

Core melt is postulated due to:

a. RXCP seal LOCA after station blackout.

Safety Issues Addressed:

a. Station blackout

b. RXCP seal LOCA after station blackout

Plant Specific Nature of the Sequence:

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

Modeling Assumptions:

a. Credit has been taken for the recovery of one vital AC bus before the time period it is needed to provide a secondary heat sink (11 hours).

b. It is assumed that a RXCP seal LOCA has occurred before power restoration.

7. Sequence: TRA-9

Sequence Frequency: 2.69E-06/year
Contribution to Core Melt: 4.06%
Initiating Event Frequency: 3.00/year
Conditional Core Melt Frequency: 8.97E-06

Sequence Description:

A transient with main feedwater available event occurs, leading to a reactor trip. Power is available to at least one of the safeguards AC buses throughout the event. Operators follow emergency operating procedures to verify the success of the reactor trip function. The secondary cooling function provided by AFW fails. MFW, while initially available is lost, and not recovered. The heat removal function provided by primary system bleed and feed also fails.

Core melt is postulated to occur due to:

a. Loss of AFW and;

b. Loss of MFW and;

c. Failure of primary system bleed and feed.

Safety Issues Addressed:

a. Transients with main feedwater available.

b. Failure of the secondary cooling function following a reactor trip transient.

c. Failure of the primary cooling function (bleed and feed) following a reactor trip transient.

Plant Specific Nature of the Sequence:

This event is a low frequency sequence considered typical for currently operating PWRs in which MFW is available and easy to establish subsequent to the loss of AFW. There is no additional failure mode observed for this sequence.

Modeling Assumptions:

a. Credit is taken for the availability of onsite power to at least one of two vital AC buses.

8. Sequence: LSP-7

Sequence Frequency: 2.17E-06/year
Contribution to Core Melt: 3.27%
Initiating Event Frequency: 4.36E-02
Conditional Core Melt Frequency: 4.98E-03

Sequence Description:

A loss of offsite power event occurs, leading to a reactor trip. The onsite emergency generators automatically start and restore power to at least one of two vital AC buses. The AFW system fails to provide the short term cooling function. The operators follow the emergency operating procedures but are unsuccessful in establishing the heat removal function provided by primary bleed and feed.

Core melt is postulated due to:

a. Loss of AFW and;

b. Failure of operator bleed and feed.

Safety Issues Addressed:

a. Loss of offsite power.

b. Failure of the secondary cooling function provided by AFW following a loss of offsite power.

c. Failure of the primary cooling function provided by operator bleed and feed following a loss of offsite power.

9. Sequence: INA-3

Sequence Frequency: 2.08E-06/year
Contribution to Core Melt: 3.14%
Initiating Event Frequency: 1.07E-04
Conditional Core Melt Frequency: 1.94E-02

Sequence Description:

A loss of instrument air event occurs and all components using air for control revert to their fail safe positions. This includes the main feedwater regulating valves which supply the steam generators. A reactor trip/turbine trip occurs as a result of the loss of main feedwater flow. AFW fails to provide the secondary cooling function and the operators fail to establish MFW locally. Instrument air is not available for long term feed and bleed through the pressurizer PORVs.

Core melt is postulated due to:

a. Loss of AFW and;

b. Loss of MFW.

Safety Issues Addressed:

a. Loss of instrument air.

b. Failure of the secondary cooling function following a loss of instrument air.

Plant Specific Nature of the Sequence:

This event is a low frequency sequence considered plant specific based upon the instrument air system design and reliability. There is no additional failure mode observed for this sequence.

Modeling Assumptions:

a. No credit is taken for function provided by primary bleed and feed because it is assumed that the pressurizer PORVs will not stay open without instrument air.

10. Sequence: LLO-2

Sequence Frequency: 1.39E-06/year
Contribution to Core Melt: 2.09%
Initiating Event Frequency: 5.00E-04
Conditional Core Melt Frequency: 2.78E-03

Sequence Description:

A large LOCA event (breaks ranging in size from 6 inches equivalent diameter to a double-ended circumferential break of the largest primary coolant loop piping) occurs, leading to a reactor trip and safety injection actuation. The low pressure safety injection pumps automatically start to provide flow to one of two RCS cold legs and one of two vessel injection penetrations. One SI accumulator injects into the intact RCS cold leg. Operators follow the emergency operating procedures to monitor accident progression and mitigating systems operation during the injection phase but are unable to establish the long term cooling function provided during the recirculation phase.

Core melt is postulated due to:

a. Loss of low pressure sump recirculation.

Safety Issues Addressed:

a. Large LOCA

b. Failure of the recirculation function following a large LOCA

Plant Specific Nature of the Sequence:

This event is a low frequency sequence considered typical for currently operating PWRs. There is no additional failure mode observed for the sequence.

Modeling Assumptions:

a. Credit has been taken for accumulator injection.

b. Credit has been taken for low pressure injection.

11. Sequence: LSP-3

Sequence Frequency: 1.23E-06/year
Contribution to Core Melt: 1.86%
Initiating Event Frequency: 4.36E-02
Conditional Core Melt Frequency: 2.82E-05

Sequence Description:

A loss of offsite power event occurs, leading to a reactor trip. The onsite emergency generators automatically start and restore power to one of two vital AC buses. The turbine driven AFW pump automatically starts and provides heat removal through at least one of two steam generators and their associated relief valves. Additional AFW pumps start when power is restored to the vital AC buses. With the successful operation of AFW, the remaining concern is the potential for a seal LOCA when the RXCP cooling function is lost due to the failure of charging and component cooling.

Core melt is postulated due to:

a. Failure of the RXCP seal cooling function provided by seal injection from charging and;

b. Failure of the RXCP seal cooling function provided by component cooling to the RXCP thermal barrier.

Safety Issues Addressed:

a. Loss of offsite power.

b. Potential RXCP seal LOCA following a loss of offsite power.

Plant Specific Nature of the Sequence:

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

Modeling Assumptions:

Credit is taken for:

a. Emergency AC power to at least one of the two vital AC buses.

b. AFW successfully providing the heat removal function.

12. Sequence: SLO-18

Sequence Frequency: 1.17E-06/year
Contribution to Core Melt: 1.77%
Initiating Event Frequency: 5.12E-03
Conditional Core Melt Frequency: 2.28E-04

Sequence Description:

A small LOCA event (a break ranging in size from 0.375 inches and 2 inches equivalent diameter) occurs, leading to a reactor trip and safety injection actuation. High pressure and the low pressure safety injection fail to operate to meet their success criteria. Operators follow emergency operating procedure ES-1.2, "Post LOCA Cooldown and Depressurization", and provide the cooldown and depressurization necessary to allow the accumulator on this intact loop to inject into the reactor coolant system RCS cold leg.

Core melt is postulated due to:

a. Loss of high pressure injection and;

b. Loss of low pressure injection.

Safety Issues Addressed:

a. Small LOCA

b. Failure of the high pressure injection function following a small LOCA

c. Failure of the low pressurization function following a small LOCA

Plant Specific Nature of the Sequence:

This event sequence is a low frequency sequence considered typical for currently operating PWRs. There is no additional failure mode observed for the sequence.

Modeling Assumptions:

a. Credit has been taken for the initial operator action to cooldown and depressurize the RCS.

b. Credit has been taken for accumulator injection.

13. Sequence: LSP-6

Sequence Frequency: 1.09E-06/year
Contribution to Core Melt: 1.65%
Initiating Event Frequency: 4.36E-02
Conditional Core Melt Frequency: 2.5E-05

Sequence Description:

A loss of offsite power event occurs, leading to a reactor trip. The onsite emergency generators automatically start and restore power to at least one of two vital AC buses. The AFW system fails to provide the short term cooling function. The operators follow the emergency operating procedure and establish primary system bleed and feed with one of two high pressure SI trains providing flow to one of two RCS cold legs and one of two pressurizer PORVs open to the PRT. The long term cooling functions provided by the high pressure recirculation fails, however.

Core melt is postulate due to:

a. Loss of AFW and;

b. Loss of high pressure recirculation.

Safety Issues Addressed:

a. Loss of offsite power.

b. Failure of the secondary cooling function provided by AFW following a loss of offsite power.

c. Failure of the long term cooling function provided by high pressure recirculation following a loss of offsite power.

Plant Specific Nature of the Sequence:

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

Modeling Assumptions:

Credit is taken for:

a. Emergency AC power to at least one of the two vital AC buses.

b. Short term cooling function provided by primary bleed and feed is successful.

B. Internal Flooding Systemic Sequences

There are a total of eighteen core melt sequences quantified for internal flooding in the Kewaunee PRA study. These sequences are presented in summary on Table 3.4.4-5 of this report.

C. Other Important Systemic Sequences

No other functional sequences were found that were considered important contributors to either the core damage frequency or poor containment performance.

D. Systemic Sequences Due to Human Error Rates

Section 3.4.4 part B describes the sensitivity analysis for the operator actions included in the Kewaunee PRA model. In addition, a special sensitivity case was performed in order to evaluate the effect of recovery actions on sequences that are not reportable based upon the screening criteria.

The base case sequences are presented on table 3.4.4-4. The sensitivity case sequences are presented on table 3.4.2-2.

An evaluation of the results of this sensitivity study revealed that the station blackout (SBO) initiator was affected. The core melt frequency for this initiator increased from 2.64E-05 to 3.37E-05. This was due to the fact that there are three recovery actions associated with SBO.

Two sequences moved above the limit for reportability. Sequence 30 in the base case (table 3.4.4-4) moved up to become sequence 15 in the sensitivity case (table 3.4.2-2). Sequence 39 in the base case (table 3.4.4-4) moved up to become sequence 23 in the sensitivity case (table 3.4.2-2).

Table 3.4.2-3 presents timing and complexity information associated with the recovery actions defined in section 3.4.4.

3.4.3 Vulnerability Screening

A. Vulnerability Criteria

WPSC defines a vulnerability as a feature in plant design, procedures, training, etc., which results in a contribution to core melt risk greater than what is expected.

Vulnerability identification is an integral part of the fault tree and core melt quantification process. The results of the quantification process are reviewed continuously for the purpose of identifying vulnerabilities. Particular attention is given to those sequences that fell within the screening criteria defined in section 3.4.1 of this report.

B. WPSC Identified Vulnerabilities

The following items represent the identified vulnerabilities for Kewaunee. Section 6 of this report describes the WPSC response to these vulnerabilities.

1. Vulnerabilities were discovered from the initial core melt quantification for the interfacing systems LOCA event (ISL). The first involves the normal operating position of motor operated valves SI-302A and SI-302B. These valves were open during normal plant operation as well as loss of coolant accident (LOCA) events. These valves are located in low pressure safety injection lines which are connected to the reactor coolant system. This configuration provides an ISL path during normal operation which was a major contributor to the core damage frequency for an ISL event.

   The second vulnerability associated with the ISL event was associated with ISL lines with four pressure isolation valves (RHR-1A/1B and SI-13A/B) that were not leak tested and therefore represented a major contribution to the core damage frequency for the ISL event. Motor operated valves RHR-1A and RHR-1B are the inlet valves from the reactor coolant system loops to the suction of the residual heat removal (RHR) pumps. Check valves SI-13A and SI-13B are located in the high pressure safety injection lines to the reactor coolant system cold legs.

2. A vulnerability was defined that related to procedural inadequacies associated with an Interfacing System LOCA (ISL) event. The event sequence modeled for ISL involves operator actions, and therefore, some human error is modeled in the PRA. Several scenarios were considered, and it was determined that the most limiting ISL scenario involves a failure of the residual heat removal (RHR) pump suction valves. When modeling this sequence, it was determimed that procedural guidance (ECA 1.2) for determining where the LOCA was occurring was not complete. This came as a result of the guidance provided for developing this procedure being based upon a generic PRA for a typical Westinghouse PWR plant.

3. During the analysis for internal flooding, it was determined that there was a potential for significant flooding propagation from the turbine building basement to the adjoining areas which contain safeguards equipment. The propagation was due to the assumption that doors that swing out of the affected room cannot

withstand the flooding forces and therefore fail. The effected doors are doors 4, 6 and 401.

4. It was also determined during the analysis for internal flooding that a major flooding event could occur as a result of the failure of a circulating water expansion joint at the main condenser. During the evaluation of this flooding source it was determined that routine inspections that could accurately assess the material condition of these expansion joints were not conducted.

5. An evaluation of the loss of offsite power (LSP) and station blackout (SBO) events indicated that the instrument air system is not as reliable as it could be. This is due to the fact that three of the six air compressors are unavailable due to the initiators. This makes the remaining three air compressors very important. Two of the three air compressors (B and C) are supplied with power from vital motor control centers MCC-52A and MCC-62A respectively. Air compressor A is supplied by a swing motor control center MCC-5262 which can be supplied by either safeguard bus 52 or bus 62. MCC-5262 is normally aligned to bus 52 and requires local operator action to switch the MCC-5262 to bus 62 if bus 52 is unavailable. A review of plant operating procedures relating to the LSP and SBO events found that they do not contain procedural steps for maintaining MCC-5262 energized so as to ensure power to at least two air compressors.

6. A vulnerability associated with the auxiliary feedwater system (AFW) was discovered during the development of AFW system fault trees. A diversion path exists that diverts condensate from the condensate storage tanks to the main condenser and therefore reduces the quantity available to the AFW pumps for secondary cooling. The path is created as a result of the failure mode associated with the makeup valve (MU-3A) to the condenser. This valve fails open on loss of instrument air or control power. If the operator fails to isolate this line, then the success of AFW in providing secondary heat removal is adversely effected. This recovery action contributes approximately 9% to the total core damage frequency.

7. Section 3.4.4 of this report presents the relative importance of the various system components in terms of their contribution to core damage. From this analysis it was determined that the auxiliary feedwater system contributed approximately 32% to the total core damage frequency. Approximately 21% is directly related to the reliability of the turbine driven auxiliary feedwater pump.

8. Another vulnerability is related to the overall availability of the station and instrument air system. Air compressors D and E are subject to frequent outages for corrective maintenance and, therefore, make the system less reliable.

9. A vulnerability was discovered associated with the charging pumps that affects the ability of these pumps to provide reactor coolant system makeup as well as reactor coolant pump seal injection for seal cooling. The specific concern is that there have been numerous cases in which a charging pump discharge relief valve opens and diverts charging pump flow back to the volume control tank.

3.4.4 Sensitivity and Importance Analysis

Once the dominant accident sequences leading to the core melt frequency were screened to determine the important individual contributors to core damage, sensitivity studies were conducted. Items considered in these studies include possible operator recovery actions, dominant core melt sequences contributing to the core melt frequency, and possible design changes that could be made to decrease the core melt frequency. Sensitivity studies were conducted on initiating event frequencies, operator actions, risk modeling, and plant design. Each sensitivity study was conducted by varying only one influence factor and holding all other factors constant. At the conclusion of the individual sensitivity studies, those factors having the most impact on the core melt frequency were combined, and a study was conducted to determine the overall impact on the core melt frequency.

Dominant initiating event frequencies were changed by reviewing system configurations, other data bases, possible alternatives that would prevent frequent occurrence of the initiating event, and the like, to determine a range of values for the frequency so that the variability of the core melt frequency could be assessed. Risk modeling sensitivities included changes to systems' success criteria, analysis assumptions, and other modeling criteria, while design alternative studies included conceptual changes to systems whose failure is a dominant contributor to the core melt frequency.

Sensitivity evaluations were performed for operator actions, common cause, test and maintenance as well as for certain system components. The evaluations were performed to determine the global effect of the parameters of interest. Failure rates were increased/reduced by a factor of 5 where a higher level of uncertainty and variability exists such as human reliability, common cause and test and maintenance unavailability. Failure rates for system components were increased/reduced by a factor of 2 since actual plant data was used and there was less uncertainty associated with these parameters.

The following sub-sections describe the various sensitivity evaluations which were performed for the Kewaunee IPE study.

A. Sensitivity Analysis for Cutoff

1. Background and Objectives

During the Kewaunee PRA project, the plant core melt frequency for internal initiating events (the sixteen categories defined in section 3.1.1 of this report) was quantified with certain probability cut-off limitations in the three stages of the core melt quantification:

a. The fault tree models were quantified with cut-off probabilities as low as practical. Generally, the objective was to quantify all fault trees with a cut-off probability of 1.0E-12 unless the code limitations or a resulting excessive number of cutsets caused this cut-off to be increased.

   In the present sensitivity analysis, the cut-off probabilities associated with fault tree quantification were not changed, thus the cut-off probabilities used for both the base case and the sensitivity analysis case are the same. These cutoff-off probabilities are used by the WESCUT and WESLGE codes.

b. The fault tree cutset files were "reduced" (i.e. the subtrees are linked into the cutsets during the reduction phase). This step precedes the accident sequence quantification. During this reduction process, another set of cut-off probabilities was used for the WLINK code to limit the total number of cutsets.

   In this reduction process, the cut-off probability is preferably kept at 1.0E-12, whenever possible (e.g. code limitations and excessive number of resulting cutsets force some of the cut-off probabilities to be increased). This applied to all sequences except Loss of Offsite Power (LSP).

   In this reduction process, the cut-off probability for the LSP sequence was preferably kept at 1.0E-09, whenever possible (e.g. code limitations and excessive number of resulting cutsets force some of the cut-off probabilities to be increased).

c. Finally, the accident sequence linking was done for the fifteen initiating event categories (except LSP) at a cutoff probability of 1.0E-10 and separately for LSP at a cut-off of 1.0E-9.

Thus, the objectives of this sensitivity analysis were to:

- Perform a reduction process for all initiating events at a cut-off probability of 1.0E-12, and

- Perform accident sequence linking for all initiating events at an order of magnitude lower than the base case: e.g.

  the fifteen initiating events were quantified at a cut-off probability of 1.0E-11 and

LSP initiating event was quantified at a cut-off probability of 1.0E-10.

2. Sensitivity Case

The sensitivity case was run on a Unix Workstation (IBM RS6000) which allows the WLINK Code System to process a larger number of cutsets (100,000 versus 9900 on an IBM PC). The sensitivity case was run with the following cut-off values.

a. The fault tree cut-off probabilities were not changed.

b. All cutset files were reduced at 1.0E-12 cut-off probability.

c. LSP accident sequences were linked at 1.0E-10 and the remaining 15 events were linked at 1.0E-11 cut-off probability, thus gaining an order of magnitude reduction with respect to the base case.

The sensitivity analysis results are presented in a report from the Westinghouse Electric Corporation Product Risk Analysis group.

3. Comparison

A comparison of the sensitivity case results with the base case results shows the following:

a. In the sensitivity analysis, an additional 10,000 cutsets were obtained.

b. The total plant core melt frequency increased by 1.6% in the sensitivity run. This shows that the base case is a very good representation of the total plant core melt frequency and it is not sensitive to an order of magnitude decrease in cut-off probability in running WLINK.

c. The largest core melt frequency increase is observed in the LSP initiating event category (19% increase). This increase was expected since the category is run with E-09 cut-off probability in the base case (as opposed to E-10). Thus, it is more sensitive to a change in the cut-off probability. However, an examination of the accident sequence frequencies associated with LSP shows that the individual accident sequence classification was not effected (namely no sequence moved from E-06 to E-05, or E-07 to E-06, etc. range)

Thus the overall conclusion of this sensitivity run is that the base case core melt frequency is not sensitive to an order of magnitude decrease in the cut-off probability.

B. Sensitivity Analysis for HRA

1. All Operator Actions Successful

A sensitivity analysis was performed in which it was assumed that all operator actions were successful. The results show that some sequences are very sensitive to human reliability failure rates. It also shows that if all operator actions were successful an improvement of 25% (4.93E-05/year) in total core melt frequency would be realized.

2. Operator Actions Failure Rates

Two sensitivity analysis cases were performed in which all human error probabilities were increased/reduced by a factor of five. The analysis in which human error probabilities were reduced by a factor of five produced nearly the same results as the analysis where all operator actions were successful.

The case in which human error probabilities were increased by a factor of five resulted in a total core melt frequency increase by a factor of three (1.98E-04) and is a reasonable average error factor for this parameter. This analysis had the greatest effect on the transients with main feedwater (TRA) and the loss of instrument air (INA) initiators whose core melt frequencies increased by factors of 3.91 and 4.64 respectively.

An inspection of the dominant core melt sequences revealed only two significant sequences. These sequences were both associated with the steam generator tube rupture (SGR) initiator which is dominated by human error probabilities. The quantification of the human error probabilities associated with these two sequences were evaluated to determine their validity. The evaluation revealed that detailed procedures exist for these operator actions and that the quantification of these probabilities was accurate and consistent with human reliability analysis guidelines.

3. Sensitivity for Recovery Actions

The NRC guidance document for preparing the IPE submittal[4] provides screening criteria for accident sequence reportability. The guidance document also requests that licensees identify and report any sequence that drops below the core damage frequency criteria because the frequency has been reduced by more than an order of magnitude by credit taken for human recovery actions. The NRC also requests information on the timing and complexity of the postulated recovery actions.

As a result of the requirements described above it is necessary to expand the sensitivity analysis for HRA to address the additional reporting requirements. To that end, an analysis was performed which included the following steps:

a. A working definition for recovery actions was created to enable the classification of all operator actions modeled in the PRA study.

   The following definition for recovery actions was developed using the examples presented in Appendix 5 of Generic Letter 88-20.[5]

   A recovery action is defined as those actions that the operators perform as a result of a system or component not performing as expected in response to plant emergency conditions. Generally, recovery actions are performed outside of the control room. However, if a control room action is unproceduralized or is not a relatively easy task or cannot be completed in a short time it would also be considered a recovery action. Also actions explicitly addressed in the EOPs are not considered recovery actions (e.g. ATWS, MFW after AFW fails and SI recirculation). Some examples of recovery actions are:

   - Recovering offsite power
   - Repairing local electrical or mechanical faults associated with plant systems or components
   - Actuating safety systems manually outside of the control room
   - Establishing auxiliary feedwater, main feedwater or steam dump flow paths locally
   - Local manual operation of failed remotely operated valves

b. The second step in this analysis was to screen all operator actions modeled in the PRA study to define the recovery actions that are included in this analysis. Table 3.4.4-2 presents the results of this screening process.

c. The next step in this analysis was to perform a complete core melt quantification that included a requantification of all fault trees. For this quantification, the probabilities for those actions classified as recovery actions were increased by an order of magnitude. The results of this quantification identifying those sequences which now fell within the criteria for reportability can be found in section 3.4.2 of this report.

d. Timing and complexity information is presented in section 3.4.2 of this report for all recovery actions modeled in the PRA study.

C. Sensitivity Analysis for Common Cause

A sensitivity analysis was performed in which it was assumed that no common cause failures could occur. The results show that some sequences are sensitive to common cause failures. The most significant effect was on the loss of service water system initiator (SWS) in which an order of magnitude increase in the core melt frequency was experienced. The analysis also showed that if no common cause failures were possible, an improvement of 25% (4.91E-05) in total core melt frequency would be realized.

Two sensitivity analysis cases were performed in which all common cause probabilities were increased/decreased by a factor of five. The analysis in which the probabilities were reduced by a factor of five produced results similar to the case in which no common cause failure occurred.

The case in which common cause probabilities were increased by a factor of five resulted in a total core melt frequency increase by a factor of two (1.51E-04) and is considered the upper bound for this parameter. This analysis had the greatest effect on the loss of instrument air system (INA), loss of service water system (SWS) and the transients without main feedwater (TRS) initiators whose frequencies increased by a factor of 5.1, 5.55 and 11 respectively.

D. Sensitivity for Component Failure rates

A sensitivity analysis was performed in which certain component failure rates were increased to determine the overall effect on total core melt frequency as well as the individual initiators. The following sub-sections describe the various component failure rate sensitivity evaluations that were performed.

1. Air Operated Valve Sensitivity

The failure rates for air operated valves were quantified using pooled data for a large number of air operated valves in the plant. The resultant failure rate was used for all air operated valves in the PRA model and, therefore, does not reflect the operating experience for any specific valve.

In this sensitivity case the failure probabilities for all air operated valves were increased by a factor of two. The results show that the PRA model is not sensitive to this parameter. The greatest effect was on the loss of instrument air system initiator (INA) whose core melt frequency increased by a factor of two. The effect on the total core melt frequency was on the order of a 10% increase (7.28E-05).

2. Motor Operated Valve Sensitivity

The failure rates for motor operated valves were quantified using pooled data for a large number of motor operated valves in the plant. The resultant failure rate was used for all motor operated valves in the PRA model and therefore does not reflect the operating experience for any specific valve.

In this sensitivity case, the failure probabilities for all motor operated valves were increased by a factor of two. The results show that the model is sensitive to individual motor valve failure rates. The effect on the total core melt frequency was on the order of a 24% increase (8.23E-05).

3. Diesel Generator Reliability

This sensitivity analysis was performed to verify what was currently known as well as to determine the overall effect of this parameter.

In this sensitivity case, the failure probabilities including common cause were increased by a factor of two. The results were, as expected, that the PRA model is sensitive to this parameter. The total core melt frequency increased by 24% (8.23E-05). The loss of offsite power (LSP) and the station blackout (SBO) initiators were the most sensitive with an increase of 14% and 56% in the core melt frequency for these initiators respectively due to the dependence on the diesel generators for success.

E. Sensitivity for Test and Maintenance Unavailabilities

This analysis was performed to define an upper bound for this parameter, due to the large uncertainty associated with it.

In this sensitivity case, the calculated test and maintenance unavailabilities were increased by a factor of five. The results show that overall the PRA model is not sensitive to this parameter. The total core melt frequency increased by 12% (7.4E-05).

F. Sensitivity for Initiating Event Frequencies

1. Internal Events IE Frequencies

Table 3.4.4-1 presents the IE frequencies for the internal events selected for analysis. The frequency for five initiators (INA, SWS, TDC, CCS and ISL) was calculated using the results of a detailed analysis and would have a higher level of uncertainty associated with them. An examination of Table 3.4.4-4 shows that only INA initiating event category has any appreciable contribution to the total plant core melt frequency. If INA initiatng event frequency is increased by a factor of 10, then its contribution to plant core melt will increase to 2.08E-05/year, and the total plant core melt frequency will become 8.5E-05/year (a 28% increase).

2. Internal Flooding IE Frequencies

Table 3.4.4-3 presents the IE frequencies for the internal flooding events selected for analysis. The frequencies for the six internal flooding initiators were conservative rather than best estimate values and therefore uncertainty/sensitivity issues need not be addressed.

G. Importance Analysis

The COMPLNK Code was used to calculate the relative importance ranking of the basic events that make up the total core melt frequency. The ranking is based upon the Fussel-Vessly algorithm. The code uses the cutset information generated from WLINK and transforms it into a file that lists the components in a fault tree or in the accident sequences in descending order of their contribution to the total failure probability. It should be noted that the mathematical total may be greater than 100%. The results of the various importance calculations are presented below.

1. Importance by Initiator

Tables 3.4.4-1 and 3.4.4-3 present the results of the importance calculation performed for all initiators considered in the PRA study.

The results of the importance calculation for internal events Table 3.4.4-1 shows that the first four events (SBO, SLO, MLO and SGR) contribute approximately 80% to the total core melt frequency.

The results of the importance calculation for internal flooding events Table 3.4.4-3 shows that the first two events (FL3 and FL4) contribute approximately 99% to the total core melt frequency for internal flooding. These are associated with flooding events occurring in diesel generator rooms A and B respectively.

2. Importance By Core Melt Sequence

Tables 3.4.4-4 and 3.4.4-5 present the results of the importance calculations performed for all sequences considered in the PRA study.

The results of the importance calculation for internal events Table 3.4.4-4 shows that the first six sequences contribute approximately 75% to the total core melt frequency.

The results of the importance calculation for the internal flooding events Table 3.4.4-5 shows that the first two sequences contribute approximately 97% to the total core melt frequency for internal flooding.

3. Importance By Cutset

Tables 3.4.4-6 and 3.4.4-7 present the results of the importance calculations performed for the dominant cutsets in the PRA study.

The results of the importance calculation for internal events Table 3.4.4-6 shows that the first twelve cutsets contribute approximately 63% to the total core melt frequency.

The results of the importance calculation for internal flooding Table 3.4.4-7 shows that the first eight cutsets contribute approximately 72% to the total core melt frequency for internal flooding.

4. Importance By Component

The results of the importance calculation for internal events shows that the auxiliary feedwater systems failure contributed approximately 32% to the total core melt frequency. Of this total 24% is related to the reliability of the AFW pumps and another 9% to an AFW system recovery action.

The results of the importance calculation for internal flooding events shows that a flooding event in diesel generator room A (in which all train A components are unavailable) results in an increase in importance for train B components.

3.4.5 Decay Heat Removal Evaluation

The only defined vulnerabilities are those described in section 3.4.3 relative to the reliability of the auxiliary feedwater system. A complete evaluation of the decay heat removal capability as well as the possible resolution of USI A-45 will be considered as part of the IPE for external events which include seismic events.

3.4.6 USI and GSI Screening

The IPE was presented as a logical means for resolving any currently Unresolved Safety Issues (USI) and Generic Safety Issues (GSI) that may apply to the specific plant. WPSC is not addressing any of these issues as part of the IPE study.

TABLE 3.4.2-1 CORE MELT SEQUENCES - TOP 13 SEQUENCES

	SEQUENCE PROBABILITY	PERCENT CONTRIB	SEQUENCE DESCRIPTION				SEQUENCE IDENTIFIER
1	1.30E-05	19.57	STATION BLACKOUT	INITIATING EVENT	OCCURS		IEV-SBO
			TURBINE DRIVEN	AFW PUMP FAILS			SYS-AF2
			POWER NOT	RESTORED	IN 2 HOURS		AC2-FAIL
2	1.21E-05	18.21	SMALL LOCA	INITIATING EVENT	OCCURS		IEV-SLO
			COOLDOWN AND	DEPRESSURIZATION	FOR CHARGING	FLOW FAILS	SYS-ES1
			HIGH PRESSURE	RECIRCULATION	FAILS		SYS-HR1
			LOW PRESSURE	RECIRCULATION	FAILS		SYS-LR2
3	8.70E-06	13.13	STATION BLACKOUT	INITIATING EVENT	OCCURS		IEV-SBO
			POWER NOT	RESTORED	IN 11 HOURS		ACX-11
4	7.42E-06	11.19	MEDIUM LOCA	INITIATING EVENT	OCCURS		IEV-MLO
			HIGH PRESSURE	RECIRCULATION	FAILS		SYS-HRO
			LOW PRESSURE	RECIRCULATION	FAILS		SYS-LR2
5	4.31E-06	6.51	STEAM GENERATOR	TUBE RUPTURE	INITIATING EVENT	OCCURS	IEV-SGR
			COOLDOWN AND	DEPRESSURIZATION	BEFORE SG	OVERFILL FAILS	SYS-OS1
			STEAM GENERATOR	INTEGRITY FAILS			SSV-FAIL
			COOLDOWN AND	DEPRESSURIZATION	USING ECA-3.1,2	FAILS	SYS-EC4
6	4.01E-06	6.06	STATION BLACKOUT	INITIATING EVENT	OCCURS		IEV-SBO
			OFF-SITE POWER	RESTORED			REC-OSP
			CHARGING FOR	SEAL INJECTION	FAILS		SYS-CHB
			CORE UNCOVERED	BY RXCP SEAL	LOCA		CCV-11
7	2.69E-06	4.06	TRANSIENT WITH	MAIN FEEDWATER	INITIATING EVENT	OCCURS	IEV-TRA
			SAFEGUARDS POWER	AVAILABLE			DEL-OSP
			AFW SYSTEM	FAILS			SYS-AF3
			MFW SYSTEM	FAILS			SYS-OM2
			BLEED AND FEED	FAILS			SYS-OB2
8	2.17E-06	3.27	LOSS OF OFFSITE	POWER INITIATING	EVENT OCCURS		IEV-LSP
			SAFEGUARDS POWER	AVAILABLE			DEL-OSP
			AFW SYSTEM	FAILS			SYS-AF3
			BLEED AND FEED	FAILS			SYS-OB5
9	2.08E-06	3.14	LOSS OF	INSTRUMENT AIR	INITIATING EVENT	OCCURS	IEV-INA
			LOSS OF	INSTRUMENT AIR	IE FAULT TREE		SYS-IAIE
			SAFEGUARDS POWER	AVAILABLE			DEL-OSP
			AFW SYSTEM	FAILS			SYS-AF5
			MFW SYSTEM	FAILS			SYS-OM3
10	1.39E-06	2.09	LARGE LOCA	INITIATING EVENT	OCCURS		IEV-LLO
			LOW PRESSURE	RECIRCULATION	FAILS		SYS-LR1

TABLE 3.4.2-1 CORE MELT SEQUENCES - TOP 13 SEQUENCES

NUMBER	SEQUENCE PROBABILITY	PERCENT CONTRIB	SEQUENCE DESCRIPTION			SEQUENCE IDENTIFIER
11	1.23E-06	1.86	LOSS OF OFFSITE	POWER INITIATING	EVENT OCCURS	IEV-LSP
			SAFEGUARDS POWER	AVAILABLE		DEL-OSP
			CHARGING SYSTEM	FAILS		SYS-CHG
			COMPONENT	COOLING WATER	SYSTEM FAILS	SYS-CCW
12	1.17E-06	1.77	SMALL LOCA	INITIATING EVENT	OCCURS	IEV-SLO
			HIGH PRESSURE	INJECTION	FAILS	SYS-HI2
			LOW PRESSURE	INJECTION	FAILS	SYS-LI2
13	1.09E-06	1.65	LOSS OF OFFSITE	POWER INITIATING	EVENT OCCURS	IEV-LSP
			SAFEGUARDS POWER	AVAILABLE		DEL-OSP
			AFW SYSTEM	FAILS		SYS-AF3
			HIGH PRESSURE	RECIRCULATION	FAILS	SYS-HR1

TABLE 3.4.2-2 SEQUENCES DUE TO HUMAN ERROR RATES FOR RECOVERY ACTIONS - ALL SEQUENCES

NUMBER	SEQUENCE PROBABILITY	PERCENT CONTRIB	SEQUENCE DESCRIPTION	SEQUENCE IDENTIFIER
1	1.30E-05	17.61	STATION BLACKOUT INITIATING EVENT OCCURS	IEV-SBO
			TURBINE DRIVEN AFW PUMP FAILS	SYS-AF2
			POWER NOT RESTORED IN 2 HOURS	AC2-FAIL
2	1.21E-05	16.39	SMALL LOCA INITIATING EVENT OCCURS	IEV-SLO
			COOLDOWN AND DEPRESSURIZATION FOR CHARGING FLOW FAILS	SYS-ES1
			HIGH PRESSURE RECIRCULATION FAILS	SYS-HR1
			LOW PRESSURE RECIRCULATION FAILS	SYS-LR2
3	9.56E-06	12.98	STATION BLACKOUT INITIATING EVENT OCCURS	IEV-SBO
			OFF-SITE POWER RESTORED	REC-OSP
			CHARGING FOR SEAL INJECTION FAILS	SYS-CHB
			CORE UNCOVERED BY RXCP SEAL LOCA	CCV-11
4	8.70E-06	11.81	STATION BLACKOUT INITIATING EVENT OCCURS	IEV-SBO
			POWER NOT RESTORED IN 11 HOURS	ACX-11
5	7.42E-06	10.07	MEDIUM LOCA INITIATING EVENT OCCURS	IEV-MLO
			HIGH PRESSURE RECIRCULATION FAILS	SYS-HRO
			LOW PRESSURE RECIRCULATION FAILS	SYS-LR2
6	4.31E-06	5.86	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT OCCURS	IEV-SGR
			COOLDOWN AND DEPRESSURIZATION BEFORE SG OVERFILL FAILS	SYS-OS1
			STEAM GENERATOR INTEGRITY FAILS	SSV-FAIL
			COOLDOWN AND DEPRESSURIZATION USING ECA-3.1,2 FAILS	SYS-EC4
7	2.69E-06	3.65	TRANSIENT WITH MAIN FEEDWATER INITIATING EVENT OCCURS	IEV-TRA
			SAFEGUARDS POWER AVAILABLE	DEL-OSP
			AFW SYSTEM FAILS	SYS-AF3
			MFW SYSTEM FAILS	SYS-OM2
			BLEED AND FEED FAILS	SYS-OB2
8	2.17E-06	2.94	LOSS OF OFFSITE POWER INITIATING EVENT OCCURS	IEV-LSP
			SAFEGUARDS POWER AVAILABLE	DEL-OSP
			AFW SYSTEM FAILS	SYS-AF3
			BLEED AND FEED FAILS	SYS-OB5
9	2.08E-06	2.83	LOSS OF INSTRUMENT AIR INITIATING EVENT OCCURS	IEV-INA
			LOSS OF INSTRUMENT AIR IE FAULT TREE	SYS-IAIE
			SAFEGUARDS POWER AVAILABLE	DEL-OSP
			AFW SYSTEM FAILS	SYS-AF5
			MFW SYSTEM FAILS	SYS-OM3
10	1.39E-06	1.88	LARGE LOCA INITIATING EVENT OCCURS	IEV-LLO
			LOW PRESSURE RECIRCULATION FAILS	SYS-LR1

TABLE 3.4.2-2 SEQUENCES DUE TO HUMAN ERROR RATES FOR RECOVERY ACTIONS - ALL SEQUENCES

NUMBER	SEQUENCE PROBABILITY	PERCENT CONTRIB	SEQUENCE DESCRIPTION	SEQUENCE IDENTIFIER
11	1.23E-06	1.67	LOSS OF OFFSITE POWER INITIATING EVENT OCCURS	IEV-LSP
			SAFEGUARDS POWER AVAILABLE	DEL-OSP
			CHARGING SYSTEM FAILS	SYS-CHG
			COMPONENT COOLING WATER SYSTEM FAILS	SYS-COW
12	1.17E-06	1.59	SMALL LOCA INITIATING EVENT OCCURS	IEV-SLO
			HIGH PRESSURE INJECTION FAILS	SYS-HI2
			LOW PRESSURE INJECTION FAILS	SYS-LI2
13	1.09E-06	1.48	LOSS OF OFFSITE POWER INITIATING EVENT OCCURS	IEV-LSP
			SAFEGUARDS POWER AVAILABLE	DEL-OSP
			AFW SYSTEM FAILS	SYS-AF3
			HIGH PRESSURE RECIRCULATION FAILS	SYS-HR1
14	7.22E-07	.98	STATION BLACKOUT INITIATING EVENT OCCURS	IEV-SBO
			OFF-SITE POWER RESTORED	REC-OSP
			CHARGING FOR SEAL INJECTION FAILS	SYS-CHB
			HIGH PRESSURE RECIRCULATION FAILS	SYS-HR1
15	6.02E-07	.82	STATION BLACKOUT INITIATING EVENT OCCURS	IEV-SBO
			RCS COOLDOWN FAILS	SYS-OCD
			POWER NOT RESTORED IN 9 HOURS	ACX-9
16	6.02E-07	.82	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT OCCURS	IEV-SGR
			AFW SYSTEM FAILS	SYS-AF1
			MFW SYSTEM FAILS	SYS-OM1
17	5.36E-07	.73	MEDIUM LOCA INITIATING EVENT OCCURS	IEV-MLO
			HIGH PRESSURE INJECTION FAILS	SYS-HI0
			LOW PRESSURE INJECTION FAILS	SYS-LI2
18	4.57E-07	.62	LARGE LOCA INITIATING EVENT OCCURS	IEV-LLO
			LOW PRESSURE INJECTION FAILS	SYS-LI1
19	4.28E-07	.58	STATION BLACKOUT INITIATING EVENT OCCURS	IEV-SBO
			OFF-SITE POWER RESTORED	REC-OSP
			CHARGING FOR SEAL INJECTION FAILS	SYS-CHB
			TURBINE DRIVEN AFW PUMP FAILS	SYS-AF2
			CORE UNCOVERED BY RXCP SEAL LOCA	CCV-2
20	4.21E-07	.57	LOSS OF SERVICE WATER INITIATING EVENT OCCURS	IEV-SWS
			LOSS OF SERVICE WATER IE FAULT TREE	SYS-SWIE
			SAFEGUARDS POWER AVAILABLE	DEL-OSP
			AFW SYSTEM FAILS	SYS-AF6
21	4.10E-07	.56	STATION BLACKOUT INITIATING EVENT OCCURS	IEV-SBO
			OFF-SITE POWER RESTORED	REC-OSP
			CHARGING FOR SEAL INJECTION FAILS	SYS-CHB
			RCS INVENTORY RESTORATION FAILS	SYS-ORI

TABLE 4.4.2-2 SEQUENCES DUE TO HUMAN ERROR RATES FOR RECOVERY ACTIONS - ALL SEQUENCES

NUMBER	SEQUENCE PROBABILITY	PERCENT CONTRIB	SEQUENCE DESCRIPTION				SEQUENCE IDENTIFIER
22	3.66E-07	.50	SMALL LOCA	INITIATING EVENT	OCCURS		IEV-SLO
			HIGH PRESSURE	INJECTION	FAILS		SYS-HI2
			LOW PRESSURE	RECIRCULATION	FAILS		SYS-LR1
23	3.47E-07	.47	STATION BLACKOUT	INITIATING EVENT	OCCURS		IEV-SBO
			OFF-SITE POWER	RESTORED			REC-OSP
			CHARGING FOR	SEAL INJECTION	FAILS		SYS-CHB
			RCS COOLDOWN	FAILS			SYS-OCD
			CORE UNCOVERED	BY RXCP SEAL	LOCA		CCV-9
24	3.00E-07	.41	VESSEL FAILURE	INITIATING	EVENT OCCURS		IEV-VEF
25	2.57E-07	.35	TRANSIENT	WITHOUT MAIN	FEEDWATER	EVENT OCCURS	IEV-TRS
			SAFEGUARDS POWER	AVAILABLE			DEL-OSP
			AFW SYSTEM	FAILS			SYS-AF3
			BLEED AND FEED	FAILS			SYS-OB2
26	2.11E-07	.29	LOSS OF DC BUS	INITIATING EVENT	OCCURS		IEV-TDC
			LOSS OF DC BUS	IE FAULT TREE			SYS-DBIE
			AFW SYSTEM	FAILS			SYS-AF4
			MFW SYSTEM	FAILS			SYS-OM4
			BLEED AND FEED	FAILS			SYS-OB3
[illegible]	2.01E-07	.27	TRANSIENT	WITHOUT MAIN	FEEDWATER	EVENT OCCURS	IEV-TRS
			SAFEGUARDS POWER	AVAILABLE			DEL-OSP
			AFW SYSTEM	FAILS			SYS-AF3
			HIGH PRESSURE	RECIRCULATION	FAILS		SYS-HR1
28	1.93E-07	.26	STEAM GENERATOR	TUBE RUPTURE	INITIATING EVENT	OCCURS	IEV-SGR
			HIGH PRESSURE	INJECTION	FAILS		SYS-HI1
			COOLDOWN AND	DEPRESSURIZATION	BEFORE SG	OVERFILL FAILS	SYS-OS1
			STEAM GENERATOR	INTEGRITY FAILS			SSV-FAIL
29	1.61E-07	.22	MEDIUM LOCA	INITIATING EVENT	OCCURS		IEV-MLO
			HIGH PRESSURE	INJECTION	FAILS		SYS-HI0
			LOW PRESSURE	RECIRCULATION	FAILS		SYS-LR1
30	1.19E-07	.16	STEAM GENERATOR	TUBE RUPTURE	INITIATING EVENT	OCCURS	IEV-SGR
			STEAM GENERATOR	ISOLATION	FAILS		SYS-ISO
			COOLDOWN AND	DEPRESSURIZATION	USING ECA-3.1,2	FAILS	SYS-EC3
31	1.00E-07	.14	LARGE LOCA	INITIATING EVENT	OCCURS		IEV-LLO
			ACCUMULATOR	INJECTION	FAILS		SYS-ACC
32	5.96E-08	.08	STEAM GENERATOR	TUBE RUPTURE	INITIATING EVENT	OCCURS	IEV-SGR
			COOLDOWN AND	DEPRESSURIZATION	BEFORE SG	OVERFILL FAILS	SYS-OS1
			STEAM GENERATOR	INTEGRITY	MAINTAINED		SSV-SUC
			COOLDOWN AND	DEPRESSURIZATION	AFTER SG	OVERFILL FAILS	SYS-OS2

TABLE 4.4.2-2 SEQUENCES DUE TO HUMAN ERROR RATES FOR RECOVERY ACTIONS - ALL SEQUENCES

NUMBER	SEQUENCE PROBABILITY	PERCENT CONTRIB	SEQUENCE DESCRIPTION				SEQUENCE IDENTIFIER
33	4.85E-08	.07	STEAMLINE BREAK	INITIATING EVENT	OCCURS		IEV-SLB
			REACTOR POWER	BELOW 10%			PWR-LOW
			MAIN STEAM	ISOLATION	FAILS		SYS-IS1
			HIGH PRESSURE	INJECTION	FAILS		SYS-HI3
34	4.40E-08	.06	ATWS WITHOUT	MAIN FEEDWATER	OCCURS		IEV-AWS
			REACTOR TRIP	BREAKERS	FAIL		AFM-BREAKER
			PRIMARY	PRESSURE RELIEF	FAILS		SYS-PPR
35	3.45E-08	.05	TRANSIENT WITH	MAIN FEEDWATER	INITIATING EVENT	OCCURS	IEV-TRA
			SAFEGUARDS POWER	AVAILABLE			DEL-OSP
			CHARGING SYSTEM	FAILS			SYS-CHG
			COMPONENT	COOLING WATER	SYSTEM FAILS		SYS-CCW
36	2.73E-08	.04	STEAMLINE BREAK	INITIATING EVENT	OCCURS		IEV-SLB
			AFW SYSTEM	FAILS			SYS-AF1
			MFW SYSTEM	FAILS			SYS-OM1
			BLEED AND FEED	FAILS			SYS-OB4
37	2.45E-08	.03	STEAMLINE BREAK	INITIATING EVENT	OCCURS		IEV-SLB
			HIGH PRESSURE	INJECTION	FAILS		SYS-HI3
			AFW SYSTEM	FAILS			SYS-AF1
			MFW SYSTEM	FAILS			SYS-OM1
38	1.67E-08	.02	SMALL LOCA	INITIATING EVENT	OCCURS		IEV-SLO
			HIGH PRESSURE	INJECTION	FAILS		SYS-HI2
			COOLDOWN AND	DEPRESSURIZATION	FOR ACC AND LI2	FAILS	SYS-OP2
39	1.56E-08	.02	LOSS OF	COMPONENT	COOLING WATER	EVENT OCCURS	IEV-CCS
			LOSS OF	COMPONENT	COOLING WATER	IE FAULT TREE	SYS-CCWIE
			AT LEAST ONE	TRAIN OF SERVICE	WATER AVAILABLE		DEL-SW
			AFW SYSTEM	FAILS			SYS-AF3
			MFW SYSTEM	FAILS			SYS-OM2
40	1.39E-08	.02	TRANSIENT WITH	MAIN FEEDWATER	INITIATING EVENT	OCCURS	IEV-TRA
			SAFEGUARDS POWER	AVAILABLE			DEL-OSP
			AFW SYSTEM	FAILS			SYS-AF3
			MFW SYSTEM	FAILS			SYS-OM2
			HIGH PRESSURE	RECIRCULATION	FAILS		SYS-HR1
41	1.33E-08	.02	MEDIUM LOCA	INITIATING EVENT	OCCURS		IEV-MLO
			HIGH PRESSURE	INJECTION	FAILS		SYS-HI0
			COOLDOWN AND	DEPRESSURIZATION	FAILS		SYS-OP1
42	1.27E-08	.02	LOSS OF	COMPONENT	COOLING WATER	EVENT OCCURS	IEV-CCS
			LOSS OF	COMPONENT	COOLING WATER	IE FAULT TREE	SYS-CCWIE
			AT LEAST ONE	TRAIN OF SERVICE	WATER AVAILABLE		DEL-SW
			CHARGING SYSTEM	FAILS			SYS-CHG

TABLE 4.4.2-2 SEQUENCES DUE TO HUMAN ERROR RATES FOR RECOVERY ACTIONS - ALL SEQUENCES

NUMBER	SEQUENCE PROBABILITY	PERCENT CONTRIB	SEQUENCE DESCRIPTION	SEQUENCE IDENTIFIER
43	1.24E-08	.02	ATWS WITHOUT MAIN FEEDWATER OCCURS	IEV-AWS
			REACTOR TRIP BREAKERS FAIL	AFM-BREAKER
			AFW SYSTEM FAILS	SYS-AFG
44	7.40E-09	.01	INTERFACING SYSTEMS LOCA INITIATING EVENT OCCURS	IEV-ISL
			RHR PIPE FAILS	BR-PIPE
45	7.10E-09	.01	ATWS WITHOUT MAIN FEEDWATER OCCURS	IEV-AWS
			RODS FAIL TO INSERT	AFM-RODS
			PRIMARY PRESSURE RELIEF FAILS	SYS-PPR
46	6.84E-09	.01	INTERFACING SYSTEMS LOCA INITIATING EVENT OCCURS	IEV-ISL
			RHR PUMP SEAL FAILS	BR-SEAL
			OPERATOR FAILS TO ISOLATE RHR PUMPS	SYS-OIP
			OPERATOR FAILS TO THROTTLE SI FLOW	SYS-OSR
47	6.60E-09	.01	INTERFACING SYSTEMS LOCA INITIATING EVENT OCCURS	IEV-ISL
			RHR PUMP SEAL FAILS	BR-SEAL
			OPERATOR FAILS TO ISOLATE RHR PUMPS	SYS-OIP
			RHR RELIEF VALVES FAIL TO CLOSE	SYS-RVC
	3.73E-09	.01	INTERFACING SYSTEMS LOCA INITIATING EVENT OCCURS	IEV-ISL
			RHR PUMP SEAL FAILS	BR-SEAL
			HIGH PRESSURE INJECTION FAILS	SYS-HI4
49	3.31E-09	.00	SMALL LOCA INITIATING EVENT OCCURS	IEV-SLO
			AFW SYSTEM FAILS	SYS-AFO
			MFW SYSTEM FAILS	SYS-OMO
			BLEED AND FEED FAILS	SYS-OB1
50	2.44E-09	.00	SMALL LOCA INITIATING EVENT OCCURS	IEV-SLO
			HIGH PRESSURE INJECTION FAILS	SYS-HI2
			ACCUMULATOR INJECTION FAILS	SYS-ACC
51	2.22E-09	.00	ATWS WITHOUT MAIN FEEDWATER OCCURS	IEV-AWS
			RODS FAIL TO INSERT	AFM-RODS
			LONGTERM SHUTDOWN FAILS	SYS-LTS
52	1.56E-09	.00	INTERFACING SYSTEMS LOCA INITIATING EVENT OCCURS	IEV-ISL
			RHR PUMP SEAL FAILS	BR-SEAL
			RHR RELIEF VALVES FAIL TO CLOSE	SYS-RVC
			OPERATOR FAILS TO THROTTLE SI FLOW	SYS-OSR
53	1.32E-09	.00	ATWS WITHOUT MAIN FEEDWATER OCCURS	IEV-AWS
			RODS FAIL TO INSERT	AFM-RODS
			AFW SYSTEM FAILS	SYS-AFG
	1.13E-09	.00	MEDIUM LOCA INITIATING EVENT OCCURS	IEV-MLO
			HIGH PRESSURE INJECTION FAILS	SYS-HIO
			ACCUMULATOR INJECTION FAILS	SYS-ACC

TABLE 4.4.2-2 SEQUENCES DUE TO HUMAN ERROR RATES FOR RECOVERY ACTIONS - ALL SEQUENCES

NUMBER	SEQUENCE PROBABILITY	PERCENT CONTRIB	SEQUENCE DESCRIPTION				SEQUENCE IDENTIFIER
55	1.07E-09	.00	ATWS WITHOUT	MAIN FEEDWATER	OCCURS		IEV-AWS
			RODS FAIL TO	INSERT			AFM-RODS
			AMSAC FAILS				SYS-AMS
56	8.73E-10	.00	LOSS OF SERVICE	WATER INITIATING	EVENT OCCURS		IEV-SWS
			LOSS OF SERVICE	WATER IE FAULT	TREE		SYS-SWIE
			SAFEGUARDS POWER	AVAILABLE			DEL-OSP
			CHARGING SYSTEM	FAILS			SYS-CHG
57	8.13E-10	.00	TRANSIENT	WITHOUT MAIN	FEEDWATER	EVENT OCCURS	IEV-TRS
			SAFEGUARDS POWER	AVAILABLE			DEL-OSP
			CHARGING SYSTEM	FAILS			SYS-CHG
			COMPONENT	COOLING WATER	SYSTEM FAILS		SYS-CCW
58	3.62E-10	.00	ATWS WITHOUT	MAIN FEEDWATER	OCCURS		IEV-AWS
			REACTOR TRIP	BREAKERS	FAIL		AFM-BREAKER
			OPERATOR ACTION	TO DEENERGIZE	CRDM POWER	SUPPLY FAILS	SYS-ORT
			LONGTERM	SHUTDOWN	FAILS		SYS-LTS
59	0.00E+00	.00	LOSS OF DC BUS	INITIATING EVENT	OCCURS		IEV-TDC
			LOSS OF DC BUS	IE FAULT TREE			SYS-DBIE
			AFW SYSTEM	FAILS			SYS-AF4
			MFW SYSTEM	FAILS			SYS-OM4
			HIGH PRESSURE	RECIRCULATION	FAILS		SYS-HR2
60	0.00E+00	.00	ATWS WITHOUT	MAIN FEEDWATER	OCCURS		IEV-AWS
			REACTOR TRIP	SIGNAL FAILS			AFM-SIGNAL
			MANUAL	REACTOR TRIP	FAILS		SYS-MRT
			OPERATOR ACTION	TO DEENERGIZE	CRDM POWER	SUPPLY FAILS	SYS-ORT
			LONGTERM	SHUTDOWN	FAILS		SYS-LTS
61	0.00E+00	.00	ATWS WITHOUT	MAIN FEEDWATER	OCCURS		IEV-AWS
			REACTOR TRIP	SIGNAL FAILS			AFM-SIGNAL
			MANUAL	REACTOR TRIP	FAILS		SYS-MRT
			PRIMARY	PRESSURE RELIEF	FAILS		SYS-PPR
62	0.00E+00	.00	ATWS WITHOUT	MAIN FEEDWATER	OCCURS		IEV-AWS
			REACTOR TRIP	SIGNAL FAILS			AFM-SIGNAL
			MANUAL	REACTOR TRIP	FAILS		SYS-MRT
			AFW SYSTEM	FAILS			SYS-AFG
63	0.00E+00	.00	ATWS WITHOUT	MAIN FEEDWATER	OCCURS		IEV-AWS
			REACTOR TRIP	SIGNAL FAILS			AFM-SIGNAL
			MANUAL	REACTOR TRIP	FAILS		SYS-MRT
			OPERATOR ACTION	TO DEENERGIZE	CRDM POWER	SUPPLY FAILS	SYS-ORT
			AMSAC FAILS				SYS-AMS

TABLE 4.4.2-2 SEQUENCES DUE TO HUMAN ERROR RATES FOR RECOVERY ACTIONS - ALL SEQUENCES

NUMBER	SEQUENCE PROBABILITY	PERCENT CONTRIB	SEQUENCE DESCRIPTION	SEQUENCE IDENTIFIER
64	0.00E+00	.00	ATWS WITHOUT MAIN FEEDWATER OCCURS	IEV-AWS
			REACTOR TRIP BREAKERS FAIL	AFM-BREAKER
			OPERATOR ACTION TO DEENERGIZE CRDM POWER SUPPLY FAILS	SYS-ORT
			AMSAC FAILS	SYS-AMS
65	0.00E+00	.00	STEAMLINE BREAK INITIATING EVENT OCCURS	IEV-SLB
			AFW SYSTEM FAILS	SYS-AF1
			MFW SYSTEM FAILS	SYS-OM1
			HIGH PRESSURE RECIRCULATION FAILS	SYS-HR1
66	0.00E+00	.00	INTERFACING SYSTEMS LOCA INITIATING EVENT OCCURS	IEV-ISL
			RHR PUMP SEAL FAILS	BR-SEAL
			AFW SYSTEM FAILS	SYS-AFO
			MFW SYSTEM FAILS	SYS-OMO
67	0.00E+00	.00	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT OCCURS	IEV-SGR
			OPERATOR FAILS TO STOP DEPRESSURIZATION BY CLOSING PORV	SYS-OSD
			HIGH PRESSURE RECIRCULATION FAILS	SYS-HR1
68	0.00E+00	.00	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT OCCURS	IEV-SGR
			HIGH PRESSURE INJECTION FAILS	SYS-HI1
			OPERATOR FAILS TO STOP DEPRESSURIZATION BY CLOSING PORV	SYS-OSD
69	0.00E+00	.00	SMALL LOCA INITIATING EVENT OCCURS	IEV-SLO
			AFW SYSTEM FAILS	SYS-AFO
			MFW SYSTEM FAILS	SYS-OMO
			HIGH PRESSURE RECIRCULATION FAILS	SYS-HR1
			LOW PRESSURE RECIRCULATION FAILS	SYS-LR2
70	0.00E+00	.00	SMALL LOCA INITIATING EVENT OCCURS	IEV-SLO
			HIGH PRESSURE INJECTION FAILS	SYS-HI2
			AFW SYSTEM FAILS	SYS-AFO
			MFW SYSTEM FAILS	SYS-OMO
71	0.00E+00	.00	MEDIUM LOCA INITIATING EVENT OCCURS	IEV-MLO
			HIGH PRESSURE INJECTION FAILS	SYS-HIO
			AFW SYSTEM FAILS	SYS-AFO
			MFW SYSTEM FAILS	SYS-OMO

TABLE 3.4.2-3

RECOVERY ACTION TIMING AND COMPLEXITY

Identifier	Time Available	Actions Required
CHB	30 minutes	Start pump from control room.
OCD	8 hours	Locally open 1 of 2 steam generator PORVs. Instructions are clearly posted near the valves.
OIP	1 hour	Locally close manual valve. Radiation concern exists.
OM3	30 minutes	Locally open 1 of 2 feedwater bypass control valves and the corresponding steam generator PORV. Bypass valve and PORV are in the same and both have instructions posted near them.
31-LO-SW1300-HE	30 minutes	Locally open 1 of 2 manual valves on component cooling water heat exchangers.
40--BUS52----HE	30 minutes	Locally start diesel generator. Open breakers on three motor control centers to strip loads from bus 52.

TABLE 3.4.4-1 TOTAL CORE MELT RESULTS

TOTAL CORE MELT FREQUENCY = 6.628E-05
NUMBER INITIATORS = 16
NUMBER OF CUTSETS = 3693

INITIATING EVENTS	IMPORTANCE	NUMBER OF CUTSETS	INITIATOR CM FREQUENCY	INITIATING EVENT FREQUENCY
STATION BLACKOUT	39.82	516	2.6400E-05	4.3500E-04
SMALL LOCA	20.56	554	1.3600E-05	5.1200E-03
MEDIUM LOCA	12.26	351	8.1300E-06	2.3600E-03
SG TUBE RUPTURE	7.98	418	5.2900E-06	6.4100E-03
LOSS OF OFFSITE POWER	6.77	658	4.4900E-06	4.3600E-02
TRANSIENTS WITH MFW	4.13	291	2.7400E-06	3.0000E+00
LOSS OF INSTRUMENT AIR	3.14	36	2.0800E-06	1.0700E-04
LARGE LOCA	2.93	266	1.9400E-06	5.0000E-04
TRANSIENTS WITHOUT MFW	.69	343	4.5900E-07	1.4000E-01
LOSS OF SERVICE WATER	.64	58	4.2200E-07	1.2200E-04
VESSEL FAILURE	.45	1	3.0000E-07	3.0000E-07
LOSS OF ONE VITAL DC BUS	.32	61	2.1100E-07	2.3500E-03
STEAMLINE BREAK	.15	56	1.0000E-07	2.5000E-03
ATWS WITHOUT MFW	.10	33	6.8500E-08	3.8400E-06
LOSS OF COMPONENT COOLING	.04	42	2.8300E-08	1.6200E-03
INTERFACING SYSTEM LOCA	.02	9	1.4000E-08	1.4800E-06

TABLE 3.4.4-2

RECOVERY ACTIONS

Description	Identifiers
Start charging pump powered by TSC diesel	CHB
Cool down the RCS during a station blackout	OCD
Isolate RHR pumps	OIP
Locally establish main feedwater	OM3
Locally open SW-1300A or SW-1300B	31-LO-SW1300-HE
Align TSC diesel to bus 52	40--BUS52----HE

TABLE 3.4.4-3 INTERNAL FLOODING TOTAL CORE MELT RESULTS

TOTAL CORE MELT FREQUENCY = 2.417E-07
NUMBER OF INITIATORS = 6
NUMBER OF CUTSETS = 851

INITIATING EVENTS	IMPORTANCE	NUMBER OF CUTSETS	INITIATOR CM FREQUENCY	INITIATING EVENT FREQUENCY
DIESEL GENERATOR ROOM A	75.40	189	1.8226E-07	5.0000E-04
DIESEL GENERATOR ROOM B	24.01	202	5.8042E-08	5.0000E-04
RELAY ROOM	.17	64	4.0195E-10	1.5000E-04
TURBINE BUILDING BASEMENT	.17	66	4.0098E-10	1.1000E-04
TURBINE BUILDING BASEMENT	.13	58	3.1769E-10	8.9000E-05
CRD EQUIPEMENT ROOM	.13	272	3.0618E-10	1.5000E-05

TABLE 3.4.4-4 IMPORTANCE BY CORE MELT SEQUENCE - ALL SEQUENCES

NUMBER	SEQUENCE PROBABILITY	PERCENT CONTRIB	SEQUENCE DESCRIPTION	SEQUENCE IDENTIFIER
1	1.30E-05	19.57	STATION BLACKOUT INITIATING EVENT OCCURS	IEV-SBO
			TURBINE DRIVEN AFW PUMP FAILS	SYS-AF2
			POWER NOT RESTORED IN 2 HOURS	AC2-FAIL
2	1.21E-05	18.21	SMALL LOCA INITIATING EVENT OCCURS	IEV-SLO
			COOLDOWN AND DEPRESSURIZATION FOR CHARGING FLOW FAILS	SYS-ES1
			HIGH PRESSURE RECIRCULATION FAILS	SYS-HR1
			LOW PRESSURE RECIRCULATION FAILS	SYS-LR2
3	8.70E-06	13.13	STATION BLACKOUT INITIATING EVENT OCCURS	IEV-SBO
			POWER NOT RESTORED IN 11 HOURS	ACX-11
4	7.42E-06	11.19	MEDIUM LOCA INITIATING EVENT OCCURS	IEV-MLO
			HIGH PRESSURE RECIRCULATION FAILS	SYS-HRO
			LOW PRESSURE RECIRCULATION FAILS	SYS-LR2
5	4.31E-06	6.51	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT OCCURS	IEV-SGR
			COOLDOWN AND DEPRESSURIZATION BEFORE SG OVERFILL FAILS	SYS-OS1
			STEAM GENERATOR INTEGRITY FAILS	SSV-FAIL
			COOLDOWN AND DEPRESSURIZATION USING ECA-3.1,2 FAILS	SYS-EC4
[illegible]	4.01E-06	6.06	STATION BLACKOUT INITIATING EVENT OCCURS	IEV-SBO
			OFF-SITE POWER RESTORED	REC-OSP
			CHARGING FOR SEAL INJECTION FAILS	SYS-CHB
			CORE UNCOVERED BY RXCP SEAL LOCA	CCV-11
7	2.69E-06	4.06	TRANSIENT WITH MAIN FEEDWATER INITIATING EVENT OCCURS	IEV-TRA
			SAFEGUARDS POWER AVAILABLE	DEL-OSP
			AFW SYSTEM FAILS	SYS-AF3
			MFW SYSTEM FAILS	SYS-OM2
			BLEED AND FEED FAILS	SYS-OB2
8	2.17E-06	3.27	LOSS OF OFFSITE POWER INITIATING EVENT OCCURS	IEV-LSP
			SAFEGUARDS POWER AVAILABLE	DEL-OSP
			AFW SYSTEM FAILS	SYS-AF3
			BLEED AND FEED FAILS	SYS-OB5
9	2.08E-06	3.14	LOSS OF INSTRUMENT AIR INITIATING EVENT OCCURS	IEV-INA
			LOSS OF INSTRUMENT AIR IE FAULT TREE	SYS-IAIE
			SAFEGUARDS POWER AVAILABLE	DEL-OSP
			AFW SYSTEM FAILS	SYS-AF5
			MFW SYSTEM FAILS	SYS-OM3
10	1.39E-06	2.09	LARGE LOCA INITIATING EVENT OCCURS	IEV-LLO
			LOW PRESSURE RECIRCULATION FAILS	SYS-LR1

TABLE 3.4.4-4 IMPORTANCE BY CORE MELT SEQUENCE - ALL SEQUENCES

	SEQUENCE PROBABILITY	PERCENT CONTRIB	SEQUENCE DESCRIPTION	SEQUENCE IDENTIFIER
11	1.23E-06	1.86	LOSS OF OFFSITE POWER INITIATING EVENT OCCURS	IEV-LSP
			SAFEGUARDS POWER AVAILABLE	DEL-OSP
			CHARGING SYSTEM FAILS	SYS-CHG
			COMPONENT COOLING WATER SYSTEM FAILS	SYS-CCW
12	1.17E-06	1.77	SMALL LOCA INITIATING EVENT OCCURS	IEV-SLO
			HIGH PRESSURE INJECTION FAILS	SYS-HI2
			LOW PRESSURE INJECTION FAILS	SYS-LI2
13	1.09E-06	1.65	LOSS OF OFFSITE POWER INITIATING EVENT OCCURS	IEV-LSP
			SAFEGUARDS POWER AVAILABLE	DEL-OSP
			AFW SYSTEM FAILS	SYS-AF3
			HIGH PRESSURE RECIRCULATION FAILS	SYS-HR1
14	6.02E-07	.91	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT OCCURS	IEV-SGR
			AFW SYSTEM FAILS	SYS-AF1
			MFW SYSTEM FAILS	SYS-OM1
15	5.36E-07	.81	MEDIUM LOCA INITIATING EVENT OCCURS	IEV-MLO
			HIGH PRESSURE INJECTION FAILS	SYS-HI0
			LOW PRESSURE INJECTION FAILS	SYS-LI2
[illegible]	4.57E-07	.69	LARGE LOCA INITIATING EVENT OCCURS	IEV-LLO
			LOW PRESSURE INJECTION FAILS	SYS-LI1
17	4.21E-07	.64	LOSS OF SERVICE WATER INITIATING EVENT OCCURS	IEV-SWS
			LOSS OF SERVICE WATER IE FAULT TREE	SYS-SWIE
			SAFECUARDS POWER AVAILABLE	DEL-OSP
			AFW SYSTEM FAILS	SYS-AF6
18	3.62E-07	.55	SMALL LOCA INITIATING EVENT OCCURS	IEV-SLO
			HIGH PRESSURE INJECTION FAILS	SYS-HI2
			LOW PRESSURE RECIRCULATION FAILS	SYS-LR1
19	3.00E-07	.45	VESSEL FAILURE INITIATING EVENT OCCURS	IEV-VEF
20	2.76E-07	.42	STATION BLACKOUT INITIATING EVENT OCCURS	IEV-SBO
			OFF-SITE POWER RESTORED	REC-OSP
			CHARGING FOR SEAL INJECTION FAILS	SYS-CHB
			HIGH PRESSURE RECIRCULATION FAILS	SYS-HR1
21	2.57E-07	.39	TRANSIENT WITHOUT MAIN FEEDWATER EVENT OCCURS	IEV-TRS
			SAFEGUARDS POWER AVAILABLE	DEL-OSP
			AFW SYSTEM FAILS	SYS-AF3
			BLEED AND FEED FAILS	SYS-OB2
22	2.11E-07	.32	LOSS OF DC BUS INITIATING EVENT OCGURS	IEV-TDC
			LOSS OF DC BUS IE FAULT TREE	SYS-DBIE
			AFW SYSTEM FAILS	SYS-AF4
			MFW SYSTEM FAILS	SYS-OM4
			BLEED AND FEED FAILS	SYS-OB3

TABLE 3.4.4-4 IMPORTANCE BY CORE MELT SEQUENCE - ALL SEQUENCES

NUMBER	SEQUENCE PROBABILITY	PERCENT CONTRIB	SEQUENCE DESCRIPTION	SEQUENCE IDENTIFIER
23	2.01E-07	.30	TRANSIENT WITHOUT MAIN FEEDWATER EVENT OCCURS	IEV-TRS
			SAFEGUARDS POWER AVAILABLE	DEL-OSP
			AFW SYSTEM FAILS	SYS-AF3
			HIGH PRESSURE RECIRCULATION FAILS	SYS-HR1
24	1.93E-07	.29	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT OCCURS	IEV-SGR
			HIGH PRESSURE INJECTION FAILS	SYS-HI1
			COOLDOWN AND DEPRESSURIZATION BEFORE SG OVERFILL FAILS	SYS-OS1
			STEAM GENERATOR INTEGRITY FAILS	SSV-FAIL
25	1.79E-07	.27	STATION BLACKOUT INITIATING EVENT OCCURS	IEV-SBO
			OFF-SITE POWER RESTORED	REC-OSP
			CHARGING FOR SEAL INJECTION FAILS	SYS-CHB
			TURBINE DRIVEN AFW PUMP FAILS	SYS-AF2
			CORE UNCOVERED BY RXCP SEAL LOCA	CCV-2
26	1.67E-07	.25	STATION BLACKOUT INITIATING EVENT OCCURS	IEV-SBO
			OFF-SITE POWER RESTORED	REC-OSP
			CHARGING FOR SEAL INJECTION FAILS	SYS-CHB
			RCS INVENTORY RESTORATION FAILS	SYS-ORI
[illegible]	1.59E-07	.24	MEDIUM LOCA INITIATING EVENT OCCURS	IEV-MLO
			HIGH PRESSURE INJECTION FAILS	SYS-HIO
			LOW PRESSURE RECIRCULATION FAILS	SYS-LR1
28	1.19E-07	.18	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT OCCURS	IEV-SGR
			STEAM GENERATOR ISOLATION FAILS	SYS-ISO
			COOLDOWN AND DEPRESSURIZATION USING ECA-3.1,2 FAILS	SYS-EC3
29	1.00E-07	.15	LARGE LOCA INITIATING EVENT OCCURS	IEV-LLO
			ACCUMULATOR INJECTION FAILS	SYS-ACC
30	6.49E-08	.10	STATION BLACKOUT INITIATING EVENT OCCURS	IEV-SBO
			RCS COOLDOWN FAILS	SYS-OCD
			POWER NOT RESTORED IN 9 HOURS	ACX-9
31	5.96E-08	.09	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT OCCURS	IEV-SGR
			COOLDOWN AND DEPRESSURIZATION BEFORE SG OVERFILL FAILS	SYS-OS1
			STEAM GENERATOR INTEGRITY MAINTAINED	SSV-SUC
			COOLDOWN AND DEPRESSURIZATION AFTER SG OVERFILL FAILS	SYS-OS2
32	4.85E-08	.07	STEAMLINE BREAK INITIATING EVENT OCCURS	IEV-SLB
			REACTOR POWER BELOW 10%	PWR-LOW
			MAIN STEAM ISOLATION FAILS	SYS-IS1
			HIGH PRESSURE INJECTION FAILS	SYS-HI3

TABLE 3.4.4-4 IMPORTANCE BY CORE MELT SEQUENCE - ALL SEQUENCES

NUMBER	SEQUENCE PROBABILITY	PERCENT CONTRIB	SEQUENCE DESCRIPTION	SEQUENCE IDENTIFIER
33	4.40E-08	.07	ATWS WITHOUT MAIN FEEDWATER OCCURS	IEV-AWS
			REACTOR TRIP BREAKERS FAIL	AFM-BREAKER
			PRIMARY PRESSURE RELIEF FAILS	SYS-PPR
34	3.45E-08	.05	TRANSIENT WITH MAIN FEEDWATER INITIATING EVENT OCCURS	IEV-TRA
			SAFEGUARDS POWER AVAILABLE	DEL-OSP
			CHARGING SYSTEM FAILS	SYS-CHG
			COMPONENT COOLING WATER SYSTEM FAILS	SYS-CCW
35	2.73E-08	.04	STEAMLINE BREAK INITIATING EVENT OCCURS	IEV-SLB
			AFW SYSTEM FAILS	SYS-AF1
			MFW SYSTEM FAILS	SYS-OM1
			BLEED AND FEED FAILS	SYS-OB4
36	2.45E-08	.04	STEAMLINE BREAK INITIATING EVENT OCCURS	IEV-SLB
			HIGH PRESSURE INJECTION FAILS	SYS-HI3
			AFW SYSTEM FAILS	SYS-AF1
			MFW SYSTEM FAILS	SYS-OM1
37	1.67E-08	.03	SMALL LOCA INITIATING EVENT OCCURS	IEV-SLO
			HIGH PRESSURE INJECTION FAILS	SYS-HI2
			COOLDOWN AND DEPRESSURIZATION FOR ACC AND LI2 FAILS	SYS-OP2
38	1.56E-08	.02	LOSS OF COMPONENT COOLING WATER EVENT OCCURS	IEV-CCS
			LOSS OF COMPONENT COOLING WATER IE FAULT TREE	SYS-CCWIE
			AT LEAST ONE TRAIN OF SERVICE WATER AVAILABLE	DEL-SW
			AFW SYSTEM FAILS	SYS-AF3
			MFW SYSTEM FAILS	SYS-OM2
39	1.51E-08	.02	STATION BLACKOUT INITIATING EVENT OCCURS	IEV-SBO
			OFF-SITE POWER RESTORED	REC-OSP
			CHARGING FOR SEAL INJECTION FAILS	SYS-CHB
			RCS COOLDOWN FAILS	SYS-OCD
			CORE UNCOVERED BY RXCP SEAL LOCA	CCV-9
40	1.39E-08	.02	TRANSIENT WITH MAIN FEEDWATER INITIATING EVENT OCCURS	IEV-TRA
			SAFECUARDS POWER AVAILABLE	DEL-OSP
			AFW SYSTEM FAILS	SYS-AF3
			MFW SYSTEM FAILS	SYS-OM2
			HIGH PRESSURE RECIRCULATION FAILS	SYS-HR1
41	1.33E-08	.02	MEDIUM LOCA INITIATING EVENT OCCURS	IEV-MLO
			HIGH PRESSURE INJECTION FAILS	SYS-HIO
			COOLDOWN AND DEPRESSURIZATION FAILS	SYS-OP1
42	1.27E-08	.02	LOSS OF COMPONENT COOLING WATER EVENT OCCURS	IEV-CCS
			LOSS OF COMPONENT COOLING WATER IE FAULT TREE	SYS-CCWIE
			AT LEAST ONE TRAIN OF SERVICE WATER AVAILABLE	DEL-SW
			CHARGING SYSTEM FAILS	SYS-CHG

TABLE 3.4.4-4 IMPORTANCE BY CORE MELT SEQUENCE - ALL SEQUENCES

NUMBER	SEQUENCE PROBABILITY	PERCENT CONTRIB	SEQUENCE DESCRIPTION			SEQUENCE IDENTIFIER
43	1.24E-08	.02	ATWS WITHOUT	MAIN FEEDWATER	OCCURS	IEV-AWS
			REACTOR TRIP	BREAKERS	FAIL	AFM-BREAKER
			AFW SYSTEM	FAILS		SYS-AFG
44	7.40E-09	.01	INTERFACING	SYSTEMS LOCA	INITIATING EVENT OCCURS	IEV-ISL
			RHR PIPE	FAILS		BR-PIPE
45	7.10E-09	.01	ATWS WITHOUT	MAIN FEEDWATER	OCCURS	IEV-AWS
			RODS FAIL TO	INSERT		AFM-RODS
			PRIMARY	PRESSURE RELIEF	FAILS	SYS-PPR
46	3.73E-09	.01	INTERFACING	SYSTEMS LOCA	INITIATING EVENT OCCURS	IEV-ISL
			RHR PUMP SEAL	FAILS		BR-SEAL
			HIGH PRESSURE	INJECTION	FAILS	SYS-HI4
47	3.31E-09	.00	SMALL LOCA	INITIATING EVENT	OCCURS	IEV-SLO
			AFW SYSTEM	FAILS		SYS-AFO
			MFW SYSTEM	FAILS		SYS-OMO
			BLEED AND FEED	FAILS		SYS-OB1
48	2.44E-09	.00	SMALL LOCA	INITIATING EVENT	OCCURS	IEV-SLO
			HIGH PRESSURE	INJECTION	FAILS	SYS-HI2
			ACCUMULATOR	INJECTION	FAILS	SYS-ACC
49	2.22E-09	.00	ATWS WITHOUT	MAIN FEEDWATER	OCCURS	IEV-AWS
			RODS FAIL TO	INSERT		AFM-RODS
			LONGTERM	SHUTDOWN	FAILS	SYS-LTS
50	1.56E-09	.00	INTERFACING	SYSTEMS LOCA	INITIATING EVENT OCCURS	IEV-ISL
			RHR PUMP SEAL	FAILS		BR-SEAL
			RHR RELIEF	VALVES	FAIL TO CLOSE	SYS-RVC
			OPERATOR FAILS	TO THROTTLE	SI FLOW	SYS-OSR
51	1.32E-09	.00	ATWS WITHOUT	MAIN FEEDWATER	OCCURS	IEV-AWS
			RODS FAIL TO	INSERT		AFM-RODS
			AFW SYSTEM	FAILS		SYS-AFG
52	1.13E-09	.00	MEDIUM LOCA	INITIATING EVENT	OCCURS	IEV-MLO
			HIGH PRESSURE	INJECTION	FAILS	SYS-HIO
			ACCUMULATOR	INJECTION	FAILS	SYS-ACC
53	1.07E-09	.00	ATWS WITHOUT	MAIN FEEDWATER	OCCURS	IEV-AWS
			RODS FAIL TO	INSERT		AFM-RODS
			AMSAC FAILS			SYS-AMS
54	8.73E-10	.00	LOSS OF SERVICE	WATER INITIATING EVENT OCCURS		IEV-SWS
			LOSS OF SERVICE	WATER IE FAULT TREE		SYS-SWIE
			SAFEGUARDS POWER	AVAILABLE		DEL-OSP
			CHARGING SYSTEM	FAILS		SYS-CHG

TABLE 3.4.4-4 IMPORTANCE BY CORE MELT SEQUENCE - ALL SEQUENCES

NUMBER	SEQUENCE PROBABILITY	PERCENT CONTRIB	SEQUENCE DESCRIPTION	SEQUENCE IDENTIFIER
55	8.13E-10	.00	TRANSIENT WITHOUT MAIN FEEDWATER EVENT OCCURS	IEV-TRS
			SAFEGUARDS POWER AVAILABLE	DEL-OSP
			CHARGING SYSTEM FAILS	SYS-CHG
			COMPONENT COOLING WATER SYSTEM FAILS	SYS-CCW
56	6.83E-10	.00	INTERFACING SYSTEMS LOCA INITIATING EVENT OCCURS	IEV-ISL
			RHR PUMP SEAL FAILS	BR-SEAL
			OPERATOR FAILS TO ISOLATE RHR PUMPS	SYS-OIP
			OPERATOR FAILS TO THROTTLE SI FLOW	SYS-OSR
57	6.58E-10	.00	INTERFACING SYSTEMS LOCA INITIATING EVENT OCGURS	IEV-ISL
			RHR PUMP SEAL FAILS	BR-SEAL
			OPERATOR FAILS TO ISOLATE RHR PUMPS	SYS-OIP
			RHR RELIEF VALVES FAIL TO CLOSE	SYS-RVC
58	3.62E-10	.00	ATWS WITHOUT MAIN FEEDWATER OCCURS	IEV-AWS
			REACTOR TRIP BREAKERS FAIL	AFM-BREAKER
			OPERATOR ACTION TO DEENERGIZE CRDM POWER SUPPLY FAILS	SYS-ORT
			LONGTERM SHUTDOWN FAILS	SYS-LTS
59	0.00E+00	.00	LOSS OF DC BUS INITIATING EVENT OCCURS	IEV-TDC
			LOSS OF DC BUS IE FAULT TREE	SYS-DBIE
			AFW SYSTEM FAILS	SYS-AF4
			MFW SYSTEM FAILS	SYS-OM4
			HIGH PRESSURE RECIRCULATION FAILS	SYS-HR2
60	0.00E+00	.00	ATWS WITHOUT MAIN FEEDWATER OCCURS	IEV-AWS
			REACTOR TRIP SIGNAL FAILS	AFM-SIGNAL
			MANUAL REACTOR TRIP FAILS	SYS-MRT
			OPERATOR ACTION TO DEENERGIZE CRDM POWER SUPPLY FAILS	SYS-ORT
			LONGTERM SHUTDOWN FAILS	SYS-LTS
61	0.00E+00	.00	ATWS WITHOUT MAIN FEEDWATER OCCURS	IEV-AWS
			REACTOR TRIP SIGNAL FAILS	AFM-SIGNAL
			MANUAL REACTOR TRIP FAILS	SYS-MRT
			PRIMARY PRESSURE RELIEF FAILS	SYS-PPR
62	0.00E+00	.00	ATWS WITHOUT MAIN FEEDWATER OCCURS	IEV-AWS
			REACTOR TRIP SIGNAL FAILS	AFM-SIGNAL
			MANUAL REACTOR TRIP FAILS	SYS-MRT
			AFW SYSTEM FAILS	SYS-AFG
63	0.00E+00	.00	ATWS WITHOUT MAIN FEEDWATER OCCURS	IEV-AWS
			REACTOR TRIP SIGNAL FAILS	AFM-SIGNAL
			MANUAL REACTOR TRIP FAILS	SYS-MRT
			OPERATOR ACTION TO DEENERGIZE CRDM POWER SUPPLY FAILS	SYS-ORT
			AMSAC FAILS	SYS-AMS

TABLE 3.4.4-4 IMPORTANCE BY CORE MELT SEQUENCE - ALL SEQUENCES

NUMBER	SEQUENCE PROBABILITY	PERCENT CONTRIB	SEQUENCE DESCRIPTION	SEQUENCE IDENTIFIER
64	0.00E+00	.00	ATWS WITHOUT MAIN FEEDWATER OCCURS	IEV-AWS
			REACTOR TRIP BREAKERS FAIL	AFM-BREAKER
			OPERATOR ACTION TO DEENERGIZE CRDM POWER SUPPLY FAILS	SYS-ORT
			AMSAC FAILS	SYS-AMS
65	0.00E+00	.00	STEAMLINE BREAK INITIATING EVENT OCCURS	IEV-SLB
			AFW SYSTEM FAILS	SYS-AF1
			MFW SYSTEM FAILS	SYS-OM1
			HIGH PRESSURE RECIRCULATION FAILS	SYS-HR1
66	0.00E+00	.00	INTERFACING SYSTEMS LOCA INITIATING EVENT OCCURS	IEV-ISL
			RHR PUMP SEAL FAILS	BR-SEAL
			AFW SYSTEM FAILS	SYS-AF0
			MFW SYSTEM FAILS	SYS-OM0
67	0.00E+00	.00	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT OCCURS	IEV-SGR
			OPERATOR FAILS TO STOP DEPRESSURIZATION BY CLOSING PORV	SYS-OSD
			HIGH PRESSURE RECIRCULATION FAILS	SYS-HR1
68	0.00E+00	.00	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT OCCURS	IEV-SGR
			HIGH PRESSURE INJECTION FAILS	SYS-HI1
			OPERATOR FAILS TO STOP DEPRESSURIZATION BY CLOSING PORV	SYS-OSD
69	0.00E+00	.00	SMALL LOCA INITIATING EVENT OCCURS	IEV-SLO
			AFW SYSTEM FAILS	SYS-AF0
			MFW SYSTEM FAILS	SYS-OM0
			HIGH PRESSURE RECIRCULATION FAILS	SYS-HR1
			LOW PRESSURE RECIRCULATION FAILS	SYS-LR2
70	0.00E+00	.00	SMALL LOCA INITIATING EVENT OCCURS	IEV-SLO
			HIGH PRESSURE INJECTION FAILS	SYS-HI2
			AFW SYSTEM FAILS	SYS-AF0
			MFW SYSTEM FAILS	SYS-OM0
71	0.00E+00	.00	MEDIUM LOCA INITIATING EVENT OCCURS	IEV-MLO
			HIGH PRESSURE INJECTION FAILS	SYS-HI0
			AFW SYSTEM FAILS	SYS-AF0
			MFW SYSTEM FAILS	SYS-OM0

TABLE 3.4.4-5 IMPORTANCE BY INTERNAL FLOODING SEQUENCE - ALL SEQUENCES

NUMBER	SEQUENCE PROBABILITY	PERCENT CONTRIB	SEQUENCE DESCRIPTION	SEQUENCE IDENTIFIER
1	1.76E-07	72.69	FLOOD IN A DIESEL GENERATOR ROOM OCCURS	IEV-FL3
			CHARGING SYSTEM FAILS	SYS-CHG
			COMPONENT COOLING WATER SYSTEM FAILS	SYS-CCW
2	4.14E-08	17.12	FLOOD IN B DIESEL GENERATOR ROOM OCCURS	IEV-FL4
			CHARGING SYSTEM FAILS	SYS-CHG
			COMPONEHT COOLING WATER SYSTEM FAILS	SYS-CCW
3	1.67E-08	6.89	FLOOD IN B DIESEL GENERATOR ROOM OCCURS	IEV-FL4
			AFW SYSTEM FAILS	SYS-AF3
			MFW SYSTEM FAILS	SYS-OM2
			BLEED AND FEED FAILS	SYS-OB2
4	6.54E-09	2.70	FLOOD IN A DIESEL GENERATOR ROOM OCCURS	IEV-FL3
			AFW SYSTEM FAILS	SYS-AF3
			MFW SYSTEM FAILS	SYS-OM2
			BLEED AND FEED FAILS	SYS-OB2
5	2.16E-10	.09	FLOOD IN RELAY ROOM OCCURS	IEV-FL5
			AFW SYSTEM FAILS	SYS-AF3
			MFW SYSTEM FAILS	SYS-OM2
			BLEED AND FEED FAILS	SYS-OB2
6	1.75E-10	.07	FLOOD IN TURBINE BUILDING BASEMENT OCCURS (2 CW PUMPS RNG)	IEV-FL2
			AFW SYSTEM FAILS	SYS-AF3
			BLEED AND FEED FAILS	SYS-OB2
7	1.72E-10	.07	FLOOD IN CRD EQUIPMENT ROOM OCCURS	IEV-FL6
			CHARGING SYSTEM FAILS	SYS-CHG
			COMPONENT COOLING WATER SYSTEM FAILS	SYS-CCW
8	1.62E-10	.07	FLOOD IN RELAY ROOM OCCURS	IEV-FL5
			CHARGING SYSTEM FAILS	SYS-CHG
			COMPONENT COOLING WATER SYSTEM FAILS	SYS-CCW
9	1.38E-10	.06	FLOOD IN TURBINE BUILDING BASEMENT OCCURS (1 CW PUMP RNG)	IEV-FL1
			AFW SYSTEM FAILS	SYS-AF3
			BLEED AND FEED FAILS	SYS-OB2
10	1.34E-10	.06	FLOOD IH CRD EQUIPMENT ROOM OCCURS	IEV-FL6
			AFW SYSTEM FAILS	SYS-AF3
			MFW SYSTEM FAILS	SYS-OM2
			BLEED AND FEED FAILS	SYS-OB2
11	1.16E-10	.05	FLOOD IN TURBINE BUILDING BASEMENT OCCURS (2 CW PUMPS RNG)	IEV-FL2
			CHARGING SYSTEM FAILS	SYS-CHG
			COMPONENT COOLING WATER SYSTEM FAILS	SYS-CCW

TABLE 3.4.4-5 IMPORTANCE BY INTERNAL FLOODING SEQUENCE - ALL SEQUENCES

NUMBER	SEQUENCE PROBABILITY	PERCENT CONTRIB	SEQUENCE DESCRIPTION	SEQUENCE IDENTIFIER
12	1.11E-10	.05	FLOOD IN TURBINE BUILDING BASEMENT OCCURS (2 CW PUMPS RNG)	IEV-FL2
			AFW SYSTEM FAILS	SYS-AF3
			HIGH PRESSURE RECIRCULATION FAILS	SYS-HR1
13	9.18E-11	.04	FLOOD IN TURBINE BUILDING BASEMENT OCCURS (1 CW PUMP RNG)	IEV-FL1
			CHARGING SYSTEM FAILS	SYS-CHG
			COMPONENT COOLING WATER SYSTEM FAILS	SYS-CCW
14	8.80E-11	.04	FLOOD IN TURBINE BUILDING BASEMENT OCCURS (1 CW PUMP RNG)	IEV-FL1
			AFW SYSTEM FAILS	SYS-AF3
			HIGH PRESSURE RECIRCULATION FAILS	SYS-HR1
15	2.39E-11	.01	FLOOD IN RELAY ROOM OCCURS	IEV-FL5
			AFW SYSTEM FAILS	SYS-AF3
			MFW SYSTEM FAILS	SYS-OM2
			HIGH PRESSURE RECIRCULATION FAILS	SYS-HR1
16	4.57E-13	.00	FLOOD IN CRD EQUIPMENT ROOM OCCURS	IEV-FL6
			AFW SYSTEM FAILS	SYS-AF3
			MFW SYSTEM FAILS	SYS-OM2
			HIGH PRESSURE RECIRCULATION FAILS	SYS-HR1
[illegible]	0.00E+00	.00	FLOOD IN B DIESEL GENERATOR ROOM OCCURS	IEV-FL4
			AFW SYSTEM FAILS	SYS-AF3
			MFW SYSTEM FAILS	SYS-OM2
			HIGH PRESSURE RECIRCULATION FAILS	SYS-HR1
18	0.00E+00	.00	FLOOD IN A DIESEL GENERATOR ROOM OCCURS	IEV-FL3
			AFW SYSTEM FAILS	SYS-AF3
			MFW SYSTEM FAILS	SYS-OM2
			HIGH PRESSURE RECIRCULATION FAILS	SYS-HR1

TABLE 3.4.4-6 IMPORTANCE BY CORE MELT CUTSET - TOP 50 CUTSETS

RANK	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
1	8.70E-06	13.13	STATION BLACKOUT INITIATING EVENT OCCURS	4.35E-04	IEV-SBO
			POWER NOT RESTORED IN 11 HOURS	2.00E-02	ACX-11
2	6.78E-06	10.23	STATION BLACKOUT INITIATING EVENT OCCURS	4.35E-04	IEV-SBO
			POWER NOT RESTORED IN 2 HOURS	2.65E-01	AC2-FAIL
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	5.88E-02	05BPT--AFW1CXPS
3	5.01E-06	7.56	SMALL LOCA INITIATING EVENT OCCURS	5.12E-03	IEV-SLO
			COMMON CAUSE FAILURE OF RHR	9.79E-04	34---RHR-----CM
4	4.24E-06	6.40	STATION BLACKOUT INITIATING EVENT OCCURS	4.35E-04	IEV-SBO
			POWER NOT RESTORED IN 2 HOURS	2.65E-01	AC2-FAIL
			OPERATOR FAILS TO ISOLATE AOV MU-3A	3.68E-02	05BAV-MU3A---HE
5	3.09E-06	4.66	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT OCCURS	6.41E-03	IEV-SGR
			STEAM GENERATOR INTEGRITY FAILS	9.84E-01	SSV-FAIL
			OPERATOR FAILS TO COOLDOWN AND DEPRESSURIZE RCS PER EOP'S	9.80E-03	36--OS1------HE
			OPERATOR FAILS TO COOLDOWN AND DEPRESSURIZE RCS PER ECA-3.1/3.2	5.00E-02	36--EC4------HE
6	2.36E-06	3.56	STATION BLACKOUT INITIATING EVENT OCCURS	4.35E-04	IEV-SBO
			OFF-SITE POWER RESTORED	1.00E+00	REC-OSP
			CORE UNCOVERED BY RXCP SEAL LOCA	7.07E-02	CCV-11
			TSC DIESEL GENERATOR FAILS TO START AND RUN	7.66E-02	10-GE-TSC-DG-PS
7	2.31E-06	3.49	MEDIUM LOCA INITIATING EVENT OCCURS	2.36E-03	IEV-MLO
			COMMON CAUSE FAILURE OF RHR	9.79E-04	34---RHR-----CM
8	2.17E-06	3.27	SMALL LOCA INITIATING EVENT OCCURS	5.12E-03	IEV-SLO
			OPERATOR FAILS TO STOP BOTH RHR PUMPS	4.23E-04	34I---LI2A---HE
9	1.89E-06	2.85	LOSS OF INSTRUMENT AIR INITIATING EVENT OCCURS	1.07E-04	IEV-INA
			COMPRESSOR 1F MECHANICAL FAILURE	8.26E-01/Yr	01-PM-SIAC1FYPR
			NO SERVICE WATER DUE TO COMMON CAUSE FAILURES	6.22E-05	02----SW-----CM
			OPERATOR FAILS TO ISOLATE AOV MU-3A	3.68E-02	05BAV-MU3A---HE
10	1.84E-06	2.78	SMALL LOCA INITIATING EVENT OCCURS	5.12E-03	IEV-SLO
			PIPE BREAK GREATER THAN 0.7 INCH DIAMETER	5.00E-01	36-PP-PBD----RP
			CONTAINMENT SUMP STRAINERS PLUGGED	7.20E-04	34RFL---SUMP-PL
11	1.70E-06	2.56	MEDIUM LOCA INITIATING EVENT OCCURS	2.36E-03	IEV-MLO
			CONTAINMENT SUMP STRAINERS PLUGGED	7.20E-04	34RFL---SUMP-PL
12	1.48E-06	2.23	TRANSIENT WITH MAIN FEEDWATER INITIATING EVENT OCCURS	3.00E+00	IEV-TRA
			COMMON CAUSE FAILURES OF AFW SYSTEM	2.12E-04	05B-SY1FAULT-CM
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-03	36-RXCP-STOP-HE
13	9.98E-07	1.51	MEDIUM LOCA INITIATING EVENT OCCURS	2.36E-03	IEV-MLO
			OPERATOR FAILS TO STOP BOTH RHR PUMPS	4.23E-04	34I---LI2A---HE

TABLE 3.4.4-6 IMPORTANCE BY CORE MELT CUTSET - TOP 50 CUTSETS

	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
14	8.65E-07	1.31	STATION BLACKOUT INITIATING EVENT OCCURS	4.35E-04	IEV-SBO
			POWER NOT RESTORED IN 2 HOURS	2.65E-01	AC2-FAIL
			MOV MS-102 FAILS TO OPEN	7.50E-03	06-MV--MS102XCC
15	7.73E-07	1.17	SMALL LOCA INITIATING EVENT OCCURS	5.12E-03	IEV-SLO
			COMMON MODE FAILURE OF TRAIN A AND TRAIN B	1.51E-04	55--SY---SIAB-CM
16	5.49E-07	.83	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT OCCURS	6.41E-03	IEV-SGR
			STEAM GENERATOR INTEGRITY FAILS	9.84E-01	SSV-FAIL
			FAILURE DUE TO COMMON CAUSE FAILURES	8.71E-05	35-OS1OS2EC4-CM
17	4.89E-07	.74	LARGE LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-LLO
			COMMON CAUSE FAILURE OF RHR	9.79E-04	34---RHR-----CM
18	4.64E-07	.70	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT OCCURS	6.41E-03	IEV-SGR
			STEAM GENERATOR INTEGRITY FAILS	9.84E-01	SSV-FAIL
			OPERATOR FAILS TO COOLDOWN AND DEPRESSURIZE RCS PER EOP'S	9.80E-03	36--OS1------HE
			MOV RHR-11 FAILS TO OPEN OR TRANSFERS CLOSED	7.50E-03	34RMV-RHR11--CC
19	4.17E-07	.63	TRANSIENT WITH MAIN FEEDWATER INITIATING EVENT OCCURS	3.00E+00	IEV-TRA
			AFW PUMP 1B MECHANICAL FAILURE	1.63E-02	05BPM--AFW1B-PS
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			NO SERVICE WATER DUE TO COMMON CAUSE FAILURES	6.22E-05	02----SW-----CM
20	3.94E-07	.59	LOSS OF OFFSITE POWER INITIATING EVENT OCCURS	4.36E-02	IEV-LSP
			AFW PUMP 1B MECHANICAL FAILURE	1.63E-02	05BPM--AFW1B-PS
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			FEEDER BREAKERS ON 4160V BUS 5 FAIL TO OPEN	4.80E-02	39-CB-BUS5FB-FO
			COMPRESSOR 1B MECHANICAL FAILURE	8.44E-02	01-PM-SIAC1B-PS
21	3.73E-07	.56	STATION BLACKOUT INITIATING EVENT OCCURS	4.35E-04	IEV-SBO
			POWER NOT RESTORED IN 2 HOURS	2.65E-01	AC2-FAIL
			AFW PUMP 1C AUX LUBE OIL PUMP MECHANICAL FAILURE	3.24E-03	05BPM-ALOP1CXPS
22	3.60E-07	.54	LARGE LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-LLO
			CONTAINMENT SUMP STRAINERS PLUGGED	7.20E-04	34RFL---SUMP-PL
23	3.56E-07	.54	MEDIUM LOCA INITIATING EVENT OCCURS	2.36E-03	IEV-MLO
			COMMON MODE FAILURE OF TRAIN A AND TRAIN B	1.51E-04	55--SY---SIAB-CM
24	3.46E-07	.52	STATION BLACKOUT INITIATING EVENT OCCURS	4.35E-04	IEV-SBO
			POWER NOT RESTORED IN 2 HOURS	2.65E-01	AC2-FAIL
			AFW PUMP 1C UNAVAILABLE DUE TO TEST OR MAINTENANCE	3.00E-03	05BPM--AFW1C-TM
25	3.45E-07	.52	LOSS OF OFFSITE POWER INITIATING EVENT OCCURS	4.36E-02	IEV-LSP
			RELIEF VALVE CVC-101B FAILS TO CLOSE	1.60E-02	35-AMCVC101B-FC
			FEEDER BREAKERS ON 4160V BUS 5 FAIL TO OPEN	4.80E-02	39-CB-BUS5FB-FO
			CCW PUMP B MECHANICAL FAILURE	1.03E-02	31-PM--CCW1B-PS

TABLE 3.4.4-6 IMPORTANCE BY CORE MELT CUTSET - TOP 50 CUTSETS

	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
26	3.18E-07	.48	SMALL LOCA INITIATING EVENT OCCURS	5.12E-03	IEV-SLO
			NO SERVICE WATER DUE TO COMMON CAUSE FAILURES	6.22E-05	02----SW-----CM
27	3.08E-07	.46	STATION BLACKOUT INITIATING EVENT OCCURS	4.35E-04	IEV-SBO
			OFF-SITE POWER RESTORED	1.00E+00	REC-OSP
			CORE UNCOVERED BY RXCP SEAL LOCA	7.07E-02	CCV-11
			OPERATOR FAILS TO ESTABLISH CHARGING FLOW	1.00E-02	35--CHB------HE
28	3.08E-07	.46	STATION BLACKOUT INITIATING EVENT OCCURS	4.35E-04	IEV-SBO
			OFF-SITE POWER RESTORED	1.00E+00	REC-OSP
			CORE UNCOVERED BY RXCP SEAL LOCA	7.07E-02	CCV-11
			OPERATOR FAILS TO RESTORE POWER TO BUS 52	1.00E-02	40--BUS52----HE
29	3.00E-07	.45	VESSEL FAILURE INITIATING EVENT OCCURS	3.00E-07	IEV-VEF
30	2.88E-07	.43	SMALL LOCA INITIATING EVENT OCCURS	5.12E-03	IEV-SLO
			MOV CC-400A FAILS TO OPEN OR TRANSFERS CLOSED	7.50E-03	34RMV-CC400A-CC
			MOV CC-400B FAILS TO OPEN OR TRANSFERS CLOSED	7.50E-03	34RMV-CC400B-CC
31	2.70E-07	.41	LOSS OF SERVICE WATER INITIATING EVENT OCCURS	1.22E-04	IEV-SWS
			SERVICE WATER SYSTEM COMMON CAUSE	1.21E-04	02-SY-SWIE---CM
			LOSS OF COOLING TO AFW PUMP 1A FAN COIL UNIT	1.00E+00	SWA-SUB
			AFW PUMP 1B MECHANICAL FAILURE	1.63E-02	05BPM--AFW1B-PS
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
32	2.54E-07	.38	TRANSIENT WITH MAIN FEEDWATER INITIATING EVENT OCCURS	3.00E+00	IEV-TRA
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			AFW PUMP 1B MECHANICAL FAILURE	1.63E-02	05BPM--AFW1B-PS
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-03	36-RXCP-STOP-HE
33	2.31E-07	.35	STATION BLACKOUT INITIATING EVENT OCCURS	4.35E-04	IEV-SBO
			OFF-SITE POWER RESTORED	1.00E+00	REC-OSP
			CORE UNCOVERED BY RXCP SEAL LOCA	7.07E-02	CCV-11
			MOV CVC-301 FAILS TO OPEN	7.50E-03	35-MV-CVC301-FO
34	2.31E-07	.35	STATION BLACKOUT INITIATING EVENT OCCURS	4.35E-04	IEV-SBO
			POWER NOT RESTORED IN 2 HOURS	2.65E-01	AC2-FAIL
			SOV AFW-111C FAILS TO OPEN	2.00E-03	05BSVAFW111CXCC
35	2.11E-07	.32	LARGE LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-LLO
			COMMON CAUSE FAILURE OF BOTH LPI TRAINS	4.21E-04	34I--LPI-----CM
36	1.75E-07	.26	LOSS OF OFFSITE POWER INITIATING EVENT OCCURS	4.36E-02	IEV-LSP
			AFW PUMP 1B MECHANICAL FAILURE	1.63E-02	05BPM--AFW1B-PS
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			DIESEL GENERATOR 1A FAILURE TO START AND RUN	2.13E-02	10-GE-DG1A---PS
			COMPRESSOR 1B MECHANICAL FAILURE	8.44E-02	01-PM-SIAC1B-PS

TABLE 3.4.4-6 IMPORTANCE BY CORE MELT CUTSET - TOP 50 CUTSETS

	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
37	1.53E-07	.23	LOSS OF OFFSITE POWER IHITIATING EVENT OCCURS	4.36E-02	IEV-LSP
			RELIEF VALVE CVC-101B FAILS TO CLOSE	1.60E-02	35-AMCVC101B-FC
			DIESEL GENERATOR 1A FAILURE TO START AND RUN	2.13E-02	10-GE-DG1A---PS
			CCW PUMP B MECHANICAL FAILURE	1.03E-02	31-PM--CCW1B-PS
38	1.47E-07	.22	MEDIUM LOCA INITIATING EVENT OCCURS	2.36E-03	IEV-MLO
			NO SERVICE WATER DUE TO COMMON CAUSE FAILURES	6.22E-05	02----SW-----CM
39	1.46E-07	.22	STATION BLACKOUT INITIATING EVENT OCCURS	4.35E-04	IEV-SBO
			OFF-SITE POWER RESTORED	1.00E+00	REC-OSP
			CORE UNCOVERED BY RXCP SEAL LOCA	7.07E-02	CCV-11
			MOV CVC-1 FAILS TO CLOSE	4.76E-03	35-MV-CVC1---FC
40	1.44E-07	.22	SMALL LOCA INITIATING EVENT OCCURS	5.12E-03	IEV-SLO
			PIPE BREAK GREATER THAN 0.7 INCH DIAMETER	5.00E-01	36-PP-PBD----RP
			MOV SI-350B FAILS TO OPEN OR TRANSFERS CLOSED	7.50E-03	34RMV-SI350B-CC
			MOV SI-350A FAILS TO OPEN OR TRAHSFERS CLOSED	7.50E-03	34RMV-SI350A-CC
41	1.44E-07	.22	SMALL LOCA INITIATING EVENT OCCURS	5.12E-03	IEV-SLO
			PIPE BREAK GREATER THAN 0.7 INCH DIAMETER	5.00E-01	36-PP-PBD----RP
			MOV SI-350B FAILS TO OPEN OR TRANSFERS CLOSED	7.50E-03	34RMV-SI350B-CC
			MOV SI-351A FAILS TO OPEN OR TRANSFERS CLOSED	7.50E-03	34RMV-SI351A-CC
	1.44E-07	.22	SMALL LOCA INITIATING EVENT OCCURS	5.12E-03	IEV-SLO
			PIPE BREAK GREATER THAN 0.7 INCH DIAMETER	5.00E-01	36-PP-PBD----RP
			MOV SI-350B FAILS TO OPEN OR TRANSFERS CLOSED	7.50E-03	34RMV-SI350B-CC
			MOV CC-400A FAILS TO OPEN OR TRANSFERS CLOSED	7.50E-03	34RMV-CC400A-CC
43	1.44E-07	.22	SMALL LOCA INITIATING EVENT OCCURS	5.12E-03	IEV-SLO
			PIPE BREAK GREATER THAN 0.7 INCH DIAMETER	5.00E-01	36-PP-PBD----RP
			MOV SI-351B FAILS TO OPEN OR TRANSFERS CLOSED	7.50E-03	34RMV-SI351B-CC
			MOV SI-350A FAILS TO OPEN OR TRANSFERS CLOSED	7.50E-03	34RMV-SI350A-CC
44	1.44E-07	.22	SMALL LOCA INITIATING EVENT OCCURS	5.12E-03	IEV-SLO
			PIPE BREAK GREATER THAN 0.7 INCH DIAMETER	5.00E-01	36-PP-PBD----RP
			MOV SI-351B FAILS TO OPEN OR TRANSFERS CLOSED	7.50E-03	34RMV-SI351B-CC
			MOV SI-351A FAILS TO OPEN OR TRANSFERS CLOSED	7.50E-03	34RMV-SI351A-CC
45	1.44E-07	.22	SMALL LOCA IHITIATING EVENT OCCURS	5.12E-03	IEV-SLO
			PIPE BREAK GREATER THAN 0.7 INCH DIAMETER	5.00E-01	36-PP-PBD----RP
			MOV SI-351B FAILS TO OPEN OR TRANSFERS CLOSED	7.50E-03	34RMV-SI351B-CC
			MOV CC-400A FAILS TO OPEN OR TRANSFERS CLOSED	7.50E-03	34RMV-CC400A-CC
46	1.44E-07	.22	SMALL LOCA INITIATING EVENT OCCURS	5.12E-03	IEV-SLO
			PIPE BREAK GREATER THAN 0.7 INCH DIAMETER	5.00E-01	36-PP-PBD----RP
			MOV CC-400B FAILS TO OPEH OR TRANSFERS CLOSED	7.50E-03	34RMV-CC400B-CC
			MOV SI-350A FAILS TO OPEN OR TRANSFERS CLOSED	7.50E-03	34RMV-SI350A-CC
47	1.44E-07	.22	SMALL LOCA INITIATING EVEHT OCCURS	5.12E-03	IEV-SLO
			PIPE BREAK GREATER THAN 0.7 INCH DIAMETER	5.00E-01	36-PP-PBD----RP
			MOV CC-400B FAILS TO OPEN OR TRANSFERS CLOSED	7.50E-03	34RMV-CC400B-CC
			MOV SI-351A FAILS TO OPEN OR TRANSFERS CLOSED	7.50E-03	34RMV-SI351A-CC

TABLE 3.4.4-6 IMPORTAHCE BY CORE MELT CUTSET - TOP 50 CUTSETS

	CUTSET PROB	PERCENT	BASIC EVENT NAME				EVENT PROB.	IDENTIFIER
48	1.44E-07	.22	SMALL LOCA	INITIATING EVENT OCCURS			5.12E-03	IEV-SLO
			PIPE BREAK	GREATER THAN	0.7 INCH	DIAMETER	5.00E-01	36-PP-PBD----RP
			MOV SI-208	FAILS TO CLOSE			7.50E-03	33RMV-SI208--FC
			MOV SI-209	FAILS TO CLOSE			7.50E-03	33RMV-SI209--FC
49	1.33E-07	.20	MEDIUM LOCA	INITIATING EVENT OCCURS			2.36E-03	IEV-MLO
			MOV SI-350B	FAILS TO OPEN	OR TRANSFERS	CLOSED	7.50E-03	34RMV-SI350B-CC
			MOV SI-350A	FAILS TO OPEN	OR TRANSFERS	CLOSED	7.50E-03	34RMV-SI350A-CC
50	1.33E-07	.20	MEDIUM LOCA	INITIATING EVENT OCCURS			2.36E-03	IEV-MLO
			MOV SI-350B	FAILS TO OPEN	OR TRAHSFERS	CLOSED	7.50E-03	34RMV-SI350B-CC
			MOV SI-351A	FAILS TO OPEN	OR TRAHSFERS	CLOSED	7.50E-03	34RMV-SI351A-CC

TABLE 3.3.4-7 IMPORTANCE BY INTERNAL FLOODING CUTSET - TOP 50 CUTSETS

	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
1	8.24E-08	34.09	FLOOD IN A DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL3
			RELIEF VALVE CVC-101B FAILS TO CLOSE	1.60E-02	35-AMCVC101B-FC
			BUS5-FAILS	1.00E+00	BUS5-FAILS
			CCW PUMP B MECHANICAL FAILURE	1.03E-02	31-PM--CCW1B-PS
2	1.90E-08	7.86	FLOOD IN A DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL3
			RELIEF VALVE CVC-101B FAILS TO CLOSE	1.60E-02	35-AMCVC101B-FC
			BUS5-FAILS	1.00E+00	BUS5-FAILS
			CCW PUMP 1B UNAVAILABLE DUE TO TEST OR MAINTENANCE	2.37E-03	31-PM--CCW1B-TM
3	1.20E-08	4.96	FLOOD IN A DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL3
			BUS5-FAILS	1.00E+00	BUS5-FAILS
			BREAKER 16201 TRANSFERS OPEH	2.40E-05	40-CB--16201-CO
4	1.20E-08	4.96	FLOOD IN A DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL3
			BUS5-FAILS	1.00E+00	BUS5-FAILS
			TRANSFORMER SUPPLY BREAKER 1-607 TRANSFERS OPEN	2.40E-05	39-CB--1-607-CO
5	1.20E-08	4.96	FLOOD IN A DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL3
			BREAKER 16204 TRANSFERS OPEH	2.40E-05	40-CB--16204-CO
			BUS5-FAILS	1.00E+00	BUS5-FAILS
	1.20E-08	4.96	FLOOD IN B DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL4
			BREAKER 15201 TRANSFERS OPEN	2.40E-05	40-CB--15201-CO
			BUS6-FAILS	1.00E+00	BUS6-FAILS
7	1.20E-08	4.96	FLOOD IN B DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL4
			TRANSFORMER SUPPLY BREAKER 1-505 TRANSFERS OPEN	2.40E-05	39-CB--1-505-CO
			BUS6-FAILS	1.00E+00	BUS6-FAILS
8	1.15E-08	4.76	FLOOD IN A DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL3
			CHARGING PUMP 1B MECHANICAL FAILURE	2.24E-03	35-PM-CHGP1B-PS
			BUS5-FAILS	1.00E+00	BUS5-FAILS
			CCW PUMP B MECHANICAL FAILURE	1.03E-02	31-PM--CCW1B-PS
9	9.70E-09	4.01	FLOOD IN A DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL3
			BUS5-FAILS	1.00E+00	BUS5-FAILS
			TRANSFORMER 1-62 FAILURE	1.94E-05	40-TR--1-62--SG
10	9.70E-09	4.01	FLOOD IH B DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL4
			TRANSFORMER 1-52 FAILURE	1.94E-05	40-TR--1-52--SG
			BUS6-FAILS	1.00E+00	BUS6-FAILS
11	4.26E-09	1.76	FLOOD IN B DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL4
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			NO SERVICE WATER DUE TO COMMON CAUSE FAILURES	6.22E-05	02----SW-----CM
			BUS6-FAILS	1.00E+00	BUS6-FAILS

TABLE 3.3.4-7 IMPORTANCE BY INTERNAL FLOODING CUTSET - TOP 50 CUTSETS

RANK	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
12	2.86E-09	1.18	FLOOD IN B DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL4
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24
			FEEDER BREAKERS ON 4160V BUS 5 FAIL TO OPEN	4.80E-02	39-CB-BUS5FB-FO
			BUS6-FAILS	1.00E+00	BUS6-FAILS
13	2.65E-09	1.10	FLOOD IN A DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL3
			CHARGING PUMP 1B MECHANICAL FAILURE	2.24E-03	35-PM-CHGP1B-PS
			BUS5-FAILS	1.00E+00	BUS5-FAILS
			CCW PUMP 1B UNAVAILABLE DUE TO TEST OR MAINTENANCE	2.37E-03	31-PM--CCW1B-TM
14	2.60E-09	1.08	FLOOD IN A DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL3
			AFW PUMP 1B MECHANICAL FAILURE	1.63E-02	05BPM--AFW1B-PS
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			BUS5-FAILS	1.00E+00	BUS5-FAILS
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-03	36-RXCP-STOP-HE
15	2.60E-09	1.08	FLOOD IN B DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL4
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			BUS6-FAILS	1.00E+00	BUS6-FAILS
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-03	36-RXCP-STOP-HE
[illegible]	2.50E-09	1.03	FLOOD IH A DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL3
			BUS5-FAILS	1.00E+00	BUS5-FAILS
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24
			FEEDER BREAKERS ON 4160V BUS 6 FAIL TO OPEN	4.20E-02	39-CB-BUS6FB-FO
17	1.64E-09	.68	FLOOD IN A DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL3
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			BUS5-FAILS	1.00E+00	BUS5-FAILS
			BREAKER FROM BUS BRB-102 TO BUS BRB-104 TRANSFERS OPEN	2.40E-05	38-CBB102-04-CO
18	1.64E-09	.68	FLOOD IN B DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL4
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			BREAKER 15208 TRANSFERS OPEH	2.40E-05	40-CB--15208-CO
			BUS6-FAILS	1.00E+00	BUS6-FAILS
19	1.38E-09	.57	FLOOD IN A DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL3
			RELIEF VALVE CVC-101B FAILS TO CLOSE	1.60E-02	35-AMCVC101B-FC
			BUS5-FAILS	1.00E+00	BUS5-FAILS
			CCW HEAT EXCH 1B UNAVAILABLE DUE TO TEST OR MAINTENANCE	1.73E-04	31-HE--CCW1B-TM
20	1.27E-09	.53	FLOOD IN A DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL3
			BUS5-FAILS	1.00E+00	BUS5-FAILS
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24
			DIESEL GENERATOR 1B FAILURE TO START AND RUN	2.13E-02	10-GE-DG1B---PS
21	1.27E-09	.53	FLOOD IN B DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL4
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24
			DIESEL GEHERATOR 1A FAILURE TO START AND RUN	2.13E-02	10-GE-DG1A---PS
			BUS6-FAILS	1.00E+00	BUS6-FAILS

TABLE 3.3.4-7 IMPORTANCE BY INTERNAL FLOODING CUTSET - TOP 50 CUTSETS

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
22	1.20E-09	.50	FLOOD IN A DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL3
			BUS5-FAILS	1.00E+00	BUS5-FAILS
			BUS 6 FAILURE	2.40E-06	39-BS-BUS6---SG
23	1.20E-09	.50	FLOOD IN A DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL3
			BUS5-FAILS	1.00E+00	BUS5-FAILS
			BUS 62 FAILURE	2.40E-06	40-BS-BUS62--SG
24	1.20E-09	.50	FLOOD IN A DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL3
			MCC-62E FAILURE	2.40E-06	40-BS-MCC62E-SG
			BUS5-FAILS	1.00E+00	BUS5-FAILS
25	1.20E-09	.50	FLOOD IN B DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL4
			BUS 5 FAILURE	2.40E-06	39-BS-BUS5---SG
			BUS6-FAILS	1.00E+00	BUS6-FAILS
26	1.20E-09	.50	FLOOD IN B DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL4
			BUS 52 FAILURE	2.40E-06	40-BS-BUS52--SG
			BUS6-FAILS	1.00E+00	BUS6-FAILS
27	1.03E-09	.43	FLOOD IN A DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL3
			RELIEF VALVE CVC-101B FAILS TO CLOSE	1.60E-02	35-AMCVC101B-FC
			BUS5-FAILS	1.00E+00	BUS5-FAILS
			MOV SW-1300B UNAVAILABLE DUE TO TEST OR MAINTENANCE	1.29E-04	02-MVSW1300B-TM
28	9.48E-10	.39	FLOOD IN B DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL4
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			TRAVELING WATER SCREEN 1A1 MECHANICAL FAILURE	3.72E-03	02-FLT-TW1A1-PS
			TRAVELING WATER SCREEN 1A2 MECHANICAL FAILURE	3.72E-03	02-FLT-TW1A2-PS
			BUS6-FAILS	1.00E+00	BUS6-FAILS
29	8.00E-10	.33	FLOOD IN A DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL3
			RELIEF VALVE CVC-101B FAILS TO CLOSE	1.60E-02	35-AMCVC101B-FC
			BUS5-FAILS	1.00E+00	BUS5-FAILS
			CHECK VALVE CC-3B FAILS TO OPEN	1.00E-04	31-CV---CC3B-FO
30	7.64E-10	.32	FLOOD IN B DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL4
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			BREAKER FROM MCC-52D FAILS TO CLOSE	3.00E-03	40-CB-52D/A6-FC
			TRAVELING WATER SCREEN 1A2 MECHANICAL FAILURE	3.72E-03	02-FLT-TW1A2-PS
			BUS6-FAILS	1.00E+00	BUS6-FAILS
31	7.64E-10	.32	FLOOD IN B DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL4
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			TRAVELING WATER SCREEN 1A1 MECHANICAL FAILURE	3.72E-03	02-FLT-TW1A1-PS
			BREAKER FROM MCC-35C FAILS TO CLOSE	3.00E-03	40-CB-35C/B8-FC
			BUS6-FAILS	1.00E+00	BUS6-FAILS

TABLE 3.3.4-7 IMPORTANCE BY INTERNAL FLOODING CUTSET - TOP 50 CUTSETS

[illegible]K	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
32	6.90E-10	.29	FLOOD IN B DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL4
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			SW PUMP 1A1 MECHANICAL FAILURE	1.40E-02	02-PM-SW1A1--PS
			ROTATING STRAINER 1A2 MECHANICAL FAILURE	7.20E-04	02-FLR-RS1A2-PR
			BUS6-FAILS	1.00E+00	BUS6-FAILS
33	6.17E-10	.26	FLOOD IN B DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL4
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			BREAKER FROM MCC-52D FAILS TO CLOSE	3.00E-03	40-CB-52D/A6-FC
			BREAKER FROM MCC-35C FAILS TO CLOSE	3.00E-03	40-CB-35C/B8-FC
			BUS6-FAILS	1.00E+00	BUS6-FAILS
34	4.98E-10	.21	FLOOD IN A DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL3
			RELIEF VALVE CVC-101B FAILS TO CLOSE	1.60E-02	35-AMCVC101B-FC
			BUS5-FAILS	1.00E+00	BUS5-FAILS
			NO SERVICE WATER DUE TO COMMON CAUSE FAILURES	6.22E-05	02----SW-----CM
35	3.19E-10	.13	FLOOD IN A DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL3
			SOV AFW-111B FAILS TO OPEN	2.00E-03	05BSVAFW111B-CC
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			BUS5-FAILS	1.00E+00	BUS5-FAILS
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-03	36-RXCP-STOP-HE
[illegible]	3.19E-10	.13	FLOOD IN B DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL4
			SOV AFW-111A FAILS TO OPEN	2.00E-03	05BSVAFW111A-CC
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			BUS6-FAILS	1.00E+00	BUS6-FAILS
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-03	36-RXCP-STOP-HE
37	3.19E-10	.13	FLOOD IN B DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL4
			SW CONTROL VALVE SOV 33734 FAILS TO OPEN	2.00E-03	17-SVSV33734-CC
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			BUS6-FAILS	1.00E+00	BUS6-FAILS
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-03	36-RXCP-STOP-HE
38	3.17E-10	.13	FLOOD IN A DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL3
			BUS5-FAILS	1.00E+00	BUS5-FAILS
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24
			AOV SW-301B FAILS TO OPEN	5.32E-03	02-AV-SW301B-CC
39	3.17E-10	.13	FLOOD IN B DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL4
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24
			AOV SW-301A FAILS TO OPEN	5.32E-03	02-AV-SW301A-CC
			BUS6-FAILS	1.00E+00	BUS6-FAILS
40	2.87E-10	.12	FLOOD IN A DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL3
			AFW PUMP 1B UNAVAILABLE DUE TO TEST OR MAINTENANCE	1.80E-03	05BPM--AFW1B-TM
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			BUS5-FAILS	1.00E+00	BUS5-FAILS
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-03	36-RXCP-STOP-HE

TABLE 3.3.4-7 IMPORTANCE BY INTERNAL FLOODING CUTSET - TOP 50 CUTSETS

	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
41	2.76E-10	.11	FLOOD IN B DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL4
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			SW PUMP 1A1 MECHANICAL FAILURE	1.40E-02	02-PM-SW1A1--PS
			SW PUMP 1A2 MECHANICAL FAILURE	2.88E-04	02-PM-SW1A2--PR
			BUS6-FAILS	1.00E+00	BUS6-FAILS
42	2.47E-10	.10	FLOOD IN A DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL3
			COMMON CAUSE FAILURES OF AFW SYSTEM	2.12E-04	05B-SY1FAULT-CM
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-03	36-RXCP-STOP-HE
43	2.47E-10	.10	FLOOD IN B DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL4
			COMMON CAUSE FAILURES OF AFW SYSTEM	2.12E-04	05B-SY1FAULT-CM
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-03	36-RXCP-STOP-HE
44	2.33E-10	.10	FLOOD IN B DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL4
			MOV MS-102 FAILS TO OPEN	7.50E-03	06-MV--MS102-CC
			NO SERVICE WATER DUE TO COMMON CAUSE FAILURES	6.22E-05	02----SW-----CM
			BUS6-FAILS	1.00E+00	BUS6-FAILS
45	2.21E-10	.09	FLOOD IN A DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL3
			BUS5-FAILS	1.00E+00	BUS5-FAILS
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24
			TRAVELING WATER SCREEN 1B2 MECHANICAL FAILURE	3.72E-03	02-FLT-TW1B2-PS
46	2.21E-10	.09	FLOOD IN B DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL4
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24
			TRAVELING WATER SCREEN 1A1 MECHANICAL FAILURE	3.72E-03	02-FLT-TW1A1-PS
			BUS6-FAILS	1.00E+00	BUS6-FAILS
47	2.19E-10	.09	FLOOD IN A DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL3
			RELIEF VALVE CVC-101B FAILS TO CLOSE	1.60E-02	35-AMCVC101B-FC
			BUS5-FAILS	1.00E+00	BUS5-FAILS
			CCW HEAT EXCHANGER 1B SHELL LEAK	2.74E-05	31-HE--CCW1B-HS
48	2.14E-10	.09	FLOOD IN B DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL4
			AFW PUMP 1A UNAVAILABLE DUE TO TEST OR MAINTENANCE	1.34E-03	05BPM--AFW1A-TM
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			BUS6-FAILS	1.00E+00	BUS6-FAILS
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-03	36-RXCP-STOP-HE
49	1.94E-10	.08	FLOOD IN A DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL3
			CHARGING PUMP 1B MECHANICAL FAILURE	2.24E-03	35-PM-CHGP1B-PS
			BUS5-FAILS	1.00E+00	BUS5-FAILS
			CCW HEAT EXCH 1B UNAVAILABLE DUE TO TEST OR MAINTENANCE	1.73E-04	31-HE--CCW1B-TM
50	1.92E-10	.08	FLOOD IN A DIESEL GENERATOR ROOM OCCURS	5.00E-04	IEV-FL3
			RELIEF VALVE CVC-101B FAILS TO CLOSE	1.60E-02	35-AMCVC101B-FC
			BUS5-FAILS	1.00E+00	BUS5-FAILS
			BREAKER 16208 TRANSFERS OPEN	2.40E-05	40-CB--16208-CO

4.0 BACK-END ANALYSIS

This section presents the Level 2 portion of the Kewaunee Nuclear Plant IPE.

4.1 Plant Data and Plant Description

4.1.1 Containment Description

The Kewaunee containment is described below, along with the containment systems which are important to containment integrity, as well as the 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 Kewaunee containment employs a 2-loop Westinghouse design with a free standing steel shell containment. The Modular Accident Analysis Program (MAAP) is used in the Kewaunee IPE to model plant and containment response to severe core melt accidents. Further discussion of MAAP and how MAAP is used in the Kewaunee IPE is contained in Section 4.1.2 of this report. For MAAP modeling and discussion purposes, the containment is sectioned into several compartments consisting of a total free volume of approximately 1,320,000 cubic feet. Figure 4.1-1 illustrates a vertical cross section of the Kewaunee containment. MAAP sections the containment into four individual compartments: the upper compartment, annular compartment, lower compartment, and cavity. The upper compartment is defined as the large containment volume located above the refueling floor (649′-6″ elevation). The annular compartment is defined as the area of containment which is below the refueling floor, but outside the secondary shield wall (i.e., missile barrier). The lower compartment is that portion of containment between the containment floor (592′-0″ elevation) and the refueling floor, but inside the secondary shield wall. The cavity includes the area in the reactor cavity and the instrument tunnel.

Kewaunee's containment system consists of two separate structures: a reactor containment vessel and a shield building. The reactor containment vessel is a low leakage steel shell, including penetrations, designed to confine the radioactive material that could be released during a core melt accident. Nominal dimensions of the Kewaunee containment are as follows[14]:

Inner Diameter (ft)	105
Interior Height (ft)	206
Cylinder Shell Thickness (in)	1.5
Dome Thickness (in)	0.75
Ellipsoidal basemat thickness (in)	1.5
Internal free volume (ft^3)	1,320,000

The containment vessel is supported on a grout base that was installed after the vessel construction was completed and tested. Both the containment vessel and the shield building are supported on a common foundation slab. Freedom of movement between the containment vessel and the shield building is virtually unlimited. With the exception of the support grout placed underneath and near the knuckle sides of the vessel, there are no structural ties between the containment vessel and the shield building above the foundation slab.

Completely enclosing the containment vessel is the 2½ foot thick concrete shield building. The shield building has the shape of a right circular cylinder with a shallow dome roof. A 5 foot annular gap is provided between the containment vessel and the shield building. Kewaunee's shield building is a medium-leakage concrete structure designed to provide the following features:

1. Protection of the containment vessel from adverse weather conditions and external missiles.

2. Biological shielding for design basis accident (DBA) conditions.

3. A means for collection and filtration of fission product leakage from the containment vessel.

The open design and significant venting areas for the sub-compartments within the Kewaunee containment help ensure a well-mixed atmosphere, a feature that inhibits combustible gas pocketing. Steel grating around the periphery of the operating deck provides a good flow path between the annular and upper compartments. This grating also provides an effective fission product removal mechanism in the form of impaction. The lower and upper compartments communicate through openings around the steam generators and their corresponding vaults.

Figure 4.1-2 illustrates the Kewaunee reactor cavity and instrument tunnel geometry. Free volume of the cavity and instrument tunnel is approximately 4700 ft^3 with a floor area of 290 ft^2. Geometry of the cavity and structures at the exit of the seal table are important features of the Kewaunee containment because they act to limit the extent of debris dispersed from the cavity following a high pressure melt ejection (HPME). The Kewaunee cavity has a total concrete basemat thickness of approximately 9.8 feet of limestone common sand. Therefore, for low pressure sequences where the debris remains in the cavity, 9.8 feet of concrete must be ablated before the corium will breach the containment boundary.

The Kewaunee containment does not facilitate flooding of the reactor cavity. Although water can readily flow from the upper compartment to the lower and annular compartment floors, water cannot access the cavity due to the instrument tunnel wall in the annular compartment. A potential flow path does exist in the form of two access hatches located on the instrument tunnel wall, approximately 2 feet off the floor of the annular compartment. These hatches are closed during normal operations, but if they were left open the cavity could be easily flooded if the RWST were injected into the containment.

Personnel access into containment is normally provided through the main personnel airlock located on the 649'-6" elevation. The equipment hatch is located in the annular compartment on the 606'-0" elevation. An emergency personnel airlock is also located in the annular compartment at the 606'-0" elevation. All three of these hatches employ non-metallic gaskets as part of their leakage barrier.

All containment penetrations are double-barrier assemblies consisting of a closed sleeve, in most cases, or a double gasketed closure for the fuel transfer tube. The mechanical penetrations are welded to the containment shell. Likewise, the electrical penetration assemblies (EPAs) are constructed to provide a leak-tight barrier. The EPAs employ a non-metallic seal and potting compound and are of D. G. O'Brien or Conax design. There are no electrical penetrations in the immediate vicinity of the seal table structure.

4.1.1.2 Containment Systems

The Kewaunee containment design includes the following three containment cooling systems:

1. Four Containment Fan Coil Units (FCUs), Consisting of Two Trains of Two FCUs Each
2. Two Internal Containment Spray (ICS) Trains
3. Two Low Pressure Residual Heat Removal Pumps and Heat Exchangers

The residual heat removal (RHR) system, although not a containment system, also provides a means of long term containment heat removal. Brief descriptions of each of the three systems listed above are provided below.

<u>Containment Fan Coil Units</u>

The containment air cooling system consists of four FCUs each capable of removing approximately 16 MW from a saturated steam-air mixture at a flow rate of 41,000 cfm. All four FCUs are located in the annular compartment. Two of them are located on the 606'-0" elevation, with the remaining two located on the 626'-0" elevation in the annular compartment. Air that has risen to the top of the containment dome is drawn down to the suction of the FCUs by two fans through two ducts that follow the contour of the containment dome upward on opposite sides of the containment. These ducts take suction at the highest point in the center of containment. The air is then forced through the FCUs where the steam is blown over a series of cooling coils which condense the steam and cool the air. The cool air is then discharged into the lower compartment where it is heated as it makes its way through the steam generator vaults to the upper compartment. The cooling water for the FCUs is supplied by the service water system. Therefore, loss of one train of service water results in the loss of two FCUs and loss of the entire service water system results in the loss of all four FCUs.

Internal Containment Spray System

The ICS system consists of two separate trains, each with a horizontal centrifugal pump, capable of delivering 1300 gpm of water designed to provide sufficient heat removal capability to maintain containment pressure below the design pressure. The ICS system provides both a potential pressure reduction mechanism and a means to remove fission products from the containment atmosphere.

The ICS system initially takes suction from the refueling water storage tank (RWST) and injects into the containment through spray headers located in the dome of the containment. At 37% RWST water level, one of the ICS pumps is turned off while the second ICS pump continues taking suction from the RWST and one of the RHR pumps is turned off and aligned to the containment sump. At 10% RWST water level, the operating ICS pump's suction is switched from the RWST to the discharge of the RHR heat exchanger. At the same time, the remaining RHR pump starts taking suction from the containment recirculation sump. Therefore, when the ICS system is in recirculation mode, the operating spray pump takes suction from the discharge of the RHR heat exchanger. Since the ICS system uses an RHR heat exchanger, it is capable of removing decay heat from containment.

Residual Heat Removal (RHR) System

The RHR system consists of two separate trains, each capable of removing decay heat. In the recirculation mode, the RHR pumps take suction from the containment recirculation sump. The containment water is cooled in the RHR heat exchangers and returned to the reactor vessel through the RHR injection nozzles. Containment cooling through use of the RHR pumps is achieved through continuous injection through the failed vessel onto the debris in the cavity. The RHR pumps and heat exchanger also provide suction and heat removal capabilities for the containment spray system in recirculation. Since the RHR pumps and heat exchanger rely on the component cooling water system (CCW) and the service water system for cooling water, loss of either one of these two systems will negate the containment heat removal capability of the RHR system, as well as the ICS system.

4.1.2 Containment Data

The Modular Accident Analysis Program (MAAP) is used in the Kewaunee IPE to provide an integrated approach to the modeling of plant and containment thermal hydraulic response and fission product behavior during severe core damage accidents. MAAP requires plant-specific input data which is compiled into a MAAP parameter file. The Kewaunee MAAP parameter file provides a complete, realistic description of the Kewaunee containment for a MAAP simulation. The parameter file data is identical for all accident sequences. The Kewaunee parameter file and its supporting documentation are included in the Kewaunee containment data collection notebook. Table 4.1-1 correlates some important plant data to the parameter file section in which they are tabulated.

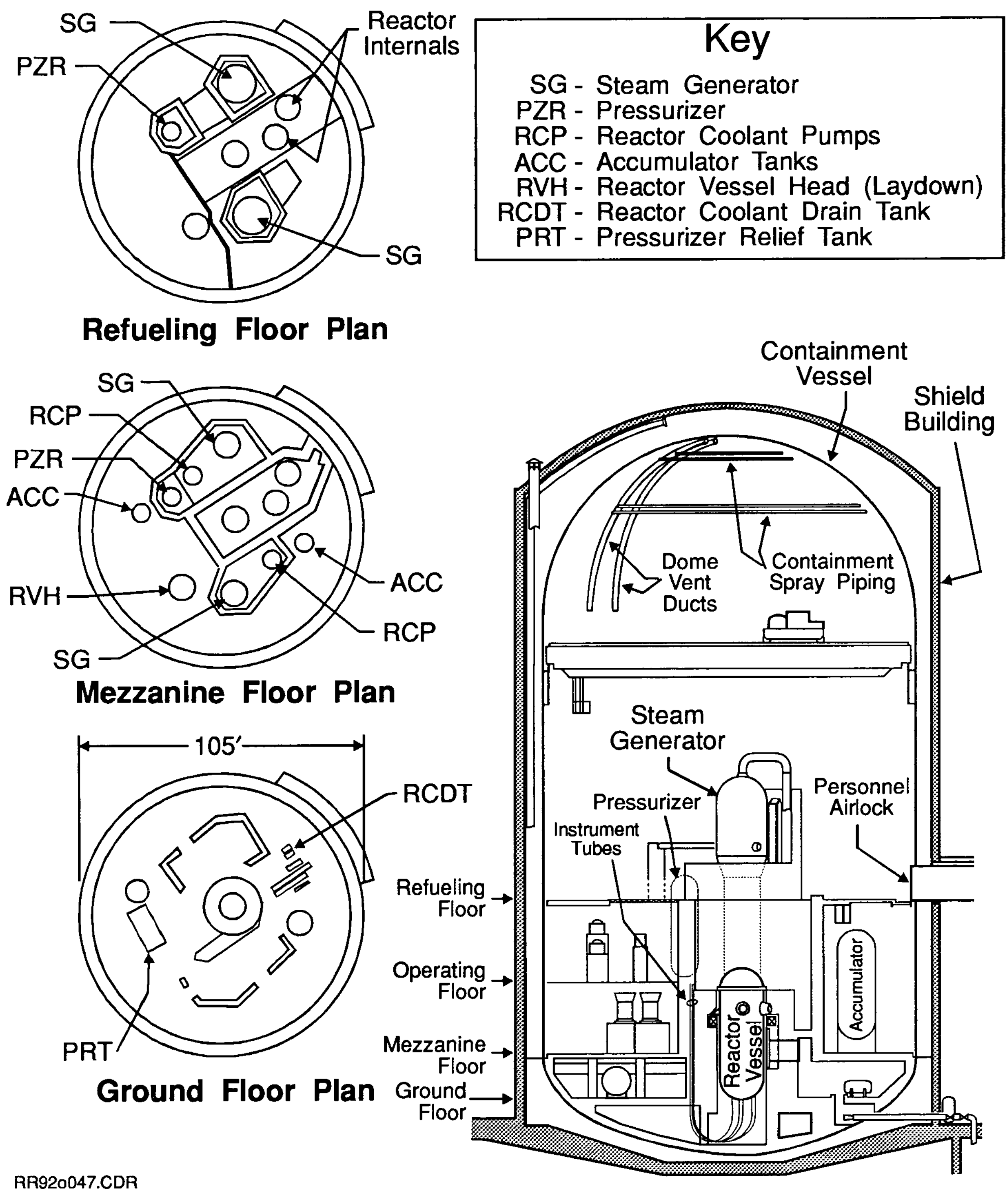


FIGURE 4.1-1

CROSS SECTION AND FLOOR PLAN OF KEWAUNEE CONTAINMENT

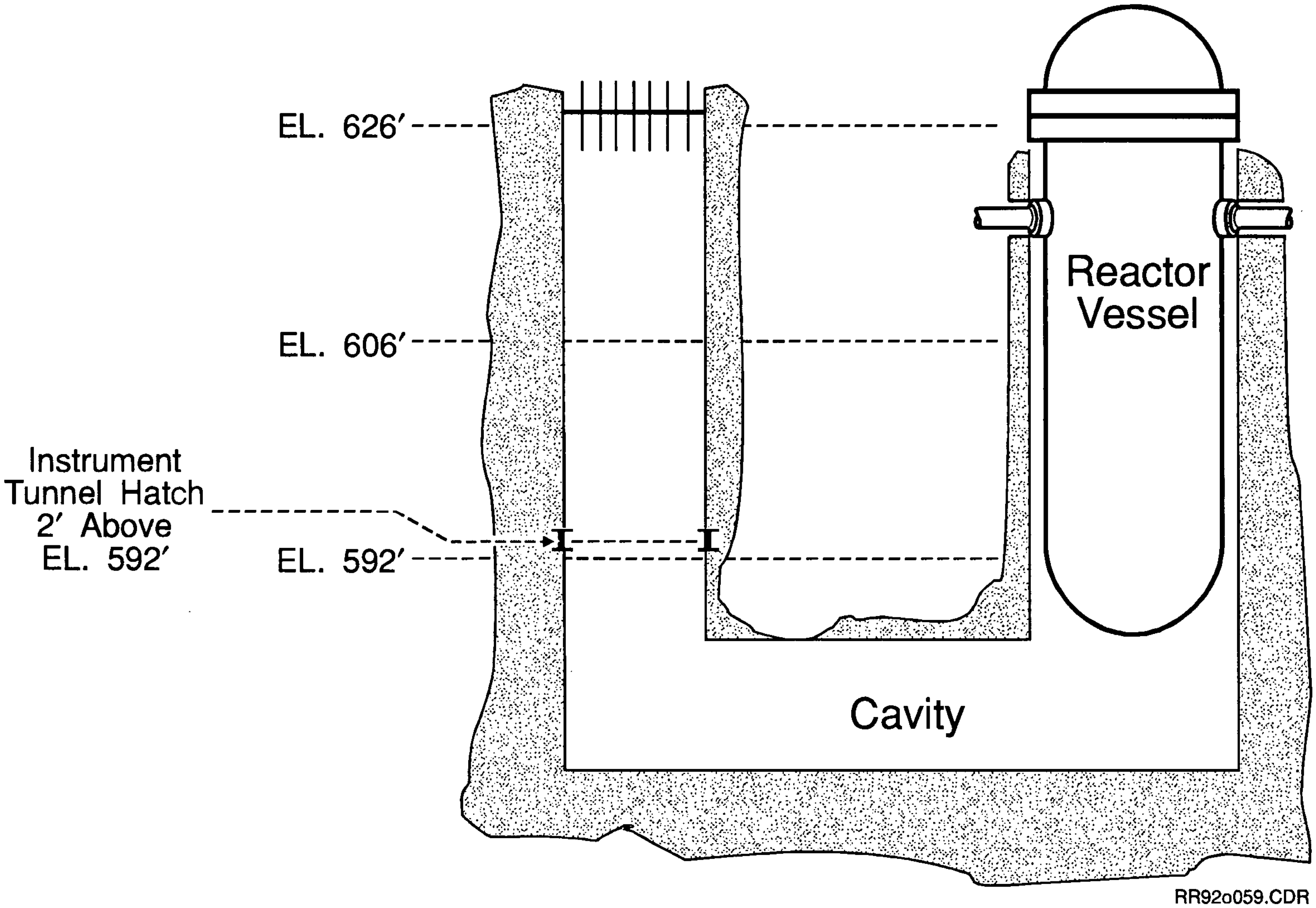
Table 4.1-1

EXAMPLES OF IMPORTANT PLANT DATA AND THEIR LOCATION IN THE KEWAUNEE MAAP PARAMETER FILE

Plant Data	Parameter File Section
Reactor Core (full power, UO_2 mass, Zr mass, mass of lower core plate and core support plate, fuel enrichment, fuel geometry)	•Core
Reactor Vessel (vessel mass, volume, wall thickness, mass of core barrel upper plenum internals, geometry	•Primary System
Primary System (hot and cold legs, volumes, elevations, scram set points)	•Primary System
Primary System (initial water level, P,T)	•Initial Conditions
Pressurizer	•Pressurizer
Pressurizer Relief Tank	•Quench Tank
Steam Generator	•Steam Generator
Accumulators (water mass, temperature)	•Engineered Safeguards
Containment Structure (volumes, areas and thicknesses, elevations, equipment mass, heat sinks, liner thickness, failure pressure)	•Upper Compartment (ACOMPT) •Lower Compartment (BCOMPT) •Annular Compartment (DCOMPT)
Containment Structure (cavity volume, floor area, basemat thickness)	•Cavity (CCOMPT)
Containment Structure (concrete properties, composition, rebar density)	•Concrete and Containment Shell
Containment Normal Conditions (T,P)	•Initial Conditions
Containment Systems (fan coolers, sprays)	•Generalized Engineered Safeguards
ECCS Injection/Recirculation (RWST water mass and temperature, charging, high-pressure and low-pressure injection, RHR HX details, pump curves, set points)	•Generalized Engineered Safeguards

FIGURE 4.1-2

KEWAUNEE CAVITY AND INSTRUMENT TUNNEL



4.2 Plant Models and Metbods for Physical Processes

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

The probabilistic models are embodied in the containment event trees (CETs) which consider all the systems and operator actions, including containment functional events, that respond to a core damage event to prevent or mitigate the release of radioactive fission products from the containment. The plant physical model is defined in the MAAP parameter file as discussed in Section 4.1.2. This parameter file provides MAAP with information required by the code to perform calculations of plant specific fission product transport and thermal hydraulic response to postulated accident sequences. It is also used to study the sensitivity of the source term to phenomenological uncertainties. The MAAP analyses are supplemented with phenomenological evaluation summaries to provide a complete physical representation of Kewaunee.

Results obtained with the probabilistic and physical plant models are closely linked. For instance, the CET structure depends on MAAP analyses to 1) define CET nodal success criteria, 2) establish timing of key events for human reliability analyses for understanding of sequence progression, and 3) determine the accident sequence outcome. Furthermore, sequences demonstrated by the quantification task to be either dominant contributors to the overall core damage frequency or of structural interest, become the basis for MAAP calculations in support of the source term analysis. Finally, MAAP analyses and phenomenological evaluation summaries are used to investigate the effect of phenomenological uncertainties on the source term assessment. The use of MAAP as suggested above provides the necessary deterministic complement to the probabilistic assessment. A detailed discussion of the containment event tree models is provided in section 4.2.1, while a closer examination of the MAAP models and the treatment of key phenomenological issues is presented here.

The Kewaunee IPE project employs a slightly modified version of MAAP PWR 3.0B Revision 18 to perform the containment and source term analysis. This code version was developed specifically for use in the Kewaunee IPE. This version of the 18.00 revision of the PWR MAAP 3.0B is very similar to the original 18.00 revision except that a few minor modifications in the thermal hydraulics routines were corrected. These modifications included providing the code with safeguards to allow the steam generator to go water solid, without compromising its ability to determine steam generator thermal hydraulic conditions. Other modifications included corrections made to the primary system routine to allow for better code performance during LOCA scenarios and enhanced decontamination factors for overlying water pools.

Source term analyses are performed following accident sequence quantification and designation of CET end states. CET end states that are representative of containment performance have their source terms quantified by Kewaunee MAAP analyses. The purpose of the source term analysis is to define and quantify the radionuclide release characteristics for a given accident sequence, which include specification of containment failure timing and fission product release magnitude. MAAP calculations provide release magnitude for selected fission product groups, release locations, release timing, and associated energy rates.

Since assumptions regarding key severe accident phenomena may dictate the analysis outcome, due consideration of phenomenological uncertainties is essential to the containment and source term analysis. The Kewaunee IPE methodology addresses the phenomenological issues in the following manner: 1) plant-specific phenomenological evaluations, 2) MAAP sensitivity studies, and 3) experimental studies of key phenomena. This three pronged approach provides a bounding assessment of source term release timing and magnitude.

Kewaunee phenomenological evaluation summaries are the principle means of addressing the impact of phenomenological uncertainties on plant response. These papers address a wide range of phenomenological issues and provide an in-depth review of plant specific features that influence the uncertainty or act to mitigate the consequences of such phenomena. The phenomenological evaluation summaries investigate both the likelihood of occurrence and the probable consequences of key severe accident phenomena. The phenomenological evaluation summaries are reviewed in section 4.2.5.3.

The phenomenological evaluation summaries are supported by available experimental information from open literature as well as information developed using the Fauske and Associates, Incorporated (FAI) experimental facilities. Results of the FAI experimental efforts are incorporated into the appropriate phenomenological evaluation summaries.

The purpose of sensitivity studies is to determine which remaining phenomenological uncertainties have a significant impact on the likelihood or timing of containment failure and the magnitude of the source term release. In performing Kewaunee MAAP calculations, a limited number of model parameters are investigated with respect to the influence 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. Generic Letter 88-20(1) and NUREG-1335(4) provide summaries of those parameters that have been judged to have a significant effect on containment failure and source terms. Section 4.4 of this document provides a detailed review of the Kewaunee IPE sensitivity analysis methods and results.

In summary, the integrated approach to the assessment of total plant response adopted in the Kewaunee IPE program links together probabilistic models in the CETs with physical plant models contained within MAAP. These models are supplemented through the use of Kewaunee phenomenological evaluation summaries to provide in-depth technical arguments that reduce phenomenological uncertainties and examine realistic plant response to severe accident phenomena.

4.2.1 Containment Event Trees (CETs)

The primary function of the CET is to describe the containment response to a core melt accident accounting for phenomenological issues, system response, and human behavior. This is accomplished by defining a set of top events along with their failure and success states. Each combination of top event success and failure states leads to a unique CET end state which provides information relevant to ex-vessel sequence progression, containment safeguards status, and source term release. Quantification of the CET is performed based on the core damage sequences that meet the screening criteria as discussed in section 4.3.3. This quantification results in the assignment of a CET end state for each of the selected Level 1 sequences. Following CET quantification, dominant sequences are selected for source term analysis. These selected sequences are representative of the entire spectrum of the Level 1 sequences quantified in the CET. Further discussion of the dominant sequence selection is contained in section 4.3. In addition to describing the accident sequence beyond core melt, the CET also serves as a directory for the binning of accident sequences in the source term analysis.

The general guidelines used for the development of a CET are summarized below:

1. CET top events and structure describe the containment response and account for human and system behavior that strongly influence the source term assessment.

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

3. The CET considers factors that dominate the containment response, thus the top events consider broad categories of systems and phenomena. For example, it is important to know whether or not water is available in the containment since this could have a major impact on debris coolability and fission product retention.

4. Containment failure modes resulting in early containment failure due to phenomenological uncertainties (i.e., early failure due to ex-vessel phenomena such as steam explosion, direct containment heating, etc...) as described in NUREG-1335[4] are not treated as separate top events, but rather through the phenomenological evaluation summaries and MAAP sensitivity studies.

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

Based on these guidelines, the structure of the CET has been arranged to first determine the status of the containment and then consider a series of nodes that describe the accident progression and containment safeguards availability. First, containment status is reflected through a decision of successful containment isolation, which has a direct bearing on source term release. Accident progression is then addressed through a series of nodes involving phenomenological concerns and the availability of systems pertaining to containment failure and source term analysis. To be consistent with the guidelines provided in Appendix A of NUREG-1335[4], the next node determines if the reactor coolant system (RCS) pressure is high or low at the time of vessel failure. A high pressure vessel failure would require consideration of some of the phenomenological issues in the Kewaunee Phenomenological Evaluation Summaries, namely, direct containment heating, vessel thrust forces, and steam explosions. The third node considers the timing of containment failure relative to vessel failure - an important parameter for source term release because this timing determines if natural fission product removal mechanisms, such as gravity, will have sufficient time to be effective in reducing source term. Subsequent decision nodes determine the status of those systems (RHR pumps, ICS, FCUs, etc...) that either prevent containment failure or mitigate fission product release given that containment failure has occurred. Details of the decision nodes (CET top events) and containment success criteria are included in the following sections as a final bridge between the Level 1 and 2 efforts, the following assumptions have been applied:

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

4.2.2 CET Top Events and Success Criteria

A number of top events can be considered that produce a CET that describes the ex-vessel sequence progression and that can be used in the source term binning process. Some of the CET top events that were given consideration were the availability of containment isolation for a given initiator, whether or not the RCS was at high pressure at vessel failure, and whether the containment failed early or late. Additionally, some severe accident phenomena (i.e., high pressure melt ejection) are modeled as CET top events. Based on a review of the NRC guidance and previous work, a CET for Kewaunee has been developed and is represented in Figure 4.2-1. The CET top events and their success criteria are defined below.

Containment Isolation Intact

Containment isolation, as used here, refers to the closure of containment penetrations to limit the release of radioactive fluids following an accident. Containment isolation is stated as part of the Level 2 sequence and is treated as a success or failure in the CET depending on the state of the specified sequence.

The impact of success or failure of this node is primarily on the timing of fission product releases. A failure to isolate containment results in an early fission product source term release following the onset of core damage. This source term release is characterized by the inability of the containment long term, natural fission product removal mechanisms (i.e., settling) to have an effect on minimizing the source term release.

High Pressure Melt Ejection (HPME)

The purpose of this node is to allow quantification of high pressure versus low pressure vessel failure. For those postulated severe accident sequences in which a substantial pressure is present in the primary system at the time of vessel failure, high pressure melt ejection could potentially displace some of the molten core debris into the seal table area in the annular compartment. Entrainment of debris occurs as the blowdown gas blows over the debris and entrains debris into the gas stream due to the large gas velocity. The gas velocity required to entrain molten debris can be characterized by the value of the superficial gas velocity required for supporting liquid films. Debris leaving the cavity is deposited in the containment annular compartment as the kinetic energy of the flowing gases decreases and the core debris becomes de-entrained.

Following RPV failure, the gas velocity and the likelihood of exceeding the "critical" velocity for entrainment increases with increasing RCS pressure. Therefore, sequences that result in high pressure melt ejection exhibit varying degrees of debris displacement and entrainment of debris from the cavity to the annular compartment. Typical low pressure sequences (i.e., large break loss of coolant accident (LOCA)) result in all of the debris remaining in the cavity. In addition to the RCS pressure, the degree of entrainment is influenced by the cavity and instrument tunnel geometries and the amount of debris present at the time of vessel failure. The determination of high pressure failure is based on a RCS pressure of 800 psig. This value is determined based upon the "cutoff" pressure for direct containment heating (DCH). Calculations have shown that if the RCS pressure is below this cutoff pressure at the time of vessel failure, entrainment of debris out of the cavity does not occur. Consequently, if the RCS pressure is above this cutoff pressure, some of the debris becomes entrained in the blowdown stream and relocates to the seal table area.

The PDS definition for each core melt sequence from the Level 1 analysis includes an indication of RCS pressure at the time of core damage. Coupled with additional knowledge of the Level 1 sequence progression, this will define the likelihood of HPME after vessel failure. "Success" for this node occurs if the primary system pressure, prior to vessel failure is above 800 psig and debris is transported out of the cavity. "Failure" is defined as the RCS pressure being below the cutoff pressure and the debris remaining in the cavity. Success and failure as stated in this node is not intended to imply that a high pressure melt ejection is more desirable, rather to point out where the debris is located.

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

Late or No Containment Failure

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

The NRC, in NUREG/CR-2300[3], has identified a number of phenomena that could potentially result in an early containment failure. Due to the uncertainty surrounding these phenomena, the NRC has recommended that they be considered in the CET. Since the likelihood of early containment failure due to these phenomenological uncertainties is highly dependent on plant specific containment geometries, the present methodology treats these items individually through phenomenological evaluation summaries. These summaries provide a detailed Kewaunee specific analysis of the various phenomena and discuss the likelihood and consequences of the phenomena.

Success of this node occurs if none of the pertinent phenomenological uncertainties result in early containment failure as discussed in the summary papers. For Kewaunee, it has been found that the occurrence of the phenomena listed in Table 4.2-1 and associated sensitivities do not threaten the containment integrity nor result in an early containment failure. Therefore, this node has a probability for success of 1, for CET quantification and best-estimate source term analysis.

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

RHR Pumps and Heat Exchangers

The operation of the RHR pumps and heat exchangers in the recirculation mode after vessel failure cools the debris bed on the cavity floor. Debris coolability is achieved when the heat removal rate from the debris bed exceeds the debris internal heat generation rate. Thus, a coolable debris configuration eventually results in a frozen or solid debris bed, while a non-coolable configuration produces a molten debris pool. Debris heat removal is achieved through several possible mechanisms. Conductive heat transfer will occur at the interface of the core debris and concrete bodies. If this is the dominant mode of heat removal, then chemical interactions between the concrete and the debris results in significant concrete ablation. Radiative heat transfer from the debris to the surrounding gases and structures also occurs if the debris bed is not sufficiently cooled. Finally, convective heat transfer characterized by nucleate boiling is very effective for debris beds immersed in a water pool.

Short term debris coolability can be obtained if the debris exiting the reactor vessel after vessel failure falls into a water pool and is rapidly quenched. However, since debris coolability occurs via nucleate boiling of the overlying water pool, it eventually boils off the water pool if the pool is not replenished. Failure to replenish the water pool results in exposing the core and eventually reheating the debris bed. Therefore, RHR injection without recirculation is classified as a failure of this node on the CET.

To replenish the water pool and maintain long term coolability, the RHR recirculation system must supply a water flow rate that exceeds the steaming rate. Thus, the water supply must be sufficient to match decay heat generation. The time available to establish RHR recirculation is dependent on the debris distribution and the initial amount of water available in the cavity at the time of vessel failure. Typically, by the time water pool dryout occurs, decay heat levels are within the heat removal capacity of the fan coolers or RHR heat exchangers. Therefore, with the operation of the RHR pumps and heat exchangers or the RHR pumps with the use of a fan cooler for heat removal capabilities, debris coolability can be achieved.

Some uncertainty exists with regard to the ability to cool the molten core debris. The NRC has stated that a debris depth of less than 25 cm (10 in) may be considered coolable. The Kewaunee containment geometry is such that if 100% of the core debris is postulated to evenly spread in the cavity, the debris depth will be 26 cm, just slightly greater than 25 cm. Therefore, due to the conservatism of 100% of the core being expelled from the vessel, if an adequate supply of water exists to ensure that the debris is submerged in a water pool throughout the accident, debris coolability can be assumed.

Success of the RHR and RHR heat exchanger nodes will prevent core-concrete attack and the ensuing generation of hydrogen and airborne fission product aerosols. Failure of this node will lead to substantial concrete ablation and could eventually breach containment due to failure of the cavity basemat. Success of this node also has an effect on fission product scrubbing and containment heat removal. Depending on the debris distribution after vessel failure, the RHR pump could provide some form of fission product scrubbing in terms of an overlying pool of water on the debris bed. Due to a high pressure melt ejection, some of the debris is dispersed into the annular compartment and therefore, the RHR pump does not have an effect in terms of fission product scrubbing. If the vessel fails at low pressure and an RHR pump is recovered soon afterward, the RHR pump is able to provide an overlying pool of water in the cavity and thus provides some form of fission product scrubbing. In RHR recirculation, the RHR heat exchanger performs containment heat removal in the form of cooling the water located in the cavity and on the lower compartment floor.

Containment Sprays With RHR Heat Exchanger

The quantity and type of radionuclides released following a reactor vessel failure are sensitive to the mechanisms available for fission product scrubbing. Fission product scrubbing refers to the removal of radioactive fission products from a gas space through the use of some form of filtration. In pressurized water reactors (PWRs), no filtered containment vents exist, therefore, fission product scrubbing is achieved through the operation of ICS. Water used in this manner is a very effective tool in removing airborne fission products. As mentioned in the RHR pump node, fission product scrubbing can also occur due to the presence of an overlying pool of water on a debris bed. Since the reactor cavity at Kewaunee is not capable of being flooded via overflow from the lower compartment, the ICS system scrubs the airborne fission product aerosols without necessarily providing an overlying pool of water on the debris bed.

Success of the ICS and RHR heat exchanger node implies that the operation of the sprays have effectively reduced the airborne fission product content. Failure of the ICS node implies that the ICS is not available to reduce the airborne fission product concentration. If the RHR pumps are operating, however, the airborne fission product inventory is not as high due to the presence of the water pool in the cavity. Since the ICS pump takes suction from the discharge of the RHR heat exchangers in recirculation mode, it also performs a containment heat removal function due to the operation of the RHR exchanger. Also note that since the ICS pump takes water from the RHR pump in recirculation mode, failure of the RHR pump also means failure of long term ICS recirculation.

Fan Coil Units

The interaction of core debris with containment water pools, mechanical structures, and the atmosphere results in heat up and pressurization of the containment. This pressurization is a function of containment free volume, the rates of condensible and non-condensible gas generation, and the rate at which the containment temperatures are increasing. Since the gas generation and temperature rise can be characterized by the level of decay heat within the core debris, it is necessary to establish some form of containment heat removal that meets or exceeds the decay heat generation. Failure to establish some form of containment heat removal results in sustained containment pressurization and eventual failure of the containment boundary.

Containment heat removal can be achieved through the use of any number of systems or combination of systems. For example, containment heat removal can be achieved with any one of the following combinations of systems:

- 1 containment fan coil unit,
- 1 RHR pump with the RHR heat exchanger, or
- 1 ICS pump taking suction from the RHR heat exchanger discharge.

Failure of all the safeguard systems listed above results in a late containment failure, while successful operation of the FCUs or any combination of the systems listed above results in containment pressures and temperatures well below the limit necessary for containment failure. However, FCUs without RHR recirculation are not able to prevent molten core-concrete interactions (MCCI).

4.2.3 CET Structures and End States

The CET top events, as described in section 4.2.2, have been arranged in a manner that takes into account the sequence progression and provides insights into the containment response to a postulated core damage accident. The combination of top events success and failure states leads to 90 possible CET end states. These end states provide a qualitative description of the ex-vessel sequence progression and source term release. Among these end states, a number of possible failure modes are defined below:

<u>Leakage</u> - The containment integrity is not challenged due to either overpressurization or basemat penetration. Minor releases of airborne fission product may occur along normal leakage pathways.

<u>Early Containment Failure</u> - Containment failure occurs immediately due to an isolation failure. This results in an early fission product release without the benefit of natural occurring fission product removal mechanisms or fission product scrubbing from ICS.

<u>Early Containment Failure - Reduced Fission Product Release</u> - This is essentially the same as the above end state, except that fission product scrubbing due to the operation of the ICS is credited.

<u>Late Containment Failure on Overpressure</u> - Containment failure due to overpressurization occurs resulting in fission product release. No fission product scrubbing has been available.

<u>MCCI Induced Containment Failure</u> - Late containment failure is expected due to basemat failure resulting from prolonged molten core-concrete interactions (MCCI). This assumes that basemat

failure occurs prior to an overpressurization failure and does not credit fission product scrubbing. Generally, this failure mode occurs within the 48 hr mission time.

Containment Failure on Overpressure Beyond Mission Time - Containment failure due to overpressurization occurs beyond mission time. Failure will result in fission product release if recovery actions are not taken.

4.2.4 CET Quantification

As a precursor to the source term analysis, CET quantification is performed using the Level 2 sequences that meet the screening criteria presented in section 4.3.3. The quantification of the CET assigns each sequence, along with its frequency, to a particular CET end state. The end states and their cumulative frequencies form the basis for binning of "like" sequences. This binning process cuts down on the number of sequences that need to undergo MAAP analysis.

The CET quantification process involves following the event tree branching logic for a given sequence to arrive at a particular CET end state. Since the combination of CET top event success and failure states leading to a particular CET end state are largely predetermined by the Level 1 sequence definitions, the split fraction for each CET branch (top event) can readily be assigned as 0's and 1's. Generally, sequences of similar initiating events that fall into the same CET end state have similar source term consequences and therefore, analyzing one of these sequences allows the rest of the sequences to be bound by the analyzed sequence. In some cases, different CET end states can be binned together. The frequencies for all the sequences with the same CET end state are summed up to determine the total frequency for each end state. The results of this quantification are presented in Figures 4.3-1a through 4.3-1c. They are described in detail in section 8.2 of the Kewaunee PRA study.

4.2.5 Containment Failure Characterization

Plant-specific phenomenological evaluations have been performed in support of the Kewaunee IPE to determine the likelihood of all postulated containment failure modes and mechanisms identified in NUREG-1335[4]. These detailed evaluations were performed systematically to address the controlling physical processes or events specific to the Kewaunee configuration. Modeling and bounding calculations, based upon extensively compiled experimental data, phenomenological uncertainties, and complemented with MAAP calculations in some cases, comprise the general approach taken in these evaluations. Several postulated containment failure mechanisms are demonstrated, through the phenomenological evaluations, to be inconsequential for the Kewaunee containment. These potential failure mechanisms are considered to be very unlikely to occur at Kewaunee since the predicted pressures resulting from a realistic assessment of these failure mechanisms are far less than the containment ultimate strength.

The failure mechanisms considered unlikely to occur within the Level 2 mission time are hydrogen detonations and deflagrations, direct containment heating, steam explosions, molten core-concrete attack, thermal attack of containment penetrations, and vessel thrust forces. More likely to occur are containment overpressurization, containment bypass sequences, and failure to isolate containment. Table 4.2-1 summarizes the results of the containment failure mode evaluations.

4.2.5.1 Containment Ultimate Strength

A plant-specific structural analysis of the Kewaunee containment was conducted to determine the ultimate internal pressure capacity and the likely failure locations associated with this pressure. The results of the ultimate pressure analysis are presented in Figure 4.2-2 in terms of a containment fragility curve. This curve shows the total failure probability for individual components as a function of containment pressure. From the total failure probability curve, it can be determined that the total mean (50%) failure pressure is 166 psia while the 5% lower bound and the 95% upper bound are 137 psia and 192 psia, respectively.

Source term analysis assumes that containment failure occurs at 137 psia due to membrane stresses in the cylindrical section of the shell exceeding the ultimate stress of steel material. Figure 4.2-2 shows that the ellipsoidal lower head is the more probable failure location, although this failure location is dominant solely due to a much larger uncertainty. This, coupled with the fact that a source term release from the lower head would benefit from the scrubbing characteristics of the soil means that the best estimate and more conservative failure location is in the mid-height region of the cylindrical steel shell.

4.2.5.2 Likely Containment Failure Modes

Containment Overpressurization

Containment overpressurization, defined as a failure mode caused by steaming and/or non-condensible gas generation, is a potential containment failure mode within the Level 2 mission time at Kewaunee. Depending on the specific accident sequence characteristics, overpressurization failures may be observed across a wide range of event times. The potential for containment overpressurization failure is dominated by failure of containment heat removal systems.

Overpressurization failure is expected to be a slow mechanism, such that the containment failure pressure is approached gradually. The resulting stresses on the containment steel shell will likely result in a large catastrophic failure of the steel shell. This is supported by the experimental evidence (i.e., Sandia 1/8 steel shell experiment) for free standing steel shell containment structures. As discussed in section 4.2.5.1, the most conservative and likely failure location is the cylindrical portion of the steel shell.

Containment Isolation Failures

Containment isolation failure is a possible containment failure mode at Kewaunee. Containment isolation failures refer to mechanical or operational failures to close containment fluid system penetrations which communicate directly with the containment or primary system prior to, or following the initiation of core damage, in order 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, required to be closed during power operation is left unisolated.

2. A fluid line that has isolation valves that are required to be closed on an isolation signal is not isolated because the valves fail to close.

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

In all the above conditions for fluid systems, all check valves in fluid lines must also fail to close in order for impaired containment isolation to occur. Critical containment penetrations (i.e., those that lead to significant fission product releases out of containment if they fail to isolate) are identified based on the following screening criteria:

1. The line penetrating containment is a containment sump or reactor cavity sump drain line, or

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

Failure to isolate containment is addressed in the CETs.

Containment Bypass

Containment bypass is another possible containment failure mode at Kewaunee. Containment bypass refers to failure of the pressure boundary between the high pressure RCS and a lower pressure line penetrating containment. This results in a direct pathway from the RCS 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 emergency core coolant system (ECCS) recirculation for long term core cooling. The likely mechanisms for this failure mode identified for Kewaunee as being significant in terms of frequency and potential hazardous consequences, are (1) an interfacing systems LOCA and (2) steam generator tube rupture sequences where the faulted steam generator cannot be isolated.

4.2.5.3 Unlikely Containment Failure Modes

Hydrogen Combustion

Potential detonability and flammability of the Kewaunee containment atmosphere is analyzed as part of the Kewaunee IPE. Detonation is evaluated based on geometric configuration and detonation cell width scaling. Both of these methods conclude that the likelihood of deflagration to detonation transition (DDT) is very low. For large dry PWR containments in general, the probability of this occurrence is so low that the consideration of containment failure due to hydrogen deflagrations and detonations is not needed in the Level 2 CETs. It is far more likely that combustible gases are consumed within containment by deflagration rather than detonation.

Plant specific analyses were performed for a station blackout at Kewaunee, the worst case scenario in terms of hydrogen production. Results demonstrate that not enough hydrogen would accumulate to produce a deflagration that could challenge the containment ultimate pressure capacity. Furthermore, the containment would most likely fail due to over-pressurization long before such a large amount of hydrogen could accumulate.

Direct Containment Heating (DCH)

Direct containment heating (DCH) is a postulated event of rapid heat transfer between finely fragmented core debris and the containment atmosphere, assuming: 1) post core melt reactor pressure vessel failure occurs at a high pressure, and 2) the high pressure melt ejection (HPME) causes extensive debris dispersal. DCH has been hypothesized as an early containment failure mechanism because the stored energy of the debris, including the potential energy released through debris oxidation is enough to cause high containment pressures if a large quantity of the core inventory participates. Thus, the extent of pressurization depends upon:

- the amount of debris that is discharged at vessel failure;
- the containment geometry, which could be conducive to, or an impediment to, debris dispersal beyond the cavity; and
- the fraction of the debris that could be finely fragmented and dispersed in the containment atmosphere.

The relevant experimental data for DCH and a mechanistic model for debris dispersal have been reviewed to evaluate the response of the Kewaunee containment during a HPME. Given the necessary RCS conditions for a HPME, containment structures have a first order (dominant) mitigating influence on the potential for DCH. The use of mechanistic models for debris dispersal, which take into account entrainment from the cavity and de-entrainment at the tunnel exit, to evaluate the containment response to a HPME, show the resulting pressurization to be less than the value necessary to challenge containment integrity.

Steam Explosions

Separate approaches are used to address in-vessel and ex-vessel steam explosions. The IDCOR work, which is consistent with the recommendation of the NRC sponsored Steam Explosion Review Group (SERG), 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 has been concluded that the slumping of molten debris into the RPV lower plenum does not result in sufficient energy release to threaten the vessel integrity, and hence, does 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 mechanisms are not of sufficient magnitude to threaten containment integrity. Shock waves generated in the cavity by ex-vessel explosive interactions decay prior to reaching the containment boundary.

Vessel Thrust Forces

The issue for this phenomenon is whether the thrust forces generated following core damage and reactor vessel lower head breach could become sufficient to cause the vessel to shift position and tear containment penetrations. The approach taken is to: 1) estimate the thrust forces generated during corium ejection, 2) compare this estimate to the dead weight of reactor vessel, and 3) determine if this postulated phenomenon significantly challenges containment integrity at Kewaunee.

The bounding analysis for the magnitude of the thrust forces when molten corium is ejected from the failed vessel at high pressure indicates that this force could just barely lift the dead weight of the vessel itself, given a credible break size in the RPV and a complete melt of the fuel and lower core support materials. This analysis is performed assuming no credit for the series of restraints that are designed to prevent any vertical or horizontal movements of the reactor vessel. Even if the vessel could shift, the Kewaunee containment is configured so that the reaction forces cannot be transmitted to the containment wall. Therefore, this postulated containment failure mode is bounded by plant design and is not capable of threatening containment integrity.

Molten Core-Concrete Interactions

If allowed to continue, core debris attack of concrete structures results in extensive erosion of the concrete, leading to one of two late containment failure mechanisms: 1) penetration of the containment basemat, or 2) sufficient deterioration of the load carrying capabilities of the primary shield wall causing the reactor vessel to shift significantly, inducing a gross mechanical failure of the mechanical penetrations connected to the reactor vessel.

This phenomenon was addressed for Kewaunee through a plant specific analysis that incorporated a review of the available experimental data and solution of a bounding calculation that 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 does not occur until well after the 48 hour Level 2 mission time.

Thermal Attack of Containment Penetrations

Containment penetration thermal attack is a postulated containment failure mechanism in which non-metallic materials in containment penetrations are exposed to elevated temperatures over a period of time causing the seal performance to decline and leak excessively. The impact on containment failure timing depends on the gas temperature, the exposure time, and the characteristics of the materials involved. A detailed Kewaunee plant specific analysis was performed that also considered the potential of the penetrations to become submerged in a molten pool.

The operational limit of the non-metallic materials has not been shown to be exceeded by the maximum gas temperatures predicted for containment compartment regions during severe accident sequences. The evaluation of debris dispersal in conjunction with the location of the mechanical and electrical penetrations reveals that it is unlikely for the penetrations at Kewaunee to be in direct contact with molten debris dispersed during a HPME scenario. The majority of the entrained debris becomes de-entrained at the seal table enclosure. There are no direct paths by which molten corium could contact any of the containment penetrations. Hence, thermal loading of penetration non-metallic materials do not cause degradation and leakage of fission products from containment under conditions expected at Kewaunee during a severe accident.

Liner Melt-Through

The potential failure of the steel shell due to direct contact with molten corium was analyzed to address the potential of an early containment failure mechanism. It is postulated that during a high pressure vessel blowdown, debris becomes entrained in the gas stream and exits the cavity at the seal table structure. Since the seal table at Kewaunee is situated in the annular compartment, this would put corium in a location where it could potentially come into contact with the steel shell.

The analysis concludes that, due to the large amount of structure located in and around the seal table structure, the debris that was ejected during a HPME becomes de-entrained. The debris that manages to escape the seal table area is expected to be small airborne particles rather than a large, monolithic debris bed capable of ablating the containment walls. Therefore, these particles are unable to generate the heat necessary to melt through the steel shell.

4.2.6 Post Core Damage Accident Progression

Loss of coolant from the RCS, either through a break in the primary system coolant boundary or a loss of heat sink (which in turn promotes over-pressurization of the RCS and subsequent loss of coolant through the safety valves), coupled with failure to inject the RWST, eventually results in uncovering the reactor core. Core damage occurs shortly afterwards, 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 once 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 the cooler material below. Eventually it remelts and moves further downward. This downward progression is mainly a function of the temperature 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 for 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 reactor vessel lower head, relocation of the molten core into the lower head is assumed to lead directly to failure of the reactor vessel; no attempt is made to take credit for potential in-vessel recovery. If the RCS is at high pressure at the time of vessel failure, then the high pressure melt ejection could possibly displace or entrain core debris out of the cavity. However, a majority of this debris is de-entrained by the seal table and other containment structures in the vicinity of the seal table. If the RCS is at low pressure at the time of vessel failure, no core debris will exit the cavity region.

If no water is available to cool the core debris or if the debris dries out, then molten core-concrete interactions (MCCI) take place. Concrete decomposition generates non-condensible gases as well as water from the concrete, resulting in additional chemical heat generation and hydrogen evolution due to further 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 none of the containment heat removal systems are available, this pressurization will eventually induce containment failure.

The time required to fail the containment by overpressurization depends upon the steaming rate and the rate of non-condensible gas generation. The failure mechanism associated with containment overpressurization is due to exceeding the ultimate strength of certain key structural components or attachments. This limit is most likely to be approached gradually, but due to the failure characteristics of steel shell containments, the failure is most likely to be a large catastrophic failure of the containment shell.

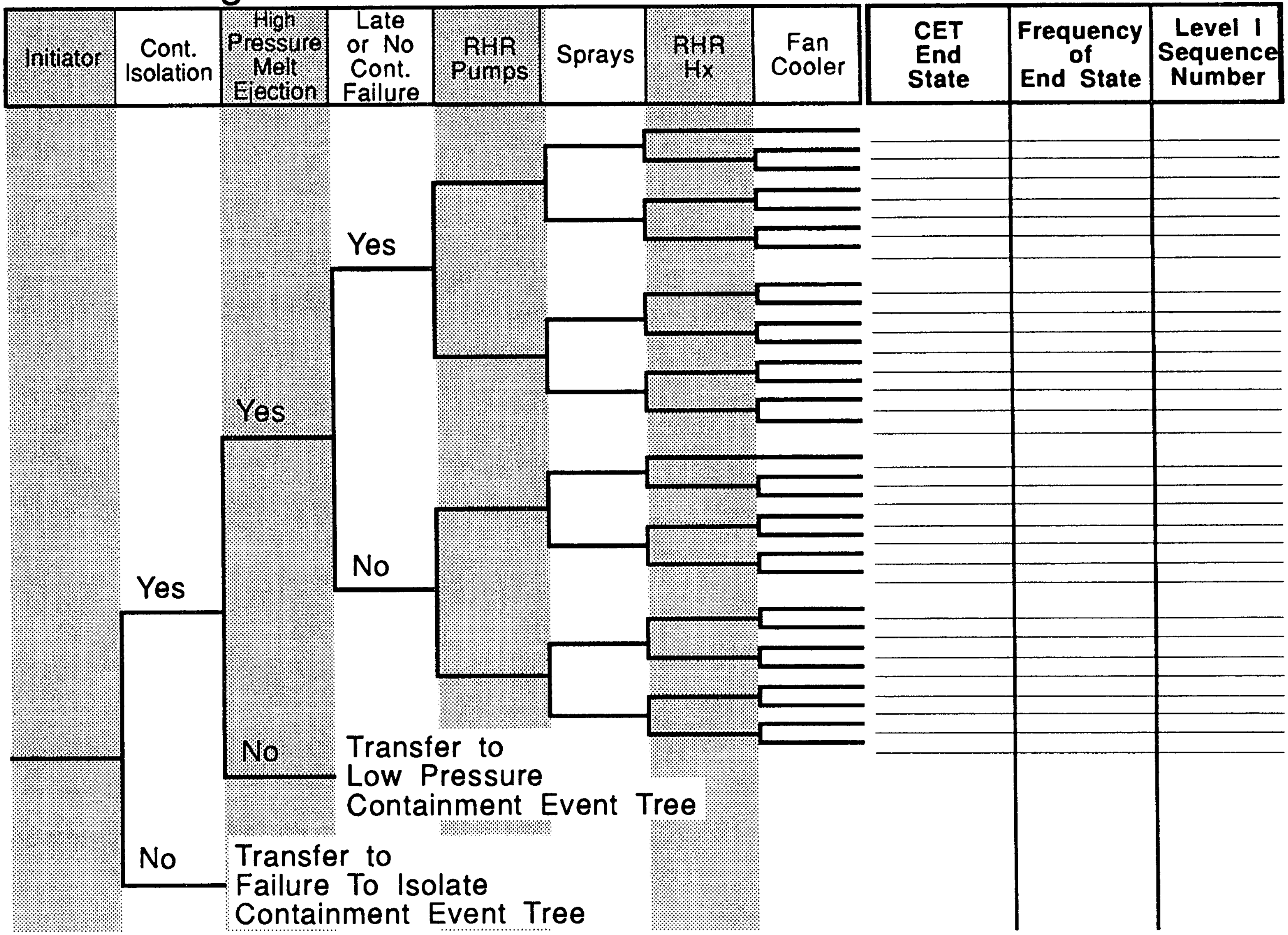
The severity of the source term depends strongly on the containment failure timing. Failure in the immediate time period surrounding vessel failure is clearly the most serious in terms of source term, since the overall airborne fission product mass produced during a severe accident is never larger than it is in the time frame directly after vessel failure. Substantial fission product retention through naturally occurring deposition mechanisms (i.e., sedimentation, impaction, etc.) is facilitated for a late containment failure.

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

FIGURE 2-1a

KEWAUNEE CONTAINMENT EVENT TREE (CET)

Wisconsin Public Service --- Kewaunee
High Pressure -- Containment Event Tree

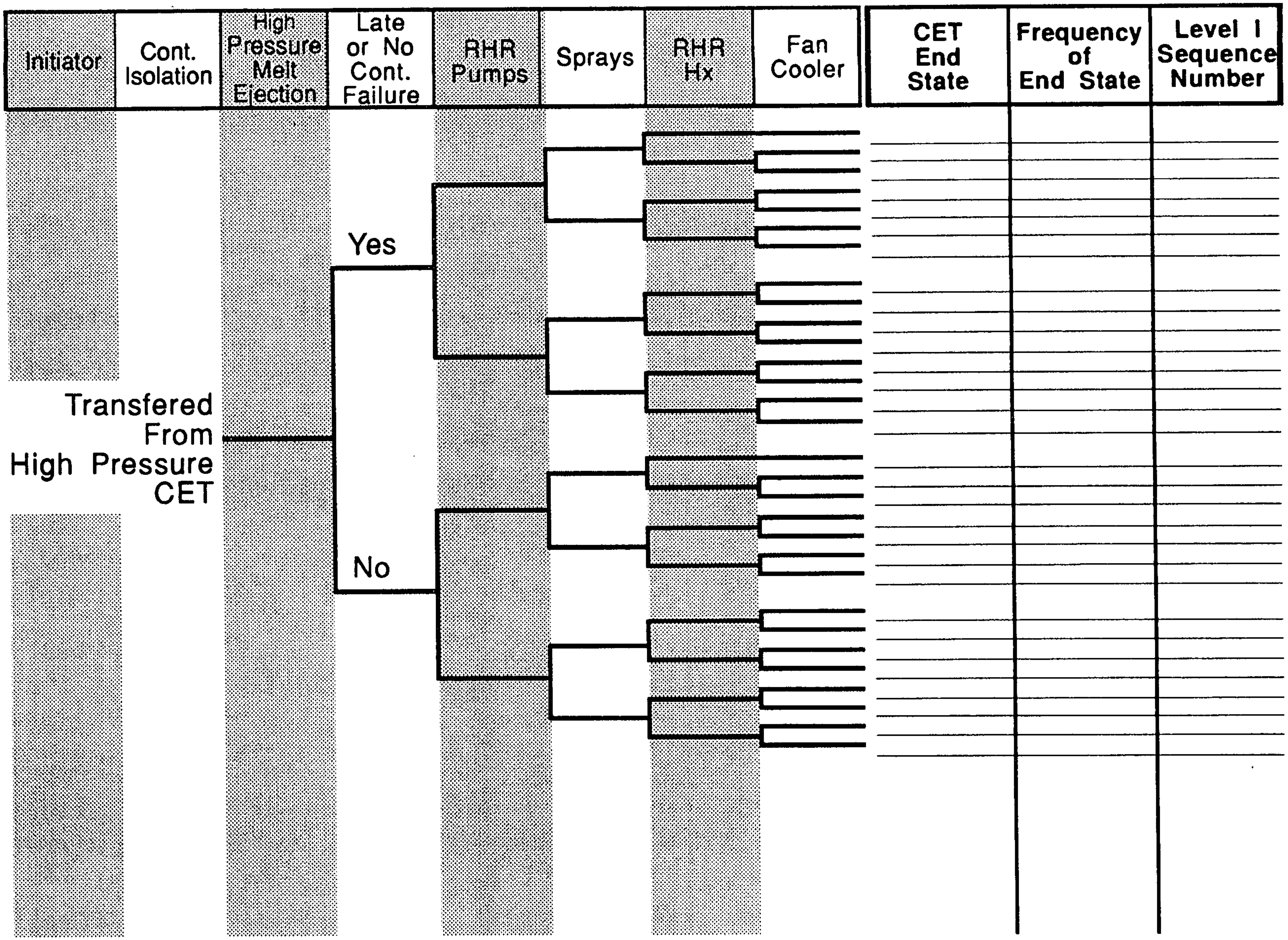


RR92J041.CDR

FIGURE 4.2-15

KEWAUNEE CONTAINMENT EVENT TREE (CET)

Wisconsin Public Service --- Kewaunee
Low Pressure -- Containment Event Tree

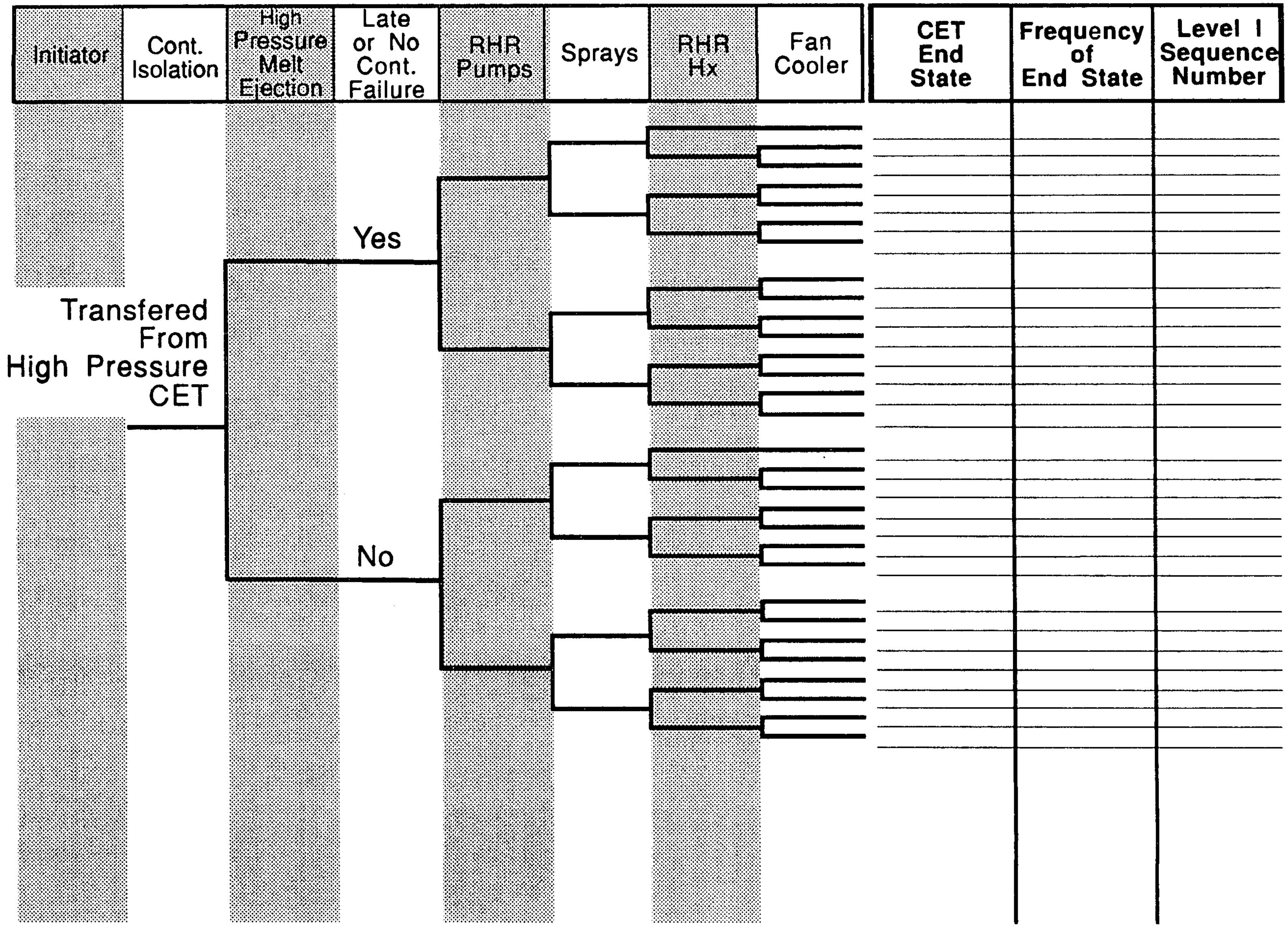


RR92J043.CDR

FIGURE [illegible]c

KEWAUNEE CONTAINMENT EVENT TREE (CET)

Wisconsin Public Service --- Kewaunee
Failure To Isolate -- Containment Event Tree



RR92J045.CDR

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

TABLE 4.2-1

PHENOMENOLOGICAL EVALUATION SUMMARIES ON POSTULATED CONTAINMENT FAILURE MODES (Continued)

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 RPV	RPV failure and failure size	No or limited RPV displacement Challenge bounded by design basis
6. Thermal Loading on Penetrations	Degradation of non-metallic components	Containment breach; leakage path	Magnitude and duration of elevated containment gas temperature Behavior of non-metallic materials at high temperature	No loss of containment integrity expected Potential for long term loss of electrical functionality
7. Liner Melt-through	RPV failure Debris dispersion Influence of containment structures	Containment breach, leakage path	Extent of debris dispersal and de-entrainment	No loss of containment integrity expected

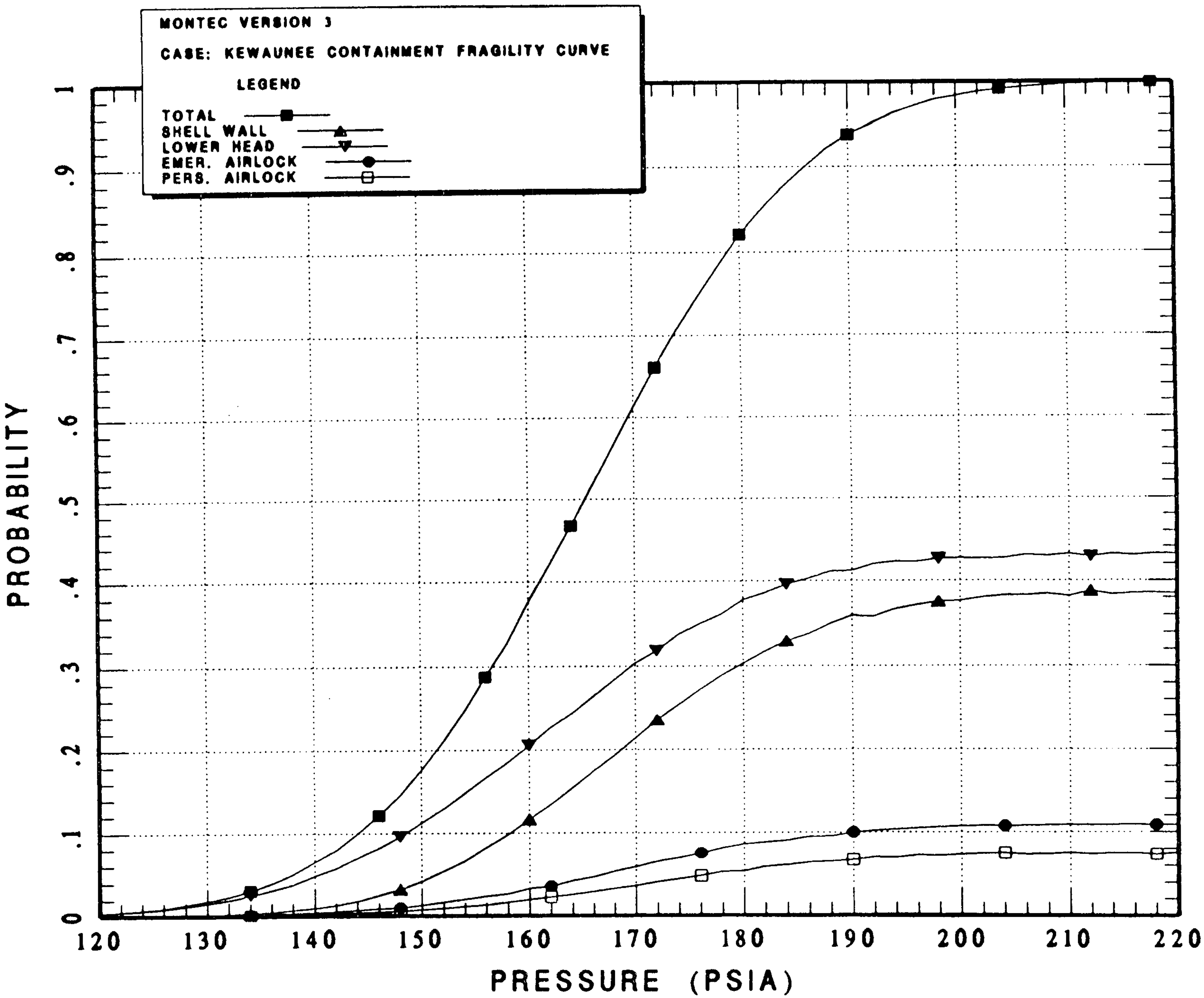
TABLE 4.2-1

PHENOMENOLOGICAL EVALUATION SUMMARIES ON POSTULATED CONTAINMENT FAILURE MODES (Continued)

FAILURE MODE	PHENOMENA	ISSUE/FAILURE MECHANISM	MAJOR UNCERTAINTY	IMPACT
8. 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
9. 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
10. Containment By-pass	Interfacing Systems LOCA SGTR	FP release path that does not pass through containment air space	FP deposition in building outside containment Number of ruptured SG tubes Size location of break outside containment Water scrubbing at break location FP deposition outside containment	Low probability of direct FP path to environment or auxiliary building

FIGURE 4

KEWAUNEE CONTAINMENT FRAGILITY CURVE



4.3 Kewaunee Level 2 Source Term Analysis

4.3.1 Introduction

The purpose of the Kewaunee source term analysis is to: 1) define the types of severe accidents that can occur and their frequency of occurrence, and 2) quantify the consequences of each event in terms of a radiological release fraction. Before source term calculations are actually performed, the Level 1 results are reorganized into a suitable form. This involves a screening process (see section 4.3.3) and grouping of similar sequences into accident sequence "bins" (see section 4.3.4) to reduce the total number of sequences to be analyzed. Source term quantification is then performed by analyzing a single, representative accident sequence from each bin. The remainder of this section discusses the results of the Kewaunee Nuclear Plant source term analysis.

4.3.2 Overview

To arrive at fission product releases, a number of phenomena and fission product pathways must first be considered through all phases of the severe accident progression. Fission products must pass through multiple barriers located along the release pathways. These barriers include the oxide fuel itself, the fuel pin cladding, reactor coolant system (RCS), and containment and auxiliary building structures.

Transport of fission products from the initially intact fuel matrix to the environment can best be presented by considering the chronological progression of a core melt scenario. During a core melt accident, the transport of fission products, including their transport state and the timing of their release from the intact or molten fuel, varies significantly between volatile and non-volatile fission products and noble gases. Due to the chemical characteristics of volatile fission products, a substantial fraction of these diffuse through the oxide fuel structure and are released into the fuel pin-cladding gap. Non-volatile fission products, on the other hand, have a much lower affinity for diffusion through the fuel oxide, and thus, are retained within the fuel material. Eventually the fuel cladding ruptures due to pressure buildup from the volatile gases being released from the fuel material. Concurrent with the cladding failure, is a release into the primary system of the accumulated volatile fission product vapors and the resident noble gases.

In a steam environment, most of the volatile fission product vapors condense and form aerosols. These released fission products may be transported to the containment or auxiliary building atmosphere via various flow paths such as the pressurizer relief and safety valves, pressurizer relief tank rupture disk, or more directly via pathways due to breaches in the RCS boundary. In the case of the volatile fission products, significant retention within the RCS could occur as the aerosols deposit on the RCS structures. However, these deposited fission products may revaporize late in the accident sequence and make their way out of the RCS following established pathways generated throughout the course of the accident.

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

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

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

As stated previously, the purpose of the Kewaunee source term analysis is to quantitatively describe the magnitude and composition of fission product releases from the containment as a result of a severe core damage accident defined by the Level 1 results. To adequately address the complexities associated with fission product transport and release, and to account for the specific Level 1 sequence descriptions, this analysis relies on the Modular Accident Analysis Program (MAAP)[45]. This code couples plant thermal hydraulic response with fission product behavior to properly model the feedback between the two. Furthermore, MAAP analyzes all phases of a severe accident, including the impact of operator actions and engineered safety features on the RCS and containment.

In regard to fission product transport, MAAP begins tracking the fission products as they exist in the intact fuel matrix. This initial fission product inventory is organized by chemical properties into 12 fission product groups within MAAP. The initial inventory of each of the 12 groups specific to Kewaunee as derived for the Kewaunee MAAP parameter file is shown in Table 4.3-1. The fission product quantities were taken from another Westinghouse 2-loop PWR with similar fuel and operating characteristics, such as cycle length. This total inventory of fission products is generally characterized as noble gases (group 1), volatile fission products (groups 2, 6, and 11) and non-volatile fission products (groups 3, 4, 5, 7, 8, 9, 10, 12). The Kewaunee source term analysis performed in this section reports the mass fraction released for each of these three categories.

4.3.3 Source Term Sequence Selection

Although source term analysis culminates with the quantification of fission product release magnitude, Level 1 results are processed through a sequence selection process before source term calculations are performed. A significant portion of this processing effort involves reorganizing the Level 1 results into a form suitable for performing the source term calculations. This effort includes containment event tree (CET) quantification and grouping of similar sequences into accident sequence bins (CET end states) to reduce the total number of sequences that need to be analyzed. Source term quantification is then performed by analyzing a single sequence from each bin or CET end state. Each of these steps are discussed in greater detail in the following sections.

Sequence selection for Level 2 analysis entails the following:

(A) Screening process: From the set of all sequences quantified during the Level 1 effort, select only a certain number of dominant sequences to actually consider for source term analysis. The screening process used on the Kewaunee Level 1 results reflects the guidelines suggested in NUREG-1335[(4)].

(B) Sequence binning: From the screened sequences identified above, quantify and group them using the CET. Sequences that result in the same CET end state have similar source terms.

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

(D) Release category: From the source term results generated from the MAAP runs selected in Step C, assign a release category to each end state identified in Step B.

A screening process (see section 4.3.3.1) is applied to the core damage sequences identified in the Level 1 analysis to reduce the number of sequences that need to be analyzed. Once the dominant sequences are determined through the screening process, these sequences are then put through the CET to determine which CET end states need to be analyzed.

CET end states group sequences with expected similar terms. This is referred to as the "binning" process. Representative sequences are then selected from each bin, or CET end state, to make MAAP runs and derive a representative fission product source term for each accident bin. The sequences that are analyzed as part of the sequence selection step are referred to as "analyzed" sequences. The remaining sequences in each bin are considered "bounded" sequences. The accident sequence characteristics of the bounded and analyzed sequences within each accident bin are expected to yield very similar source term results.

Once the sequences are analyzed, release categories are assigned to describe the containment failure mode and the fission product release magnitude. These release categories are assigned to each CET end state based on the results of the MAAP analysis. The source term results and assigned release category for a CET end state are based on MAAP results for the analyzed sequences. The frequency for a release category is the sum of the frequencies for both the analyzed and bounded sequences assigned to that release category.

4.3.3.1 Screening Process

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

The top Level 1 plant damage states in order of initiating event are listed in Table 3.1.5-1. The table includes all plant damage states with a frequency greater than 1.0E-10 per reactor year. This list of sequences account for more than 99% of the total core damage frequency. The table below lists the NUREG-1335[(4)] screening criteria that are applied to the plant damages listed in Table 3.1.5-1. Section 2.1.6 of NUREG-1335[(4)] describes the screening process for the Level 1 results. The following table lists the NRC requirement which is met during the Level 2 screening process.

<u>**NUREG-1335**</u>

Any systemic sequence that contributes 1E-7 or more per reactor year

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

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

Systemic sequences which contribute 1E-8 to containment bypass

In any case, should not exceed 100 most significant sequences

In order to insure that the NUREG-1335[4] screening criteria were met, the top 100 systemic sequences were chosen to be the Level 1 sequences to be considered in the Level 2 analysis. Table 4.3-2 summarizes these sequences in order of decreasing frequency.

4.3.3.2 Sequence Binning and Selection of Representative Sequences

The binning process is based on the expected containment response and status of containment systems during the course of the accident. The set of 100 sequences that were determined during the screening process to be the sequences to be bound or analyzed were input to the CET and the CET end state for these sequences have been noted. The results of the binning process reveal 13 different CET end states which are summarized in Figure 4.3.1. Figure 4.3-1 also denotes the frequency of each CET end state along with the number of Level 1 sequences that end in each perspective CET end state. For the purposes of binning, the sequence bins are referred to as CET end states.

The CET end states designators shown in Figure 4.3.1 indicate the state of the primary system pressure at vessel failure and the availability of the containment safeguards and whether or not containment isolation is successful. The 5 letter end state designators are described below:

Designator 1: H - denotes vessel failure at high pressure
L - denotes vessel failure at low pressure

Designator 2: A - denotes availability of RHR recirculation after vessel failure
F - denotes failure of RHR recirculation after vessel failure

Designator 3: A - denotes availability of internal containment sprays
F - denotes failure of internal containment sprays

Designator 4: A - denotes availability of containment fan coil units
F - denotes failure of containment fan coil units

Designator 5: A - denotes successful containment isolation
F - denotes failure to isolate containment

Once the sequences in each CET end state are determined, a representative sequence from each end state must be selected for source term analysis. This representative sequence is termed the "analyzed" sequence. The analyzed sequence is selected based on the frequency of occurrence and sequence characteristics. The source term result for the analyzed sequence is then assigned to all other sequences in that end state. Those remaining sequences are then referred to as "bounded" sequences. The process of assigning the source term computed for the analyzed sequence to the bounded sequences is accounted for by summing over all the sequences within a CET end state to determine the cumulative frequency associated with the reported source term. The list of CET end states along with the analyzed sequence from each end state is summarized in Table 4.3-3.

4.3.3.3 Release Categories

Release categories are defined in Table 4.3-4 as a function of containment failure timing (early vs. late), containment failure mode (overpressurization, impairment, or bypass) and the fractional airborne release of fission products to the environment. Based on the source term results of the analyzed sequences, release categories are assigned to the CET end states based on the source term analysis performed on the analyzed sequences.

4.3.4 Source Term Analysis

This section describes the 13 sequences that were analyzed using MAAP. Selection of these sequences is based on section 4.3.3.2. Several assumptions are made for the MAAP calculations and are outlined here since they significantly affect the calculated source term results.

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

(2) Based on the results of the phenomenological evaluation summary on direct containment heating and the location of the seal table, the debris that is dispersed from the cavity due to a high pressure melt ejection is modeled as a dry debris pool in the upper compartment. At Kewaunee, the floor beneath the seal table is located on the 616′-0″ elevation which is approximately 34 feet above the floor of the lower compartment. The debris, if dispersed, is expected to form a dry bed. To properly simulate this, the debris is dispersed to the refueling pool floor. Realistically, the debris on the 626′-0″ elevation will always be dry and modeling the debris in the upper compartment is the best alternative solution.

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

(4) The release fraction for the volatile and non-volatile fission products that are recorded in Table 4.3-5 and in the following sequence descriptions are obtained by selecting the largest release fraction from each fission product group. For the volatile fission products, the release fraction is chosen based on the higher value between cesium iodide (CsI) and cesium hydroxide (CsOH). The non-volatile fission product release fraction is chosen based on the higher release fraction for barium oxide (BaO) and molybdenum dioxide (MoO_2).

Each analyzed sequence is described below.

Station Blackout - Sequence 5 (PDS SBO-18)

Sequence Description:

This accident scenario is initiated by a station blackout. The Level 2 mission time for this sequence and all the Level 2 sequences is 48 hours. The following Level 1 event tree nodes are modeled for the source term analysis:

AF2 Successful operation of the turbine driven AFW pump

LR1 Successful initiation of RHR injection and recirculation after 24 hours

FCH Successful initiation of containment fan coil units after 24 hours

ICS Successful recovery of the internal containment sprays after 24 hours

CI Successful containment isolation

A station blackout results in the loss of all containment safeguards until AC power is restored. For this sequence, the turbine driven auxiliary feedwater (AFW) pump is available for 8 hours. AC power is restored after 24 hours, along with all the containment safeguards systems.

Sequence Quantification:

The reactor scrams immediately upon the loss of AC power and the turbine driven auxiliary feedwater (TDAFW) pump starts to remove decay heat from the reactor core. AFW is assumed operational for 8 hours, the life of the safeguards batteries. Since cooling water is no longer provided to the reactor coolant pump (RXCP) seals, a 35 gpm/pump leak is assumed to occur ½ hour after the station blackout event occurs. This leak combined with the availability of AFW slightly depressurizes the reactor coolant system (RCS) to around 1300 psia. Once AFW is lost and the core uncovers, the RCS begins to heat up and pressurize to around 2500 psia. For this sequence core uncovery occurs roughly 15 hours after accident initiation with vessel failure occurring 3 hours later.

Since the RCS fails at high pressure, some of the corium is dispersed out of the cavity. Kewaunee's seal table is located one floor higher than the basemat of the lower compartment. Therefore, when the debris exits the seal table it does not disperse into a pool of water. Since MAAP can only disperse corium to the floor of the lower compartment, where a large water pool is present any time the RWST is injected, or to the refueling pool in the upper compartment which can be kept dry even when the RWST is injected, the corium displacement is modeled to the upper compartment.

After 24 hours, AC power is recovered and all the containment safeguards systems are available. The containment pressure is approximately 46 psia when the RHR pumps inject into the failed vessel to flood the cavity, and the containment fan coil units and internal containment spray (ICS) initiate to reduce containment pressure. With all these systems available, the containment is returned to safe stable state.

For this sequence 79 lbs of hydrogen is burned in the lower compartment just after vessel failure. The containment hydrogen inventory after the 48 hour mission time is approximately 540 pounds. A majority of this hydrogen is generated in-vessel due to 56% of the zirconium being oxidized prior to vessel failure. Since there is no concrete ablation in this case very little hydrogen is generated ex-vessel.

The release of fission products from the containment is limited to normal containment leakage since the containment does not fail for this accident scenario. Using CsI as an indicator for the volatile fission products and MoO_2 for the non-volatile fission products, the environmental release for this scenario is calculated to be:

	Airborne Release @48 hrs.
Noble gases (%)	0.32
Volatile fission products (%)	1.61E-03
Non-volatile fission products (%)	2.37E-06

Station Blackout - Sequence 72 (PDS SBO-5)

Sequence Description:

This station blackout is very similar to Sequence 5 in the sense that power is recovered after 24 hours. The following Level 1 event tree nodes for this sequence are modeled for source term analysis:

FCH Successful recovery of containment fan coil units after 24 hours

ICS Successful recovery of internal containment sprays after 24 hours

CI Successful containment isolation

Unlike Sequence 5, this sequence does not have AFW available and therefore, fails the vessel much earlier than the sequence in which AFW is available. AC power is recovered after 24 hours and the ICS and fan coolers are recovered, but RHR recirculation fails due to the failure of a check valve to re-open when the switch to recirculation is made. The valve failure allows the RHR pumps to supply water to the suction of the ICS pumps, but not to the reactor vessel.

Sequence Quantification:

For this station blackout, AFW is not available. Therefore, no form of decay heat removal exists and vessel failure comes much earlier. As a result of the station blackout, cooling water is no longer provided to the RXCP seals and a 35 gpm/pump leak is assumed to occur ½ hour after the station blackout event occurs. The RCS water inventory is lost primarily through the pressurizer safety valves lifting and the pump seal LOCA. Core uncovery occurs approximately 4.4 hours after the blackout with vessel failure occurring nearly 2 hours later.

The reactor vessel generally fails at high pressures in a station blackout, which results in the debris being dispersed into the upper compartment (see Sequence 1 for discussion of Kewaunee debris dispersal). The debris in the cavity and the refueling pool begin to heat up and pressurize containment to approximately 66 psia. At this point, 24 hours have elapsed, and the ICS and containment fan coil units are recovered and immediately begin to reduce the containment temperatures and pressures.

For this sequence no hydrogen burns occur. The containment hydrogen inventory after the 48 hour mission time is approximately 600 pounds. A majority of this hydrogen is generated in-vessel due to 54% of the zirconium being oxidized prior to vessel failure. Since there is no concrete ablation in this case very little hydrogen is generated ex-vessel.

The release of fission products from the containment is limited to normal containment leakage since the containment did not fail for this accident scenario. Using CsI as an indicator for the volatile fission products and MoO_2 for the non-volatile fission products, the environmental release for this scenario is calculated to be:

	Airborne Release @48 hrs.
Noble gases (%)	0.59
Volatile fission products (%)	5.03E-03
Non-volatile fission products (%)	2.54E-06

Loss of Offsite Power - Sequence 23 (PDS LSP-5)

Sequence Description:

This sequence is initiated by a loss of offsite power with the ability to establish emergency AC power to one of the 4.16 kV busses from the diesel generators. The following event tree nodes are modeled for the source term analysis:

OSP Onsite power established

FCH Successful operation of fan coil units for containment heat removal

CHG Successful operation of charging pumps for RXCP seal cooling

CI Successful containment isolation

This sequence assumes failure of AFW, low pressure recirculation, and ICS, and failure of the operators to initiate feed and bleed. Since at least one of the 4.16 kV buses is available the containment fan coil units are operational.

Sequence Quantification:

The sequence progression, in terms of how the RCS responds, is very similar to the sequence response of Sequence 72. Since there is no form of decay heat removal, the RCS pressurizes to the setpoint of the pressurizer PORVs, and eventually drains through the pressurizer PORVs. The successful operation of the charging pumps assures proper RXCP seal cooling, thus eliminating the concern of a pump seal LOCA. The core uncovers 4.4 hours after the loss of offsite power event, with vessel failure occurring roughly two hours later. Due to the lack of any vessel cooldown and depressurization, the vessel fails at high pressure.

Since the vessel fails at high pressure, the corium is distributed between the cavity and the upper compartment refueling pool in order to properly model debris dispersal at Kewaunee (see Sequence 1 for discussion of Kewaunee debris dispersal). Due to the lack of any containment or vessel injection systems, the RWST is not injected and therefore, very little steam is present in containment with the fan coil units operable. The containment pressure at the end of the 48 mission time is well below the ultimate failure pressure.

Due to the low steam concentration in containment, some hydrogen burns are identified in the lower compartment. Approximately 100 lbs of hydrogen is burned in containment with 475 lbs of hydrogen remaining in containment at the end of the Level 2 mission time. Most of the hydrogen is generated in-vessel due to the oxidation of 53% of the zirconium inventory. This oxidation resulted in approximately 580 lbs of hydrogen being generated in the primary system.

The release of fission products from the containment is limited to normal containment leakage since the containment does not fail for this accident scenario. Using CsI as an indicator for the volatile fission products and MoO_2 for the non-volatile fission products, the environmental release for this scenario is calculated to be:

	Airborne Release @48 hrs.
Noble gases (%)	0.57
Volatile fission products (%)	8.62E-03
Non-volatile fission products (%)	1.74E-05

Transient with Main Feedwater - Sequence 14 (PDS TRA-7)

Sequence Description:

This sequence is initiated by a transient with main feedwater available. The following event tree nodes for this sequence are modeled in the source term analysis:

OSP Offsite power is available

CI Containment isolation is successful

This sequence assumes failure of AFW, failure of operator actions to establish main feedwater or bleed and feed, and failure of fan coil units, ICS, and low pressure injection after vessel failure. Since the operators fail to initiate feed and bleed actions, the SI pumps do not inject the RWST.

Sequence Quantification:

With the failure of all feedwater systems as well as all the injection systems, the primary system starts to heat up and pressurize to the pressurizer PORV setpoint. With pressurizer PORV opening approximately 2 hours after the transient begins, core uncovery occurs roughly ½ hour after the PORV lifts. Soon after the core uncovers, the fuel starts heating up and vessel failure occurs approximately 75 minutes after core uncovery. Due to the lack of any form of vessel cooldown or depressurization, the RCS pressure is near the pressurizer PORV setpoint at the time of vessel failure.

Since the primary system fails at high pressure, the corium is dispersed out of the cavity and into the refueling pool in the upper compartment (see Sequence 5 for discussion of Kewaunee debris dispersal). Due to the lack of any containment heat removal or vessel injection system, the containment pressure at the end of the Level 2 mission time is only 10 psi from the ultimate containment failure pressure of 137 psia. Without any accident management recovery schemes, the containment fails around 50 hours. The analysis below shows source term results for this transient when the transient is allowed to run for 60 hours.

Due to a relatively low steam concentration in containment, some hydrogen burning occurs in the lower compartment. Approximately 104 lbs of hydrogen is burned in the lower compartment with roughly 685 lbs present in containment at the end of the Level 2 mission time. Most of the hydrogen is generated in-vessel due to the oxidation of the cladding.

The release of fission products from containment is limited to normal containment leakage since the containment does not fail within the 48 hour mission time. Since containment failure occurs only a couple hours after the end of the Level 2 mission time, the values for the source term release at 60 hours are listed in parentheses next to the 48 hour release. Using CsI for an indicator for the volatile fission products and BaO for the non-volatile fission products, the environmental release for this sequence is calculated to be:

	Airborne Release @48 hrs.
Noble gases (%)	0.64 (97.7)
Volatile fission products (%)	1.74×10^{-3} (4.16×10^{-1})
Non-volatile fission products (%)	7.68×10^{-6} (1.85×10^{-4})

Station Blackout - Sequence 4 (PDS SBO-30)

Sequence Description:

This sequence is a station blackout in which power recovery within the 48 hour mission time is not successful. The following Level 1 event tree nodes for this sequence were modeled for source term analysis:

AF2 Successful operation of the turbine driven auxiliary feedwater pump

CI Successful containment isolation

Since power is not restored within the 48 hour mission time, none of the containment safeguards are available for containment heat removal or debris coolability.

Sequence Quantification:

The reactor scrams immediately upon the loss of AC power and the TDAFW pump starts to remove decay heat from the reactor core. The TDAFW pump is assumed operational for 8 hours, the life of the safeguards batteries. Since cooling water can no longer be provided to the reactor coolant pump seals, a 35 gpm/pump leak is assumed to occur ½ hour after the station blackout event occurs. This leak, combined with the availability of AFW slightly depressurizes the primary system to around 1300 psia. Once AFW shuts off and the core uncovers, the RCS begins to heat up and pressurize back up to around 2500 psia. For this sequence, core uncovery occurs roughly 15 hours after accident initiation with vessel failure occurring 3 hours later.

Since the vessel fails at high pressure, the corium is distributed between the cavity and the upper compartment refueling pool in order to properly model debris dispersal at Kewaunee (see Sequence 1 for discussion of Kewaunee debris dispersal). Due to the lack of any containment or vessel injection systems, the RWST is not injected, therefore limiting the pressurization due to steaming. Since no form of containment heat removal is available, the containment pressures and temperatures are continuously increasing at the end of the 48 mission time. The maximum containment pressure is roughly 74 psia with temperature around 450°F. If no accident management actions are taken, the containment will eventually fail on overpressure.

Due to the low steam concentration in containment, some hydrogen burns occur in the lower compartment. Approximately 90 lbs of hydrogen is burned in containment with 480 lbs of hydrogen present in containment at the end of the Level 2 mission time. Most of the hydrogen is generated in-vessel due to the oxidation of 53% of the zirconium inventory. This oxidation results in approximately 570 lbs of hydrogen being generated in the RCS.

The release of fission products from the containment is limited to normal containment leakage since the containment does not fail in this accident scenario. Using CsI as an indicator for the volatile fission products and MoO_2 for the non-volatile fission products, the environmental release for this scenario is calculated to be:

	Airborne Release @48 hrs.
Noble gases (%)	0.4
Volatile fission products (%)	2.96E-03
Non-volatile fission products (%)	8.53E-06

Steam Generator Tube Rupture - Sequence 6 (PDS SGR-4)

Sequence Description:

This accident scenario is initiated by a failure of one of the steam generator tubes. The following event tree nodes for this sequence are modeled in the source term analysis:

HI1 Successful operation of the high pressure SI pumps

AF1 Successful operation of auxiliary feedwater to intact steam generator

ISO Main steam isolation valves close

This sequence is a direct bypass sequence due to a stuck open secondary safety valve on the broken steam generator. This stuck open safety valve is a direct result of the operator failure to cool down and depressurize the RCS prior to steam generator overfill.

Sequence Quantification:

A steam generator tube rupture is postulated to occur at the top of the U section of the tubes. This is conservative because it results in a minimum time to tube uncovery. Since the RCS pressure is much greater than the steam generator pressure, reactor coolant flows from the primary system to the secondary side of the broken steam generator. RCS pressure also decreases due to the expansion of the steam bubble in the pressurizer. Once the reactor trips, core power rapidly decreases to decay heat levels, steam flow to the turbine is terminated and main feedwater is isolated. Due to the decrease in RCS pressure, SI injection is initiated and continues until the RWST is drained, since recirculation is not possible due to the lack of water available in containment. SI injection is terminated approximately 6.75 hours after it begins.

Since the operators fail to cool down and depressurize the RCS, the leak of reactor coolant into the steam generator continues until the steam generator goes water solid and the safety valves automatically lift to decrease pressure in the secondary side. The containment bypass occurs due to the failure of the safety valve to reseat itself. This stuck open safety valve results in continuous flow of the secondary side water inventory out of containment resulting in a direct bypass of the containment. Eventually the core uncovers due to continuous flow of reactor coolant out of the RCS. The core uncovers roughly 17 hours after the tube rupture is initiated. Since the coolant inventory is depleted, the core begins to heat and fails the vessel 3 hours later.

Since the vessel fails at low pressure, all the corium remains in the cavity. With all the corium in the cavity and no water available to cool the debris, 3.9 feet of the concrete basemat is ablated. This ablation results in significant hydrogen generation throughout the course of the accident. Concrete ablation liberates water, which is retained in the concrete. This water reacts with the unreacted zirconium in the corium to produce hydrogen. At the end of the 48 hour mission time, 520 lbs of hydrogen were present in containment. Due to the operation of the fan coil units, steam is continuously being condensed in containment. Since steam is being condensed, the containment is not steam inerted and 818 lbs of hydrogen is burned in containment.

The containment fan coil units keep the containment pressure well below the ultimate failure pressure. The maximum containment pressure is approximately 22 psia at the end of the 48 hour mission time. If an external water source is not made available to the RHR pumps, the corium will eventually ablate through the basemat and fail containment due to extensive concrete attack.

Since containment is bypassed, the source term calculated is based on the fission products that escape through the stuck open safety valve. Using CsI as an indicator for the volatile fission products and MoO_2 for the non-volatile fission products, the following environmental release is calculated for this sequence:

	Airborne Release @48 hrs.
Noble gases (%)	98.6
Volatile fission products (%)	16.4
Non-volatile fission products (%)	1.07E-01

Large LOCA - Sequence 75 (PDS LLO-4)

Sequence Description:

This sequence is initiated by a 12 inch break in the intermediate leg at the junction of the RXCP and the intermediate leg piping. The following Level 1 event tree nodes for this sequence are modeled for source term analysis:

ACC Successful injection of the accumulators

LI1 Successful operation of RHR injection

FCH Successful operation of containment fan coil units

ICS Successful operation of internal containment sprays

CI Successful containment isolation

Although RHR injection is successful, the RHR pumps are incapable of providing recirculation to the reactor vessel due to the failure of a check valve to reopen when the pumps are restarted in recirculation mode. ICS and containment fan coil units are available for containment heat removal.

Sequence Quantification:

Due to the large break in the RCS, a rapid blowdown of the RCS into the lower compartment causes the lower compartment pressure to increase to the initiation setpoint for ICS. Shortly after the RCS blowdown, the RCS pressure drops below the RHR shutoff head and RHR injection is initiated. Approximately 40 minutes after injection was initiated, the RWST water level reaches the lo-lo setpoint. The switch to RHR recirculation is successful, but due to the failure of a check valve when the switch to recirculation was made, the RHR pumps are no longer capable of injecting into the vessel. ICS recirculation is available since the check valve is downstream of the point where ICS takes suction from the discharge of the RHR heat exchanger.

Due to the lack of core cooling, the core uncovers approximately 65 minutes after RHR injection is terminated. This leads to core heatup and vessel failure at 3.0 hours after accident initiation. Since the vessel fails at low pressure, all the corium is retained in the cavity. Due to the failure of RHR recirculation, the only way to get water in the cavity to cool the debris is by continuously operating ICS which communicates with the cavity via the bypass area around the reactor vessel. Since ICS remains on for the duration of the accident, the water entering the cavity is capable of preventing concrete ablation. At the end of the 48 hour mission time the containment pressure is well below the ultimate containment pressure and, due to the debris being quenched and containment heat removal available, the containment ends up in a safe stable state.

No hydrogen is burned in containment due to the lack of in-vessel hydrogen generation. Only 38% of the clad is oxidized in-vessel and no hydrogen is generated ex-vessel due to no concrete ablation occurring in the cavity. At the end of the 48 hour Level 2 mission time, the 420 lbs of hydrogen that was generated in-vessel is distributed among the containment compartments.

The release of fission products from the containment is limited to normal containment leakage since the containment does not fail for this accident scenario. Using CsOH as an indicator for the volatile fission products and MoO_2 for the non-volatile fission products, the environmental release for this scenario is calculated to be:

	Airborne Release @48 hrs.
Noble gases (%)	0.51
Volatile fission products (%)	2.03E-04
Non-volatile fission products (%)	1.53E-06

Small LOCA - Sequence 2 (PDS SLO-19)

Sequence Description:

This accident scenario is a 1 inch break in the intermediate leg at the junction of the RXCP and the intermediate leg piping. The following Level 1 event tree nodes are successful and are modeled in the source term analysis:

HI2 Successful operation of high pressure SI injection

AF0 Successful operation of auxiliary feedwater

FCH Successful operation of containment fan coil units

CI Successful containment isolation

This sequence has successful operation of the SI pumps, but failure of high pressure recirculation due to the failure of the RHR pump. This failure also eliminates the possibility of the ICS recirculation. Operation of the fan coil units does allow for containment heat removal, but failure of the RHR pumps indicate that the debris is not quenched in the cavity.

Sequence Quantification:

This small LOCA is defined as a 1 inch break in the RCS with a failure of the operators to cool down and depressurize the primary system. Since the RCS is not depressurized, the only way the RWST is injected into the vessel is via the SI pumps. The SI pumps start injecting approximately 3 minutes after the LOCA begins. This injection continues for 4.75 hours until the RWST reaches the lo-lo setpoint, injection is stopped, and recirculation fails. The RCS pressure at the time of vessel failure is approximately 700 psia which is below the direct containment heating cutoff pressure (800 psia). The sequence is therefore considered a low pressure sequence.

Since the RCS fails at a low pressure, all the corium remained in the cavity. With all the corium in the cavity and no water available to cool the debris, 4.8 feet of the concrete basemat is ablated. This ablation results in significant hydrogen generation throughout the course of the accident. Concrete ablation liberates water that is retained in the concrete. This water reacts with the unreacted zirconium in the corium to produce hydrogen. At the end of the 48 hour mission time, 1255 lbs of hydrogen is present in containment. Due to the operation of the fan coil units, steam is continuously being condensed in containment. Since steam is being condensed, the containment is not steam inerted and 1003 lbs of hydrogen is burned in containment.

The containment fan coil units keep the containment pressure well below the ultimate failure pressure. The maximum containment pressure is approximately 46 psia at the end of the 48 hour mission time. If the RHR pumps are not recovered, the corium eventually ablates through the basemat and fails containment due to extensive concrete attack.

Since the containment does not fail, the release of fission products from the containment is limited to normal containment leakage. Using CsOH as the indicator for volatile fission products and BaO for the non-volatile fission products, the environmental release for this accident scenario is calculated to be:

	Airborne Release @48 hrs.
Noble gases (%)	0.45
Volatile fission products (%)	4.61E-03
Non-volatile fission products (%)	1.33E-04

Medium LOCA - Sequence 61 (PDS MLO-6)

Sequence Description:

This sequence is caused by a 2 inch break in the intermediate leg at the junction between the intermediate leg and the RXCP. The following Level 1 event tree nodes for this sequence are modeled in the source term analysis:

LI2 Successful operation of low pressure RHR injection

AF0 Successful operation of auxiliary feedwater

OP1 Successful cooldown and depressurization of the RCS

FCH Successful operation of containment fan coil units

CI Successful containment isolation

This medium LOCA is defined as a 2 inch LOCA in which the SI pumps fail, but injection is achieved by successful cooldown and depressurization of the RCS using the pressurizer PORVs. RHR injection is successful, but the switch to recirculation is not successful. Therefore, RHR and ICS recirculation are not operable once the RWST reaches the lo-lo setpoint. Consequently, the fan coil units are the only form of containment heat removal available.

Sequence Quantification:

The accident scenario is initiated by a 2 inch break in the RCS. Since the SI pumps are not available, the operators successfully cooldown and depressurize the primary system to below the RHR shutoff head. RCS cooldown and depressurization is achieved through the use of the pressurizer PORVs. This allows the RHR pumps to inject the RWST into the primary system to provide temporary core cooling. The RWST Level reaches the lo-lo setpoint approximately 5.3 hours after accident initiation. Since RHR recirculation is not successful, injection ceases at this time. Once injection is terminated, the core uncovers roughly 3.8 hours later with vessel failure occurring 2 hours after core uncovery.

Due to RCS cooldown and depressurization, the primary system fails at a low pressure. Since the vessel fails at low pressure, all the corium remains in the cavity. Without the availability of RHR recirculation, the cavity remains dry throughout the course of the accident and concrete ablation does occur. Approximately 5 feet of the cavity basemat is eroded at the end of the 48 hour mission time. As mentioned earlier, concrete ablation liberates water that is retained in the concrete. This water reacts with the unreacted zirconium in the corium to produce hydrogen. At the end of the 48 hour mission time, 1300 lbs of hydrogen is present in containment. Due to the operation of the fan coil units, steam is continuously being condensed in containment. Since steam is being condensed, the containment is not steam inerted and 983 lbs of hydrogen are burned in containment.

The containment fan coil units keep the containment pressure well below the ultimate failure pressure. The maximum containment pressure is approximately 43 psia at the end of the 48 hour mission time. If the RHR pumps are not recovered, the corium will eventually ablate through the basemat and fail containment due to extensive concrete attack.

Since the containment does not fail, the release of fission products from the containment is limited to normal containment leakage. Using CsOH as the indicator for volatile fission products and BaO for the non-volatile fission products, the environmental release for this accident scenario is calculated to be:

	Airborne Release @48 hrs.
Noble gases (%)	0.46
Volatile fission products (%)	4.2E-03
Non-volatile fission products (%)	1.65E-04

Large LOCA - Sequence 48 (PDS LLO-6)

Sequence Description:

This large LOCA is initiated by a 12 inch break in the intermediate leg at the junction of the intermediate leg and the RXCP. The following Level 1 event tree nodes for this sequence are modeled in the source term analysis:

ACC Successful operation of accumulators

LI1 Successful operation of low pressure RHR injection

CI Successful containment isolation

Operators are successful with injection the RWST, but fail to go to RHR recirculation. Due to failure of RHR recirculation, ICS also fails to go to recirculation. The sequence definition also states that the containment fan coil units fail to actuate.

Sequence Quantification:

Due to the large break in the RCS, a rapid blowdown of the RCS into the lower compartment causes the lower compartment pressure to increase to the initiation setpoint for the ICS. Shortly after the RCS blowdown, the RCS pressure drops below the RHR shutoff head and RHR injection is initiated. Approximately 40 minutes after injection is initiated, the RWST level reaches the lo-lo setpoint. The switch to RHR recirculation is not successful, so RHR and ICS injection are terminated.

Due to the lack of core cooling, the core uncovers approximately 75 minutes after RHR injection is terminated. This leads to core heatup and vessel failure roughly 3.4 hours after accident initiation. Since the vessel fails at low pressure, all the corium is retained in the cavity. Due to the failure of RHR recirculation, there is no way to get water into the cavity to cool the debris. This leads to approximately 5.2 feet of the cavity basemat being ablated. With all this concrete ablation occurring throughout the accident, the containment hydrogen inventory is continuously increasing. At the end of the 48 hour mission time, 2200 lbs of hydrogen is present in containment, with only 102 lbs being burned in containment.

Not much hydrogen is burned in containment due to the large steam concentration throughout containment. This high steam concentration is the direct result of the lack of any form of containment heat removal (i.e., fan coil units or ICS). Since no heat is being removed from containment, the containment pressure is continuously increasing. At the end of the 48 hour mission time, the maximum containment pressure is approximately 83 psia with an average gas temperature of 345°F. If no accident management actions are taken, containment failure due to overpressurization will actually occur.

The release of fission products from the containment is limited to normal containment leakage since the containment did not fail for this accident scenario. Using CsOH as an indicator for the volatile fission products and BaO for the non-volatile fission products, the environmental release for this scenario is calculated to be:

	Airborne Release @48 hrs.
Noble gases (%)	0.61
Volatile fission products (%)	1.34E-02
Non-volatile fission products (%)	1.27E-04

Station Blackout - Sequence 81 (PDS SBO-2)

Sequence Description:

Sequence 81 is defined as a station blackout with power recovery between 2 and 24 hours, but with containment failing to isolate. The Level 1 event tree nodes and primary system response are identical to that of Sequence 72 in which the TDAFW pump fails to operate.

Sequence Quantification:

The differences occur after vessel failure due to the failure to isolate containment. For purposes of modeling the containment isolation failure size, the Level 1 portion of the Kewaunee IPE does not assign a failure probability to each penetration, therefore source terms are calculated for several different penetration sizes. For conservatism, the 36 inch containment purge line is assumed to be the representative isolation failure for the failure to isolate end states. Sensitivity studies were done on other isolation failure sizes and are recorded in Table 4.4-4.

With a 36 inch isolation failure in the upper compartment, the source term calculated for this sequence was fairly large in relation to all the other dominant sequences which were limited to containment leakage. Using CsOH as an indicator for the volatile fission products and MoO_2 for the non-volatile fission products, the following environmental release was calculated for this sequence:

	Airborne Release @48 hrs.
Noble gases (%)	91.5
Volatile fission products (%)	4.1
Non-volatile fission products (%)	5.92×10^{-3}

Small LOCA - Sequence 84 (PDS SLO-10)

Sequence Description:

This small LOCA sequence assumes a 1 inch break in the intermediate leg at the junction of the RXCP and the intermediate leg. The following Level 1 event tree nodes for this sequence are modeled in the source term analysis:

HI2 Successful operation of SI injection

AFO Successful operation of auxiliary feedwater

FCH Successful operation of containment fan coil units

This sequence assumes failure of the operators to cool down and depressurize the RCS and establish low pressure recirculation, which in turn negates high pressure SI and ICS recirculation. The operators also fail to isolate containment for this sequence.

Sequence Quantification:

For a small LOCA (1 inch) with AFW and high pressure injection, the primary system pressure steadily decreases to approximately 1000 psi until SI injection terminates, 6.5 hours after the accident is initiated. Once SI is terminated, the primary system pressure drops just below the RHR shutoff head. Once the break is uncovered, the water in the primary system slowly boils off until the core uncovers 8 hours after SI is discontinued. The primary system starts to pressurize as the water begins to boil off, but the pressure at the time of vessel failure is slightly lower than the pressure necessary to disperse corium out of the cavity. Vessel failure occurs approximately 2 hours later.

Since the vessel fails at low pressure all the corium is retained in the cavity. With the absence of low pressure recirculation, the debris is not quenched and approximately 4.7 feet of the concrete basemat is ablated at the end of the 48 hour mission time. Since the containment is not isolated, containment failure, in a sense, has already occurred, but failure of the concrete basemat will occur if debris coolability is not restored. Containment fan coil units keep the containment temperature low, along with the containment pressure (which is never high due to the isolation failure).

For this sequence, significant hydrogen is generated due to the extensive concrete ablation that occurs in the cavity. With the operation of the fan coil units, the steam concentration is fairly low and as a result of this, approximately 770 lbs of hydrogen is burned in containment. The hydrogen that is not burned, escapes containment via the isolation failure. Roughly 500 lbs of hydrogen is generated in-vessel and 1700 lbs is generated due to concrete ablation.

With a 36 inch isolation failure in the upper compartment, the source term calculated for this sequence is fairly large in relation to all the other dominant sequences, which are limited to containment leakage. Using CsI as an indicator for the volatile fission products and BaO for the non-volatile fission products, the following environmental release is calculated for this sequence:

	Airborne Release @48 hrs.
Noble gases (%)	98.6
Volatile fission products (%)	7.7
Non-volatile fission products (%)	1.27E-01

Station Blackout - Sequence 97 (PDS SBO-31)

Sequence Description:

This sequence is identical to the station blackout in Sequence 4, but with containment failing to isolate. Therefore, the Level I event tree nodes that are modeled and the RCS behavior up to vessel failure are identical.

Sequence Quantification:

The differences occur after vessel failure due to the failure to isolate the containment. For purposes of modeling the containment isolation failure size, the Level 1 portion of the Kewaunee IPE does not assign a failure probability to each penetration. Therefore, source terms are calculated for several different penetration sizes. For conservatism, the 36 inch containment purge and vent line is assumed to be the representative isolation failure for the failure to isolate end states. Sensitivity studies are done on other isolation failure sizes and are recorded in Table 4.4-4.

Because the vessel fails at high pressure, the corium is dispersed between the cavity and the refueling pool. Since the containment was not isolated the pressure never exceeds atmospheric pressure, except for a pressure spike that occurs directly after vessel failure. The temperature in containment is approximately 469°F at the end of the 48 hour Level 2 mission time. The containment hydrogen inventory at 48 hours is very low, since a majority of the hydrogen leaves containment via the isolation failure. Approximately 600 lbs of hydrogen is generated in-vessel due to the oxidation of the zirconium cladding.

With a 36″ isolation failure in the upper compartment, the source term calculated for this sequence is fairly large in relation to all the other dominant sequences, which are limited to containment leakage. Using CsI as an indicator for the volatile fission products and MoO_2 for the non-volatile fission products, the following environmental release is calculated for this sequence:

	Airborne Release @48 hrs.
Noble gases (%)	97.9
Volatile fission products (%)	7.6
Non-volatile fission products (%)	1.68E-02

4.3.5 Source Term Results

Several tables are compiled to summarize the source term results. Table 4.3-5 presents selected accident progression parameters for each analyzed sequence. This contains information such as accident timing and conditions, hydrogen burn data, and radiological release. Note that there are no cases in which containment overpressure failure occurs within 48 hours. Also, only sequences in which the containment is bypassed or impaired have volatile releases greater than 0.01%. For these sequences, the fission product releases after 48 hours are shown in Table 4.3-5.

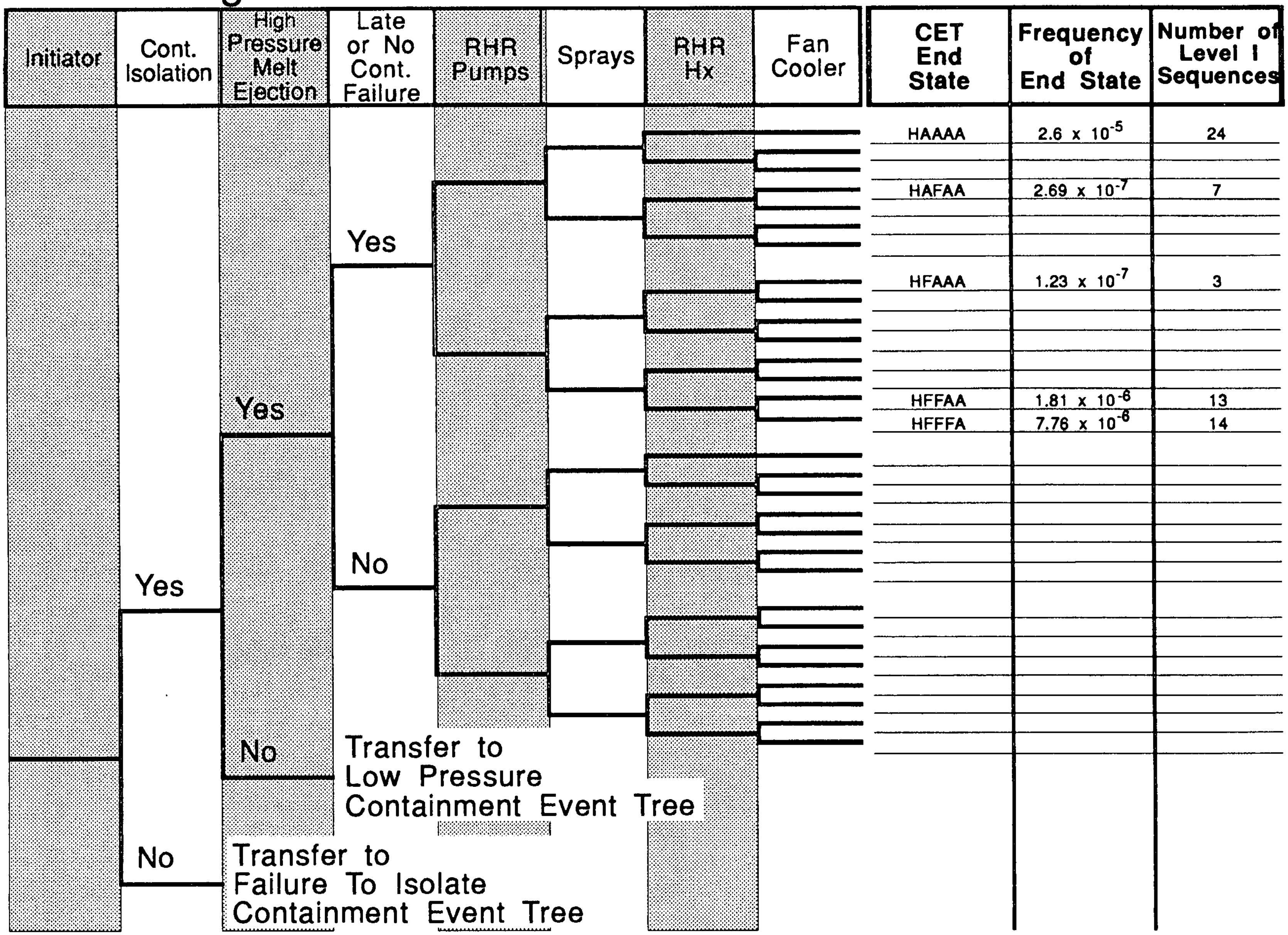
Table 4.3-6 summarizes the source term results by release category, and shows the conditional probability of each release category given core damage. These results show that, should a core damage event occur at Kewaunee, there is a 92% probability that the radionuclide release would represent less than 0.1% of the volatile fission products. Successful recovery to a safe stable state, in which the core debris is cooled and decay heat is being removed from containment, is expected for 43% of the core damage sequences. A significant amount (49%) of the core damage sequences would require some additional recovery action not credited in the IPE in order to prevent eventual containment failure, although in any case containment failure does not occur within 48 hours.

Lastly, Table 4.3-7 summarizes the containment failure modes identified during the source term analysis. These results show that, for the 49% of core damage sequences that would require additional recovery actions not credited in the IPE, 36% of these sequences are concrete basemat ablation failures after 48 hours.

FIGURE 4.3-1a

CET END STATES FOR SOURCE TERM ANALYSIS

Wisconsin Public Service --- Kewaunee
High Pressure -- Containment Event Tree

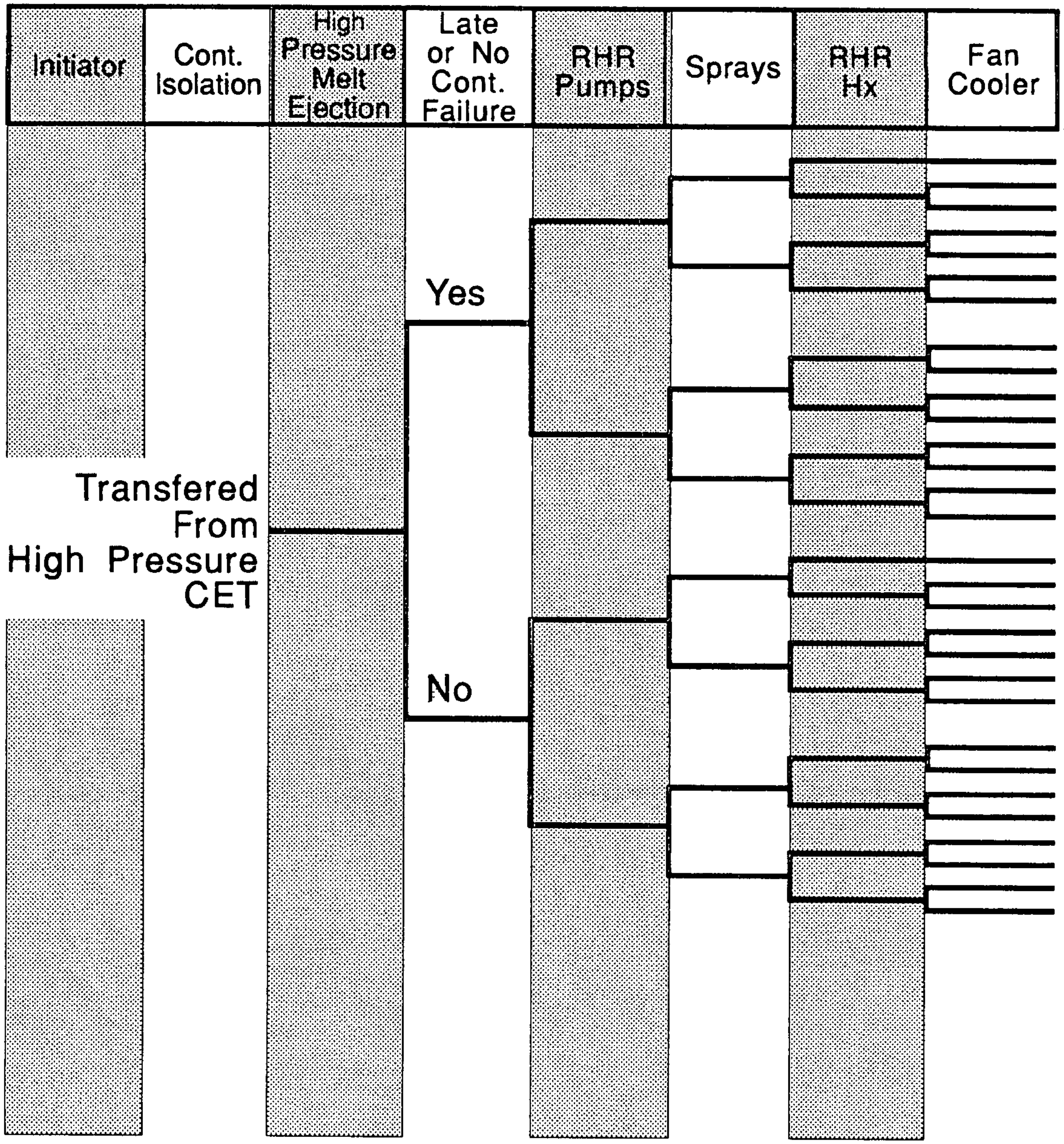


RR92J041.CDR

FIGURE 4.3-1b

CET END STATES FOR SOURCE TERM ANALYSIS

Wisconsin Public Service --- Kewaunee
Low Pressure -- Containment Event Tree



CET End State	Frequency of End State	Number of Level I Sequences
LAAAA	5.3×10^{-7}	4
LAFAA	4.27×10^{-8}	2
LFAAA	8.82×10^{-9}	3
LFFAA	2.39×10^{-5}	11
LFFFA	5.13×10^{-7}	7

RR92J043.CDR

FIGURE 4.3-1c

CET END STATES FOR SOURCE TERM ANALYSIS

Wisconsin Public Service --- Kewaunee
Failure To Isolate -- Containment Event Tree

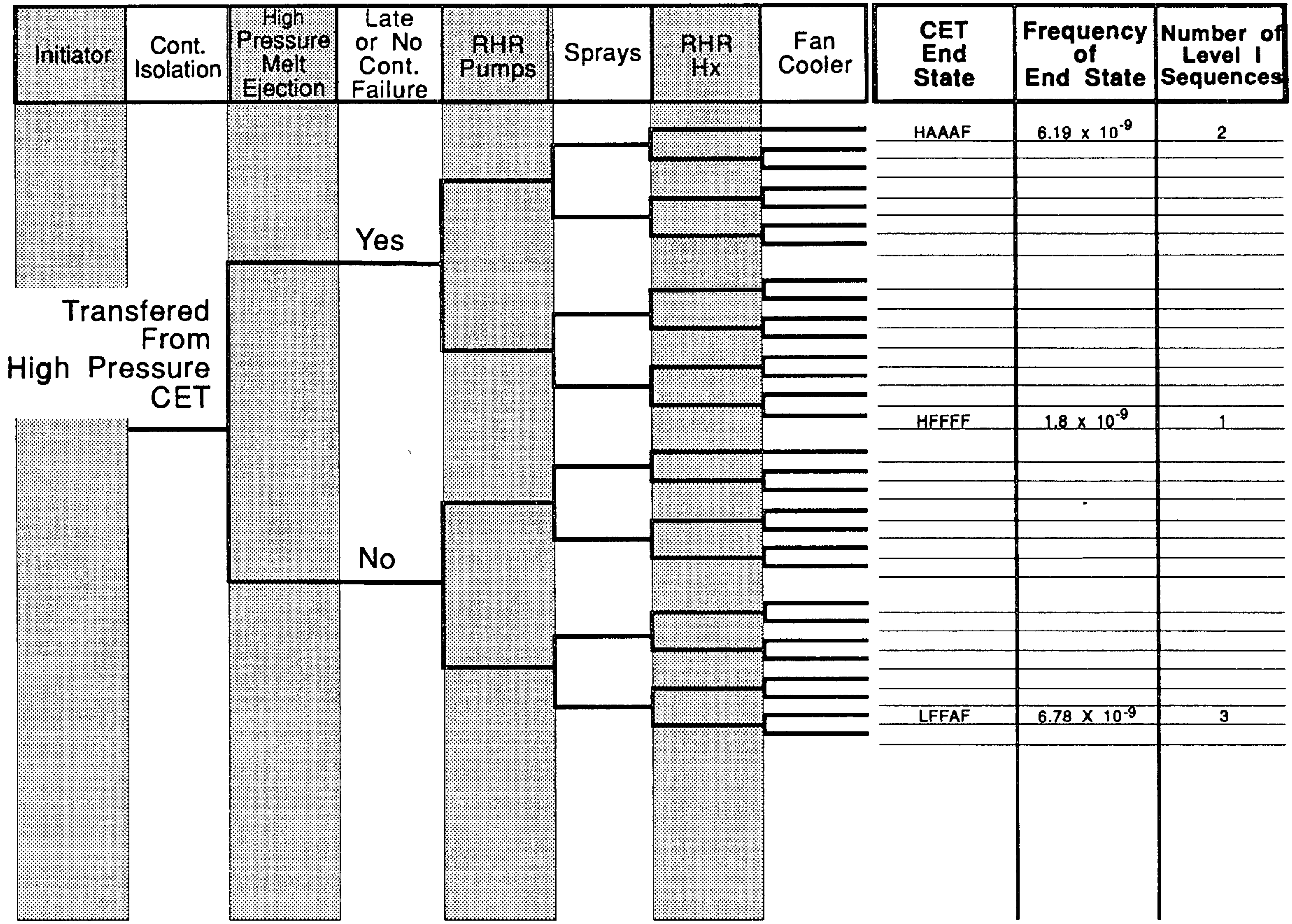


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TABLE 4.3-1

INITIAL INVENTORY OF FISSION PRODUCT GROUPS

Fission Product Group	Initial Inventory (lb)
1) Noble gases (Xe, Kr)	641
2) CsI (volatile)	55
3) TeO_2	0
4) SrO	138
5) MoO_2	511
6) CsOH (volatile)	410
7) BaO	185
8) La_2O_3 (Pr_2O_3 + Nd_2O_3, Sm_2O_3 + Y_2O_3)	1001
9) CeO_2	416
10) Sb	4
11) Te_2 (volatile)	54
12) UO_2 (NpO_2 + PuO_2)	120,370

TABLE 4.3-2

KEWAUNEE NUCLEAR PLANT TOP 100 DOMINANT SEQUENCES

Sequence No.	Initiator From Table 3.1.5-1	CET End State	Frequency
1	SBO-1	HAAAA	1.25×10^{-5}
2	SLO-9	LFFAA	1.21×10^{-5}
3	MLO-9	LFFAA	7.42×10^{-6}
4	SBO-30	HFFFA	4.35×10^{-6}
5	SBO-18	HAAAA	4.35×10^{-6}
6	SGR-4	BYPASS	4.31×10^{-6}
7	SBO-25	HAAAA	4.01×10^{-6}
8	INA-2	HFFFA	2.08×10^{-6}
9	TRA-1	HAAAA	2.02×10^{-6}
10	LSP-1	HAAAA	1.83×10^{-6}
11	LL0-5	LFFAA	1.35×10^{-6}
12	LSP-10	HFFAA	1.23×10^{-6}
13	SLO-3	LFFAA	1.17×10^{-6}
14	TRA-7	HFFFA	6.74×10^{-7}
15	LSP-9	LFFAA	6.55×10^{-7}
16	SGR-1	BYPASS	6.02×10^{-7}
17	MLO-3	LFFAA	5.35×10^{-7}
18	LLO-2	LFFAA	4.57×10^{-7}
19	SWS-1	HFFFA	4.21×10^{-7}
20	LSP-7	LAAAA	3.49×10^{-7}
21	SLO-7	LFFFA	3.18×10^{-7}
22	VEF-1	HAAAA	3.0×10^{-7}
23	LSP-1	HFFAA	2.67×10^{-7}
24	TRS-1	HAAAA	2.2×10^{-7}
25	SGR-3	BYPASS	1.93×10^{-7}
26	SBO-8	HAAAA	1.79×10^{-7}
27	SBO-15	HFFAA	1.6×10^{-7}
28	SBO-9	HAAAA	1.53×10^{-7}

TABLE 4.3-2

KEWAUNEE NUCLEAR PLANT TOP 100 DOMINANT SEQUENCES (Continued)

Sequence No.	Initiator From Table 3.1.5-I	CET End State	Frequency
29	MLO-7	LFFFA	$1.47x10^{-7}$
30	SGR-2	BYPASS	$1.19x10^{-7}$
31	TDC-1	HAAAA	$1.18x10^{-7}$
32	TRS-8	LFFAA	$1.14x10^{-7}$
33	SBO-13	HAAAA	$1.01x10^{-7}$
34	LLO-1	LAAAA	$1.0x10^{-7}$
35	SBO-3	HAFAA	$9.33x10^{-8}$
36	TDC-5	HFFFA	$9.28x10^{-8}$
37	LSP-2	HAFAA	$8.66x10^{-8}$
38	TRS-6	LAAAA	$7.62x10^{-8}$
39	LSP-6	HFFFA	$6.54x10^{-8}$
40	SGR-5	BYPASS	$5.96x10^{-8}$
41	SBO-16	HAAAA	$4.91x10^{-8}$
42	SLO-6	LFFAA	$4.4x10^{-8}$
43	AWS-3	HAAAA	$4.4x10^{-8}$
44	SBO-6	HFFAA	$3.44x10^{-8}$
45	SBO-20	HAFAA	$3.39x10^{-8}$
46	LSP-8	LAFAA	$3.22x10^{-8}$
47	TRA-11	HFFAA	$3.21x10^{-8}$
48	LLO-6	LFFAA	$3.16x10^{-8}$
49	TRS-5	HFFFA	$3.06x10^{-8}$
50	SLB-1	HAAAA	$2.85x10^{-8}$
51	SLB-7	HAAAA	$2.71x10^{-8}$
52	SLB-6	HFFFA	$2.39x10^{-8}$
53	SBO-27	HAFAA	$2.38x10^{-8}$
54	SLB-3	HFFAA	$1.78x10^{-8}$
55	SBO-14	HAFAA	$1.56x10^{-8}$
56	CCS-2	HFFAA	$1.56x10^{-8}$

TABLE 4.3-2

KEWAUNEE NUCLEAR PLANT TOP 100 DOMINANT SEQUENCES (Continued)

Sequence No.	Initiator From Table 3.1.5-1	CET End State	Frequency
57	SBO-17	HAAAA	1.51×10^{-8}
58	CCS-1	HFFAA	1.27×10^{-8}
59	SBO-23	HFFAA	1.26×10^{-8}
60	AWS-2	HAAAA	1.24×10^{-8}
61	ML0-6	LFFAA	1.22×10^{-8}
62	SLO-1	HAAAA	1.2×10^{-8}
63	MLO-1	HAAAA	1.13×10^{-8}
64	TRA-3	HAFAA	1.12×10^{-8}
65	TRS-7	LAFAA	1.05×10^{-8}
66	SBO-11	HFFAA	1.02×10^{-8}
67	TRA-9	LFFAA	9.23×10^{-9}
68	SBO-4	HAAAA	8.01×10^{-9}
69	ISL-1	BYPASS	7.4×10^{-9}
70	AWS-1	HAAAA	7.1×10^{-9}
71	TRS-4	HFFAA	6.86×10^{-9}
72	SBO-5	HFAAA	5.99×10^{-9}
73	SLO-5	LFFFA	5.85×10^{-9}
74	SLO-11	LFFFA	5.85×10^{-9}
75	LLO-4	LFAAA	4.9×10^{-9}
76	SLO-2	HFFFA	4.72×10^{-9}
77	TRA-6	HFFAA	4.59×10^{-9}
78	TDC-2	HAFAA	4.53×10^{-9}
79	TRA-8	LAAAA	4.4×10^{-9}
80	LSP-11	HFFFA	4.39×10^{-9}
81	SBO-2	HAAAF	4.39×10^{-9}
82	LSP-4	HFAAA	4.27×10^{-9}
83	LSP-3	HAAAA	3.87×10^{-9}
84	SLO-10	LFFAF	3.52×10^{-9}

TABLE 4.3-2

KEWAUNEE NUCLEAR PLANT TOP 100 DOMINANT SEQUENCES (Continued)

Sequence No.	Initiator From Table 3.1.5-1	CET End State	Frequency
85	MLO-11	LFFFA	$2.97x10^{-9}$
86	SBO-12	HFFFA	$2.94x10^{-9}$
87	SBO-21	HAAAA	$2.86x10^{-9}$
88	MLO-5	LFFFA	$2.63x10^{-9}$
89	TRA-12	HFFFA	$2.35x20^{-9}$
90	SLO-8	LFAAA	$2.09x10^{-9}$
91	SBO-22	HFAAA	$2.08x10^{-9}$
92	TDC-4	HFFAA	$2.01x10^{-9}$
93	ML0-2	HFFFA	$1.99x10^{-9}$
94	SBO-28	HAAAA	$1.9x10^{-9}$
95	ML0-8	LFAAA	$1.83x10^{-9}$
96	SBO-19	HAAAF	$1.8x10^{-9}$
97	SBO-31	HFFFF	$1.8x10^{-9}$
98	SLB-4	HFFFA	$1.76x10^{-9}$
99	SLO-4	LFFAF	$1.69x10^{-9}$
100	ML0-10	LFFAF	$1.57x10^{-9}$

TABLE 4.3-3

KEWAUNEE NUCLEAR PLANT ANALYZED SEQUENCE TABLE

CET End State	End State Frequency	Analyzed Sequence	Release Category
HAAAA	2.6×10^{-5}	SBO-18	S
HAFAA	2.69×10^{-7}	Bounded by SBO-18	S
HFAAA	1.23×10^{-7}	SBO-5	S
HFFAA	1.81×10^{-6}	LSP-5	S
HFFFA	7.76×10^{-6}	SBO-30 & TRA-7	A
LAAAA	5.3×10^{-7}	Bounded by SBO-18	S
LAFAA	4.27×10^{-8}	Bounded by SBO-18	S
LFAAA	8.82×10^{-9}	LLO-4	S
LFFAA	2.39×10^{-5}	SLO-9 & MLO-6	A
LFFFA	5.13×10^{-7}	LLO-6	A
HAAAF	6.19×10^{-9}	SBO-2	G
HFFFF	1.8×10^{-9}	SBO-31	G
LFFAF	6.78×10^{-9}	SLO-10	G
BYPASS	5.28×10^{-6}	SGR-4	T

TABLE 4.3-4

RELEASE CATEGORY DEFINITION

Release Category	Definition
A	No containment failure within 48 hour mission time but failure could eventually occur without accident management action; noble gases and less than 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 4.3-4

RELEASE CATEGORY DEFINITION (Continued)

Release Category	Definition
L	Late containment failure with noble gases and up to 1% of the volatiles released (containment failure greater than six hours after vessel failure; containment not bypassed; isolation successful).
M	Late containment failure with noble gases and up to 10% of the volatiles released (containment failure greater than six hours after vessel failure; containment not bypassed; isolation successful).
N	Late containment failure with noble gases and up to 1% of the volatiles and up to 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 gases and more than 10% of the volatiles released.
U	Containment failure prior to vessel failure with the noble gases and more than 10% of the volatile fission products released (containment isolation impaired).
V	Early containment failure with noble gases and more than 10% of the volatiles released (containment failure within 6 hours of vessel failure; containment not bypassed; isolation successful).
W	Late containment failure with noble gases and more than 10% of the volatiles released (containment failure greater than 6 hours after vessel failure; containment not bypassed; isolation successful).

KEWAUNEE NUCLEAR PLANT
SOURCE TERM ANALYSIS RESULTS
MAAP RUN SUMMARY TABLE

SEQUENCE TYPE	Blackout	Blackout	Loss of Power	Blackout	Transient	SGTR	LLOCA	SLOCA
Sequence No.	5	72	23	4	14	6	75	2
Sequence Frequency	$4.35x10^{-6}$	$5.99x10^{-9}$	$2.67x10^{-7}$	$4.35x10^{-6}$	$6.74x10^{-7}$	$4.31x10^{-6}$	$4.9x10^{-9}$	$1.21x10^{-5}$
Sequence Designator	SBO-18	SBO-5	LSP-5	SBO-30	TRA-7	SGR-4	LLO-4	SLO-9
CORE/CONTAINMENT RESPONSE								
Time of Core Uncovery (hr)	14.9	4.4	4.4	14.9	2.5	17.0	1.7	10.6
Onset of Core Melt (hr)	15.9	5.0	5.0	15.9	3.0	18.8	2.1	11.2
Time of Vessel Failure (hr)	18.2	6.2	6.2	18.2	3.8	19.9	3.0	12.7
Time of Containment Failure (hr)	> 48.	> 48.	> 48.	> 48.	> 48.	Bypass	> 48.	> 48.
Maximum Containment Pressure (psia)	47.7	66.7	45.3	73.9	131.4	22.3	38.1	45.7
Maximum Containment Temperature (°F) (Upper/Annular Compartments)	385.	442.	403.	443.	467.	189.	276.	302.
Cavity Water Level @ 48 hrs (ft)	Flooded	0.	0.	0.	0.	0.		0.
Fraction of Clad Reacted in Vessel	0.562	0.545	0.534	0.526	0.532	0.435	0.379	0.46
H2 Mass Burned (lbm)	79.	0.	95.	89.	104.	818.	0.	1003.
Cavity Concrete Ablation Depth 48 hrs (ft)	0.	0.	0.	0.	0.	3.9	0.	4.8
FISSION PRODUCT DISTRIBUTION AT END OF MISSION TIME								
Noble Release (%)	0.32	0.59	0.57	0.4	0.64	98.6	0.51	0.45
Volatile FP Release (%)	$1.61x10^{-3}$	$5.03x10^{-3}$	$8.62x10^{-3}$	$2.96x10^{-3}$	$1.74x10^{-3}$	16.4	$2.03x10^{-4}$	$4.61x10^{-3}$
Non-Volatile FP Release (%)	$2.37x10^{-6}$	$2.54x10^{-6}$	$1.74x10^{-5}$	$8.53x10^{-6}$	$7.68x10^{-6}$	$1.07x10^{-1}$	$1.53x10^{-6}$	$1.33x10^{-4}$
Volatile FP Retained in Primary System %	92.8	82.3	79.3	88.9	91.3	53.2	52.7	87.1

KEWAUNEE NUCLEAR PLANT
SOURCE TERM ANALYSIS RESULTS
MAAP RUN SUMMARY TABLE (Continued)

SEQUENCE TYPE	LLOCA	MLOCA	Blackout	SLOCA	Blackout
Sequence No.	48	61	97	84	81
Sequence Frequency	$3.06x10^{-8}$	$1.22x10^{-8}$	$1.8x10^{-9}$	$3.52x10^{-9}$	$4.39x10^{-8}$
Sequence Designator	LLO-6	MLO-6	SBO-31	SLO-10	SBO-2
CORE/CONTAINMENT RESPONSE					
Time of Core Uncovery (hr)	1.8	9.1	14.8	12.2	12.2
Onset of Core Melt (hr)	2.2	9.9	15.8	13.0	13.1
Time of Vessel Failure (hr)	3.4	11.2	18.1	14.4	15.2
Time of Containment Failure (hr)	> 48.	> 48.	Not Isolated	Not Isolated	Not Isolated
Maximum Containment Pressure (psia)	83.3	42.5	27.1	17.7	29.1
Maximum Containment Temperature (°F) (Upper/Annular Compartments)	341.	287.	469.	269.	424.
Cavity Water Level @ 48 hrs (ft)	0.	0.	0.	0.	Flooded
Fraction of Clad Reacted in Vessel	0.365	0.4	0.538	0.459	0.584
H2 Mass Burned (lbm)	102	981	0.	769	0.
Cavity Concrete Ablation Depth 48 hrs (ft)	5.2	5.0	0.	4.7	0.
FISSION PRODUCT DISTRIBUTION AT END OF MISSION TIME					
Noble Release (%)	0.61	0.46	97.9	98.6	91.5
Volatile FP Release (%)	$1.34x10^{-2}$	$4.2x10^{-3}$	7.6	7.7	4.1
Non-Volatile FP Release (%)	$1.27x10^{-4}$	$1.65x10^{-4}$	$1.68x10^{-2}$	$1.27x10^{-1}$	$5.92x10^{-3}$
Volatile FP Retained in Primary System %	47.3	69.5	75.5	68.9	83.4

TABLE 4.3-6

KEWAUNEE NUCLEAR PLANT
AIRBORNE RELEASE CATEGORIES AND PROBABILITIES

Release Category	Definition	Frequency	Conditional Probability[1,2]
S	No containment failure (leakage only, successful maintenance of containment integrity; containment not bypassed; isolation successful)	2.88×10^{-5}	0.43
T	Containment bypassed with noble gases and more than 10% of volatiles released	5.28×10^{-6}	0.08
G	Containment failure prior to vessel failure with noble gases and up to 10% of the volatiles released (containment isolation impaired)	1.48×10^{-8}	2.24×10^{-4}
A	No containment failure within 48 hr mission time, but failure could eventually occur without accident management action; noble gases and less than 0.01% volatiles released	3.22×10^{-5}	0.49

NOTES:

1. Conditional probability of release category given core damage.
2. Core damage frequency for Level 2 = 6.6×10^{-5}/yr.

TABLE 4.3-7

SUMMARY OF POTENTIAL CONTAINMENT FAILURE MODES

Containment Failure Mode	Frequency	Conditional Probability
Leakage	$2.88x10^{-5}$	0.43
MCCI - Induced failure after mission time (source term limited to leakage)	$2.39x10^{-5}$	0.36
Late containment failure due to overpressurization after mission time (source term limited to leakage)	$8.3x10^{-6}$	0.13
Early containment failure (containment isolation impaired)	$8.58x10^{-9}$	$1.3x10^{-4}$
Early containment failure with fission product scrubbing credited (containment isolation impaired)	$6.19x10^{-9}$	$9.4x10^{-5}$
Containment bypassed - no fission product scrubbing credited	$5.28x10^{-6}$	0.08

4.4 Kewaunee Sensitivity Analyses

NUREG-1335[4] has identified in-vessel and ex-vessel phenomena that could have an impact on containment failure timing and the related source term release given a core damage accident. Sensitivity analyses were performed as part of the Kewaunee IPE to address these phenomena and their uncertainties as well as to provide insights into uncertainties associated with the MAAP modeling assumptions.

4.4.1 Methodology

A three-prong approach was employed in addressing uncertainties associated with the back-end analysis. The primary approach addresses the phenomena identified in NUREG-1335[4] by performing detailed phenomenological evaluation summaries described in section 4.2.5.3. These plant specific evaluations were performed to assess the likelihood of an early containment failure, as well as to determine the structural capacity of the Kewaunee containment. Uncertainties associated with modeling the phenomena identified in NUREG-1335[4] are discussed in the individual evaluations, but do not impact the conclusion of the conservatively based phenomenological summaries.

Table 4.4-1 lists the phenomena identified in NUREG-1335[4] for sensitivity study, along with the means by which these sensitivities are addressed in the Kewaunee IPE. The issues of interest include:

- Hydrogen burn completeness
- In-vessel hydrogen production and core relocation
- Reactor vessel failure mode
- Hot leg creep rupture failure for a high pressure sequence
- Containment failure pressure and area
- Volatile fission product release and retention in the primary system
- Ex-vessel debris coolability

Phenomenological uncertainties not considered in the phenomenological evaluations are addressed by performing MAAP sensitivity studies. These were done by performing changes to various MAAP model parameter in selected base case sequences. The recommended range of these MAAP model parameters for IPE sensitivity analyses is documented in the EPRI document EPRI TR-100167[46]. Table 4.4-2 summarizes the applicable MAAP sensitivities that are performed to meet the requirements of EPRI TR-100167[46]. Sensitivity sequence identifiers are listed in

Tables 4.4-1 and 4.4-2 to correlate the MAAP sensitivity runs to the accident phenomena that they address.

The above two approaches addressed all previously identified phenomenological uncertainties that are relevant to the Kewaunee back-end analysis. Additional uncertainties were identified during the Kewaunee source term investigation and have been incorporated as part of the back-end study. Thus, additional MAAP sensitivity runs were performed to both assess the effect of specific equipment or operator actions on the outcome of particular sequences and to investigate any sequence that would provide greater understanding of accident timing, plant response, or source term release. These additional sensitivity cases are summarized in Table 4.4-3.

4.4.2 MAAP Sensitivity Sequences

Sequences that were used in investigating and modeling uncertainties are based on certain analyzed source term sequences discussed in section 4.3.4. In the sensitivity sequence descriptions that follow, only the deviation from the base case sequences are emphasized. Table 4.4-4 illustrates selected accident progression parameters for each sensitivity sequence, as well as for the associated base case sequence. For convenience, the MAAP sensitivity sequences are identified by their PDS assignment in Table 3.1.5-1 along with a MAAP model parameter or acronym, which addresses the sensitivity, appended to the base case sequence. These identifiers tie Table 4.4-4 back to Tables 4.4-1 through 4.4-3 as well as the base case sequence descriptions in section 4.3.4.

Hydrogen Burn Completeness (Sequence MLO-6_FLPHI)

Purpose: This analysis assessed the effect of the flame flux multiplier on hydrogen burn completeness. The base case source term calculation assumes FLPHI = 2.

Condition: The MAAP model 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 ICS and fan coil units.

Results: The sensitivity results are essentially identical to the base case, except that the flame flux multiplier causes 22 more pounds of hydrogen to burn in containment.

In-vessel Hydrogen Production/Core Relocation (Sequence SBO-30_FCRBLK and LLO-6_FCRBLK)

Purpose: This analysis 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 once melting has started in that node. MAAP's core blockage model accounts for the effects of channel blockage phenomena such as geometric deformation, hydraulic diameter reduction, relocation of unreacted zircalloy to lower core nodes, and surface area-to-volume reduction after melting.

The core blockage model is not used in the base case source term calculation.

Conditions: The core blockage model is implemented by setting the MAAP model parameter FCRBLK equal to 1.

Results: The blockage model was implemented for both a high and low pressure sequence. The overall accident timings and containment response are similar to their perspective base case. For the high pressure sequence, the clad oxidation is reduced to 41% compared to 53% for this high pressure base case. For the low pressure sequence, the clad oxidation is reduced to 28% from 37% for the low pressure base case. Due to the lower in-vessel clad oxidation, more zirconium is available to oxidize in the cavity. With added energy of the highly exothermic oxidization of zirconium, approximately 0.7 feet more concrete is ablated in the sensitivity case. This also results in a slightly higher pressure in containment at the end of the 48 hour mission time. Also, for the low pressure sequence, there is a significant difference in the amount of volatile fission products that are retained in the RCS. In the base case, 47% of the volatile fission product inventory is held up in the RCS, whereas in the sensitivity study, approximately 60% is held up.

Hot Leg Creep Rupture Failure (Sequence SBO-30_HLCR)

Purpose: In the base case, an elevated hot leg/surge line temperature up to 925°F (770 K) occurs prior to vessel failure. These elevated RCS conditions approached the range for potential creep rupture failure of the hot leg. This sensitivity assessed the effects on source term and containment response should creep rupture of the hot leg occur.

Conditions: Hot leg failure based on a 1.07 ft^2 break size is assumed to occur prior to vessel failure when the hot leg temperature exceeds 900°F.

Results: This sequence is quite different from the base case in the sense that the vessel fails at low pressure due to the rapid depressurization caused by the hot leg break. This leads to the debris remaining in the cavity instead of being dispersed into the upper compartment. Since the debris remains in the cavity and no systems are available to cool the debris, 4.5 feet of concrete is ablated in the cavity. This ablation leads to considerable hydrogen being generated (1700 lb at 48 hrs) in containment. Due to the rapid blowdown of steam and water out of the RCS, the reduction in steam inventory causes the clad oxidation fraction to be significantly lower. The pressures and temperatures in containment are significantly higher in the sensitivity case than in the base case. The sensitivity case results in a final pressure of 98.5 psia whereas the base case shows a final pressure of 74 psia.

Failure of RPV Lower Head Due to Creep Rupture (Sequences SBO-30_CREEP and LLO-6_CREEP)

Purpose: MAAP assumes vessel failure occurs approximately 1 minute after core relocation to the lower plenum due to the thermal attack on the lower head penetrations. This failure mode may not always be the case as demonstrated during the TMI-2 accident. If vessel failure does not occur at a lower head penetration, failure is assumed to occur approximately 30 minutes after core relocation due to creep rupture of the reactor vessel lower head. This sensitivity addresses the difference in containment response and source term due to creep failure of the lower head. This sensitivity was performed for both a high and low pressure vessel failure sequence.

Condition: Creep failure of the lower head is achieved by setting the model parameter TTRX, which controls the time delay at vessel failure after core relocation, to 30 minutes and the radius of the failure (XRPV) to the radius of the cylindrical section of the reactor vessel.

Results: The sensitivity cases are nearly identical to the base cases, except that for the low pressure sensitivity cases in which the corium is held up in the primary system 30 minutes longer than the base case, more of the volatile fission products are retained in the primary system.

Reduced Dehris Coolahility (Sequence LLO-4_FCHF)

Purpose: This sequence considers the effect of debris coolability on the source term release. Sequences in which the core debris is flooded are most sensitive to the uncertainty. Therefore, a large LOCA with a flooded cavity was selected since the corium would remains in the cavity.

Conditions: The critical heat flux for the debris-water interface is controlled by the MAAP model parameter FCHF. The base case sequence was performed with FCHF equal to 0.1 and the sensitivity case was analyzed with a value of 0.02 for FCHF.

Results: The reduced debris coolability has no effect on source term, but has a significant effect on the accident progression. Due to the reduced critical heat flux coefficient, the debris in the cavity is not quenched. The corium temperature is above 3000°F until late in the accident when it drops off to approximately 2800°F. Due to these high corium temperatures 3.7 feet of concrete ablation occurred. This concrete ablation causes significant hydrogen production and as a result of the fan coil units reducing the steam concentration, a substantial amount of hydrogen burning occurs in containment. Table 4.4-4 shows the maximum containment pressure and temperature to be 98.6 psia and 1070°F respectively. This is due to a hydrogen burn that occurs approximately 2.5 hours after vessel failure. The final containment temperature and pressure are well below the ultimate containment failure conditions.

Containment Isolation Failure Area

Purpose: The Kewaunee containment isolation notebook does not indicate a failure probability for each individual penetration. Therefore, the 3 failures to isolate end states identified during the binning process were analyzed for 3 different isolation failure sizes (i.e., 36 inch purge line, 18 inch containment vacuum breaker, and 2 inch H_2 control sample line). By analyzing all three of these different size penetrations, the effect of the containment isolation failure area is adequately addressed. This sensitivity addresses the effect on source term for the 3 penetration sizes listed above.

Condition: Containment impairment is modeled in MAAP by adjusting setting the containment failure pressure (PCF) to atmospheric pressure and adjusting the containment failure area (ACFPR) to model the isolation failure area for a 2, 18, or 36 inch penetration.

Results: The sensitivity cases are almost identical to their respective base cases except for the source term releases, which are summarized in Table 4.4-4.

Opening Hatch on Instrument Tunnel (Sequence LLO-4_HATCH and LLO-6_HATCH)

Purpose: Kewaunee's instrument tunnel has two submarine type hatches located approximately 2 feet off the floor of the annular compartment. These hatches are closed during normal operations which renders the cavity dry for all sequences in which RHR recirculation is not available. Since the cavity is dry for most sequences, significant amount of concrete ablation occurs for sequences in which the corium remains in the cavity. By opening this hatch, the cavity would be wet for most cases in which the RWST is injected into containment. This sensitivity analysis investigated the effect on source term and containment response if the hatches in the instrument tunnel were left open. The sequence was analyzed for a large LOCA with no safeguards and one with only ICS and fan coil units available.

Conditions: This sensitivity was achieved by setting the curb height in the lower compartment ZCURBB to approximately 2 feet and the minimum floor area from the cavity to the lower compartment (ATNEX) to 4.5 ft^2.

Results: This sensitivity was performed for two sequences; one with containment heat removal systems available and one without these systems available. For the sensitivity case in which the fan coil units and ICS recirculation are available, the containment response is nearly identical to the base case. For the case in which no containment safeguards are available and the RHR pumps inject the RWST, but failed to go to recirculation, the containment fails in approximately 31 hours. With both the instrument tunnel hatches open, the cavity is flooded and the debris initially quenched. But due to the lack of any containment heat removal or recirculation to cool the water, the RWST is slowly boiled away and fails containment on overpressure. Containment failure is modeled as a large break of the steel shell which results in practically all the noble gases being released and 2% volatiles being released. At the end of the sequence, the volatile fission products are continuously escaping containment at a rate of 0.04% per hour.

Uncoolable Debris Bed in Refueling Pool (SBO-30-ARP)

Purpose: For high pressure vessel failure sequences, the debris that is entrained out of the cavity is displaced into the refueling pool in the upper compartment. MAAP can transport corium to either the refueling pool in the upper compartment or the floor of the lower compartment. Kewaunee's seal table is located one floor (about 34 feet) above the floor of the lower compartment. If the debris is placed in the lower compartment, it could be readily covered by a pool of water. At Kewaunee, the seal table area is not capable of sustaining a pool of water. Therefore, the debris is placed in the refueling pool where it remains dry. This sensitivity addresses the possibility of the debris ending up in an uncoolable configuration on a dry floor.

Condition: This sensitivity was modeled by reducing the refueling pool area (ARP) by 80%. Thus, when the high pressure melt ejection forces the debris into the refueling pool, the large amount of debris is left in deep uncoolable configuration.

Results: This sensitivity case is not much different from the base case except that 4 feet of concrete is ablated in the refueling pool. The ablation is enough to ablate a hole in the floor around the seal table and fall onto the floor below. No equipment should be affected by this relocation of corium. Due to the concrete attack, larger concentrations of hydrogen are present in containment at the end of the sequence.

Spray Recovery Effect on Source Term (Sequence SBO-2_RECOVER)

Purpose: This sequence is quantified as station blackout with power recovery between 2 and 24 hours with a failure to isolate containment. For modeling purpose in the source term analysis, the power is assumed to be recovered at 24 hours. If AC power is recovered earlier, the expected volatile source term could be reduced through continuous operation of the ICS.

Conditions: AC power was assumed to recover at 12 hours.

Results: This sensitivity case is not much different than the base case except that the overall volatile fission product frequency is reduced by approximately 50%. This reduction is due to recovery of the sprays 6 hours after vessel failure. If the sprays are recovered earlier, the source reduction would be even greater.

TABLE 4.4-1

KEWAUNEE SENSITIVITY ANALYSES TO ADDRESS UNCERTAINTIES IDENTIFIED IN NUREG-1335

Phenomenon	Analyses Performed	MAAP Sensitivity Case Identifiers (Where Applicable)
• Performance of containment heat removal systems	• MAAP sequences during the level 2 CET quantification process established containment success criteria for all containment heat removal systems; no further sensitivity analysis is required	N/A
• In-vessel phenomena		
- H_2 production and combustion in containment	• Hydrogen combustion is discussed in detail in a phenomenological evaluation summary (see section 4.2.5.3)	N/A
	• MAAP sequence with an increased value of "flame flux multiplier" to promote burn completeness	MLO-6_FLPHI
	• MAAP sequences (high and low pressure) with core blockage to assess different levels of in-vessel hydrogen production	SBO-30_FCRBLK LLO-6_FCRBLK
- Core relocation characteristics	• MAAP sequences (high and low pressure) with core blockage parameter activated	SBO-30_FCRBLK LLO-6_FCRBLK
- Fuel/coolant interactions	• In-vessel steam explosions addressed in phenomenological evaluation summary (see section 4.2.5.3), no further sensitivity analysis is required	N/A

TABLE 4.4-1

KEWAUNEE SENSITIVITY ANALYSES TO ADDRESS UNCERTAINTIES IDENTIFIED IN NUREG-1335 (Continued)

Phenomenon	Analyses Performed	MAAP Sensitivity Case Identifiers (Where Applicable)
• In-vessel phenomena (continued)		
- Mode of RPV melt-through	• Thrust forces at RPV failure addressed in phenomenological evaluation summary (see section 4.2.5.3), no further sensitivity analysis is required	N/A
- Induced failure of RCS pressure boundary	• Base-case station blackout sequences assumed induced RXCP seal LOCAs of 35 gpm/pump	N/A
	• MAAP sequence with hot leg creep rupture failure performed for a station blackout with no core blockage	SBO-30_HLCR
	• MAAP sequences (high and low pressure) that assume vessel failure occurs due to creep rupture of the lower head instead of failure of a lower head penetration	SBO-30_CREEP LLO-6_CREEP
• Ex-vessel Phenomena		
- Direct containment heating (at high RCS pressure)	• DCH is addressed in a phenomenological evaluation summary (see section 4.2.5.3); no further sensitivity analysis is required	N/A

TABLE 4.4-1

KEWAUNEE SENSITIVITY ANALYSES TO ADDRESS UNCERTAINTIES IDENTIFIED IN NUREG-1335 (Continued)

Phenomenon	Analyses Performed	MAAP Sensitivity Case Identifiers (Where Applicable)
• Ex-vessel Phenomena (continued)		
- Potential for early containment failure due to pressure load	• Potential early containment failure due to ex-vessel steam explosions or hydrogen combustion are addressed in phenomenological evaluation summaries (see section 4.2.5.3), no further sensitivity analyses are required	N/A
- Early failure via debris attack of containment penetrations	• Containment penetration thermal attack is addressed in a phenomenological evaluation summary (see section 4.2.5.3); no further sensitivity analysis is required	N/A
- Long-term core-concrete interaction	• Molten core-concrete interaction (MCCI) is addressed in a phenomenological evaluation summary (see section 4.2.5.3). Also, most base case MAAP analyses show long-term MCCI; no further sensitivity analysis is required	N/A
--Water availability	• MAAP sequence that models hatch on instrument tunnel open to allow water to flow into cavity	LLO-4_HATCH LLO-6_HATCH
--Debris coolability	• MAAP sequence (wet cavity) with reduced critical heat flux multiplier to reduce debris coolability	LLO-4_FCHF

TABLE 4.4-2

SUMMARY OF APPLICABLE MAAP SENSITIVITY STUDIES
SUGGESTED IN EPRI TR-100167

MAAP Parameters	Analyses Performed	MAAP Sensitivity Case Identifiers (Where Applicable)
• ACFPR Containment failure area. Base cases assume 5 ft^2 for large failure mechanism.	No sensitivity analysis required since containment overpressure failure within 48 hours is not observed for any base case sequences.	N/A
• FCHF Coefficient in critical heat flux formula for debris coolability. Base cases assumes 0.1.	Wet cavity case where debris is flooded with water and FCHF = 0.02.	LLO-4_FCHF
• TJBRN Gas jet temperature required for combustion to occur as H_2/CO jet enters an O_2 bearing room. Base cases assume 1060 K (1450°F).	Hydrogen combustion is addressed in detail in the phenomenological evaluation summary (see Section 4.2.5.3).	N/A
• FCRBLK Flag to activate (1) or deactivate (0) the IDCOR core blockage model. Base cases assume FCRBLK = 0.	High pressure station blackout and low pressure LOCA with FCRBLK = 1.	LLO-6_FCRBLK SBO-30_FCRBLK

TABLE 4.4-2

SUMMARY OF APPLICABLE MAAP SENSITIVITY STUDIES SUGGESTED IN EPRI TR-100167 (Continued)

MAAP Parameters	Analyses Performed	MAAP Sensitivity Case Identifiers (Where Applicable)
• TAUTO Auto ignition temperature for hydrogen and carbon monoxide burns. Base cases assumed 983 K (1310°F).	Hydrogen combustion is addressed in detail in the phenomenological evaluation summary (see section 4.2.5.3).	N/A
• FLPHI Flame flux multiplier used in computing burn completeness and rates. Base cases assumed FLPHI = 2.	Low pressure LOCA with CCUs working and FLPHI = 10.	MLO-6_FLPHI
• ABB Induced rupture area for RXCP seal LOCAs or hot leg creep rupture. Base cases assumed no hot leg rupture and, for station blackout sequences, the seal leakage is taken to be 35 gpm/pump.	High pressure station blackout with an induced 1 ft^2 LOCA in the hot leg when the hot leg temperature exceeds 900°F.	SBO-30_HLCR
• PCF Containment failure pressure. Base cases assume 137 psia.	No sensitivity analysis required since base cases use a conservative low-end value (see Section 4.2.5.1). Higher failure pressures are bounded by base-case results.	N/A

Table 4.4-3

ADDITIONAL MAAP SENSITIVITY STUDIES

Classes	Analyses Performed	MAAP Sensitivity Case Identifiers
Base case isolation failures assume 36" containment purge lines not isolated. However, there are other penetrations that are smaller that could fail to isolate.	MAAP sequences with failure to isolate were run with 18″ and 2″ failures to address the possibility of the other penetrations failing to isolate.	SBO-31_18″ SBO-31_2″ SLO-10_18″ SLO-10_2″ SBO-2_18″ SBO-2_2″
Kewaunee's instrument tunnel has 2 hatches, approximately 2 feet off the floor of the annular compartment, that could be left open to allow water into the cavity.	MAAP sequences with ZCURBB = 2.0 ft were analyzed for a large LOCA sequence.	LLO-4_HATCH LLO-6_HATCH
MAAP assumes vessel failure occurs approximately 60 seconds after core relocation due to thermal attack of in-core penetrations. TMI-2 demonstrates that this may not be the case. Instead the RPV would fail much later due to creep rupture of the lower head.	MAAP sequences (high and low pressure) with the tine delay for vessel failure equal to 30 minutes and the radius of vessel failure set equal to the radius of the cylindrical section of the reactor vessel were analyzed.	SBO-30_CREEP LLO-6_CREEP
Station blackout recovery is always assumed to be at 24 hrs, but if recovered earlier, ICS could significantly reduce source term for sequences when the containment is not isolated.	MAAP sequence with recovery of ICS prior to vessel failure.	SBO-2_RECOVER

KEWAUNEE NUCLEAR PLANT
SENSITIVITY ANALYSIS RESULTS
MAAP RUN SUMMARY TABLE

SEQUENCE TYPE	Blackout	Blackout	Blackout	Blackout	Blackout	Blackout	Blackout
Systemic Sequence Identifier	SBO-30	SBO-30	SBO-30	SBO-30	SBO-30	SBO-2	SBO-2
Sensitivity	Base Case	SBO-30_FCRBLK	SBO-30_HLCR	SBO-30_CREEP	SBO-30_ARP	Base Case	SBO-2_18″
CORE/CONTAINMENT RESPONSE							
Time of Core Uncovery (hr)	14.9	14.9	14.9	14.9	14.9	4.3	4.3
Onset of Core Melt (hr) (1200°F)	15.9	15.9	17.0	15.9	15.9	4.9	4.9
Time of Vessel Failure (hr)	18.2	18.3	18.9	18.7	18.2	6.1	6.1
Time of Containment Failure (hr)	> 48.	> 48.	> 48.	> 48.	> 48.	Not Isolated	Not Isolated
Maximum Containment Pressure (psia)	73.9	86.7	98.5	71.3	76.	29.6	30.7
Maximum Containment Temperature (°F) (Upper/Annular Compartments)	443.	439.	359.	439.	404.	424.	415.
Cavity Water Level @ 48 hrs (ft)	0.	0.	0.	0.	0.	Flooded	Flooded
Fraction of Clad Reacted in Vessel	0.526	0.404	0.404	0.531	0.631	0.557	0.643
H_2 Mass Burned (lbm)	89	0.	112	0.	108	0.	0.
Cavity Concrete Ablation Depth @ 48 hrs (ft)	0.	0.	4.0	0.	4.2(*)	0.	0.
FISSION PRODUCT DISTRIBUTION AT END OF MISSION TIME							
Noble Release (%)	0.4	0.42	0.41	0.4	.4	98.9	98.1
Volatile FP Release (%)	2.96×10^{-3}	5.76×10^{-3}	5.94×10^{-3}	4.86×10^{-3}	1.04×10^{-3}	8.5	7.9
Non-Volatile FP Release (%)	8.53×10^{-6}	1.34×10^{-6}	1.27×10^{-4}	4.72×10^{-6}	2.46×10^{-4}	2.08×10^{-2}	5.31×10^{-2}
Volatile FP Retained in Primary System (%)	88.9	84.7	62.6	85.9	94.5	66.6	62.9

NOTE: (*) Concrete ablation would be in the upper compartment.

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KEWAUNEE NUCLEAR PLANT
SENSITIVITY ANALYSIS RESULTS
MAAP RUN SUMMARY TABLE (Continued)

SEQUENCE TYPE	Blackout	SLOCA	SLOCA	SLOCA	Blackout	Blackout	Blackout
Systemic Sequence Identifier	SBO-2	SLO-10	SLO-10	SLO-10	SBO-31	SBO-31	SBO-31
Sensitivity	SBO-2_2″	Base Case	SLO-10_18″	SLO-10_2″	Base Case	SBO-31_18″	SBO-31_2″
CORE/CONTAINMENT RESPONSE							
Time of Core Uncovery (hr)	4.3	12.2	12.1	12.2	14.8	14.8	14.9
Onset of Core Melt (hr) (1200°F)	5.0	13.0	13.0	13.0	15.8	15.8	15.8
Time of Vessel Failure (hr)	6.2	14.4	14.3	14.4	18.1	18.0	18.1
Time of Containment Failure (hr)	Not Isolated	Not Isolated	Not Isolated	Not Isolated	Not Isolated	Not Isolated	Not Isolated
Maximum Containment Pressure (psia)	42.9	17.7	18.2	37.7	27.1	27.4	34.8
Maximum Containment Temperature (°F) (Upper/Annular Compartments)	446.	279.	273.	278.	469.	457.	458.
Cavity Water Level @ 48 hrs (ft)	Flooded	0.	0.	0.	0.	0.	0.
Fraction of Clad Reacted in Vessel	0.574	0.459	0.458	0.458.	0.538	0.623	0.608
H_2 Mass Burned (lbm)	0.	769.	483.	445.	0.	0.	0.
Cavity Concrete Ablation Depth @ 48 hrs (ft)	0.	4.7	4.6	4.6	0.	0.	0.
FISSION PRODUCT DISTRIBUTION AT END OF MISSION TIME							
Noble Release (%)	86.8	98.6	96.5	72.6	97.9	97.1	75.4
Volatile FP Release (%)	2.7	7.7	5.8	1.3	7.6	5.4	1.4
Non-Volatile FP Release (%)	4.15×10^{-3}	1.27×10^{-1}	8.33×10^{-2}	5.52×10^{-2}	1.68×10^{-2}	3.39×10^{-2}	2.7×10^{-2}
Volatile FP Retained in Primary System (%)	79.2	68.9	69.7	84.9	75.5	85.1	89.6

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KEWAUNEE NUCLEAR PLANT
SENSITIVITY ANALYSIS RESULTS
MAAP RUN SUMMARY TABLE (Continued)

SEQUENCE TYPE	LLOCA	LLOCA	LLOCA	LLOCA	LLOCA	LLOCA	LLOCA
Systemic Sequence Identifier	LLO-6	LLO-6	LLO-6	LLO-6	LLO-4	LLO-4	LLO-4
Sensitivity Sequence Designator	Base Case	LLO-6_FCRBLK	LLO-6_CREEP	LLO-6_HATCH	Base Case	LLO-4_HATCH	LLO-4_FCHF
CORE/CONTAINMENT RESPONSE							
Time of Core Uncovery (hr)	1.8	1.8	1.7	1.8	1.7	1.7	1.7
Onset of Core Melt (hr) (1200°F)	2.2	2.2	2.2	2.2	2.1	2.1	2.1
Time of Vessel Failure (hr)	3.4	3.4	3.7	3.2	3.1	3.1	3.1
Time of Containment Failure (hr)	> 48.	> 48.	> 48.	30.8	> 48.	> 48.	> 48.
Maximum Containment Pressure (psia)	83.3	88.8	93.4	136.7	38.1	38.0	98.6
Maximum Containment Temperature (°F) (Upper/Annular Compartments)	341.	348.	348.	340.	276.	276.	1070.
Cavity Water Level @ 48 hrs (ft)	0.	0.	0.	Flooded		Flooded	Flooded
Fraction of Clad Reacted in Vessel	0.365	0.276	0.339	0.344	0.379	0.375	0.375
H_2 Mass Burned (lbm)	102.	247.	245.	0.	0.	0.	692.
Cavity Concrete Ablation Depth @ 48 hrs (ft)	5.2	5.9	5.9	0.	0.	0.	3.7
FISSION PRODUCT DISTRIBUTION AT END OF MISSION TIME							
Noble Release (%)	0.61	0.62	0.62	99.7	0.51	0.51	0.51
Volatile FP Release (%)	$1.34x10^{-2}$	$7.89x10^{-3}$	$7.98x10^{-3}$	2.2	$2.03x10^{-4}$	$1.99x10^{-4}$	$1.99x10^{-4}$
Non-Volatile FP Release (%)	$1.27x10^{-4}$	$1.47x10^{-4}$	$1.37x10^{-4}$	$1.22x10^{-4}$	$1.53x10^{-6}$	$2.19x10^{-6}$	$2.19x10^{-6}$
Volatile FP Retained in Primary System (%)	47.3	59.6	61.7	64.4	52.7	52.7	53.2

KEWAUNEE NUCLEAR PLANT
SENSITIVITY ANALYSIS RESULTS
MAAP RUN SUMMARY TABLE (Continued)

SEQUENCE TYPE	MLOCA	MLOCA	Blackout	Blackout
Systemic Sequence Identifier	MLO-6	MLO-6	SBO-2	SBO-2
Sensitivity	Base Case	MLO-6_FLPHI	Base Case	SBO-2_RECOVER
CORE/CONTAINMENT RESPONSE				
Time of Core Uncovery (hr)	9.1	9.1	4.3	4.3
Onset of Core Melt (hr) (1200°F)	9.9	9.8	4.9	4.9
Time of Vessel Failure (hr)	11.2	11.0	6.1	6.1
Time of Containment Failure (hr)	> 48.	> 48.	Not Isolated	Not Isolated
Maximum Containment Pressure (psia)	42.5	41.7	29.6	29.6
Maximum Containment Temperature (°F) (Upper/Annular Compartments)	287.	285.	424.	429.
Cavity Water Level @ 48 hrs (ft)	0.	0.	Flooded	Flooded
Fraction of Clad Reacted in Vessel	0.4	0.42	0.557	0.557
H_2 Mass Burned (lbm)	981.	1003.	0.	0.
Cavity Concrete Ablation Depth @ 48 hrs (ft)	5.0	5.0	0.	0.
FISSION PRODUCT DISTRIBUTION AT END OF MISSION TIME				
Noble Release (%)	0.46	0.46	98.6	95.5
Volatile FP Release (%)	4.2×10^{-3}	4.34×10^{-3}	8.5	4.6
Non-Volatile FP Release (%)	1.65×10^{-4}	1.79×10^{-4}	2.08×10^{-2}	2.08×10^{-2}
Volatile FP Retained in Primary System (%)	69.5	69.5	66.6	77.1

4.5 Summary of Back-End Results

The design of the Kewaunee containment reduces the frequency and magnitude of potential radiological releases. The large, dry containment provides for approximately 1.32 million cubic feet of free volume. A containment failure analysis reveals that the containment can withstand more than three times the design pressure. The structural strength and volume features allow the containment to withstand significant mass and energy releases without failing.

The Kewaunee source term analysis has estimated sequence progression and source term releases for those sequences that are significant contributors to the total Kewaunee core damage frequency. The back-end analysis considers and reports source term results for over 99% of the total core damage frequency. Not only has this provided quantitative information regarding containment failure probabilities and source term releases, but it has yielded useful, quantitative insights into the performance of the Kewaunee containment as well.

Results of the Kewaunee back-end analysis revealed that none of the analyzed core damage sequences go to containment failure within 48 hours. Thus, most of the fission products are retained for an extended period of time, allowing settling and impaction mechanisms to reduce the airborne fission product inventory available for release should the containment eventually fail. Also, volatile fission product releases greater than 0.01% only occur in those sequences in which the containment is bypassed or impaired. The frequency of an uncontrolled fission product release (due to containment bypass or impairment) is 5.3×10^{-6} per reactor year. This volatile release frequency is dominated by steam generator tube rupture events (5.28×10^{-6}/yr), but is also due to interfacing systems LOCA (7.4×10^{-9}/yr) and containment isolation failures (1.48×10^{-8}/yr).

Table 4.3-6 summarizes the source term results by release category, and shows the conditional probability of each release category given core damage. These results show that, should a core damage event occur at Kewaunee, there is a 92% probability of containment success. In other words, there is a 92% probability that the final barrier to fission product release is not breached, impaired, or bypassed within the 48 hour mission time. However, a significant fraction of the core damage events at Kewaunee (49%) require some additional recovery actions not credited in the IPE in order to prevent eventual containment failure at some time beyond 48 hours. It is anticipated that actions such as refilling the RWST, recovery of failed containment safeguards equipment, along with staffing of the emergency operations facility and technical support center would mitigate these sequences and result in a safe, stable configuration within containment.

Other insights pertaining to the Kewaunee containment are:

- The most important feature of the Kewaunee containment with respect to fission product retention is its ability to remain intact for tens of hours following core damage. This robustness allows for natural fission product deposition mechanisms to remove airborne fission products from the containment atmosphere, and provide adequate time for additional accident mitigation activities to be implemented.

- Kewaunee is not vulnerable to early containment failure.

- The RCS provides good fission product retention, even after vessel failure and during containment bypass sequences, due to deposition on primary system structures.

- Sensitivity studies were performed with the hatches on the instrument tunnel open and closed. Presently the hatches are closed, this results in extensive concrete ablation for sequences where RHR recirculation is available after vessel failure. Since the containment pressurization is due solely to non-condensible gas generation, containment failure does not occur within the Level 2 mission time. With the hatches open, containment failure occurs for sequences in which RHR recirculation is not successful and no containment heat removal is established. When containment heat removal is available, the hatches allow water in the cavity to prevent core-concrete attack.

5.0 UTILITY PARTICIPATION AND INTERNAL REVIEW TEAM

5.1 IPE Program Organization

Prior to initiating the IPE project, WPSC management decided that the large magnitude of resources necessary to meet this commitment warranted an internal commitment for long-term use of PRA in decisions associated with Kewaunee. With this decision, WPSC chose to establish a group whose primary function was to develop and apply the Kewaunee PRA. When this project began, WPSC staff had the limited PRA experience associated with performance of a PRA on the auxiliary feedwater system at Kewaunee. For this reason, outside contractor support was obtained to train the WPSC personnel involved in the Kewaunee PRA and to work with them in the initial stages of each portion of the project. Westinghouse Electric Corporation was contracted for Level 1 PRA support, and their IPE partner Fauske and Associates, Incorporated for the Level 2 containment performance analysis.

The WPSC PRA staff consists of a group supervisor who serves as the project manager of the Kewaunee PRA, one engineer and one former senior reactor operator. All three members of the group have operations experience as one member was once a shift supervisor and the other two are former shift technical advisors. In addition, the group members have experience in operator training, licensing, core thermal hydraulics, design modifications and technical support of the Kewaunee plant. Initially, WPSC had only 2 people working the PRA project and they were performing approximately 50-60% of the work, once the third member was added, WPSC gradually took over all of the Level 1 work and a larger portion of the Level 2 work. The PRA received a great deal of support from other departments in the nuclear organization during the different phases of the project as expertise in different areas was required. The WPSC PRA staff has maintained a good working relationship with the PRA contractors which has allowed for a very thorough technology transfer to the WPSC staff.

The following describes the task-by-task participation of the WPSC PRA staff in the development of the Kewaunee PRA:

1. Data Collection and Analysis - WPSC collected all the plant specific data and developed the data base used to support determining initiating event frequencies and equipment failure data. Support was received from the engineer in the maintenance department responsible for Kewaunee's Reliability Centered Maintenance program and operations personnel who helped classify plant trips by initiators.

2. Initiating Event Analysis - WPSC developed the initiating event frequencies for all events based on recommended methodologies from Westinghouse.

3. Accident Sequence (Event Tree) Analysis - Initially WPSC worked closely with Westinghouse on the development of event trees. Once initial core melt quantification was complete, WPSC took over the entire effort incorporating changes based on internal and external review efforts, MAAP results, etc.

4. System Analysis - WPSC PRA staff developed all of the system fault trees and notebooks associated with the Kewaunee PRA except the notebook and the 128 (75 which are used in the current models) fault trees associated with reactor protection, engineered safeguards feature actuation circuitry and diesel generator sequencer circuitry. The Westinghouse analyst who performed this work received a great deal of technical support from the WPSC PRA group, Kewaunee plant maintenance engineering and Kewaunee plant instrumentation and control engineering.

5. Human Reliability Analysis - Westinghouse provided WPSC training in the area of human reliability analysis. WPSC performed the actual analysis in-house. Support was received from operations crew members and staff.

6. Core Melt Quantification - WPSC performed all of the core melt quantifications and assessed the results using the Westinghouse WLINK Code.

7. Plant Damage State Quantification - WPSC performed all of the plant damage state quantification and assessed the results using the Westinghouse WLINK code.

8. Containment Performance Analysis - Fauske and Associates, Incorporated (FAI) was the lead for this effort. WPSC PRA staff performed all of the MAAP runs and authored portions of the containment performance notebook. WPSC also reviewed and approved all other portions of this analysis. Due to the close proximity of WPSC and FAI, personnel from the two companies regularly met face-to-face and worked together on this effort.

9. Sensitivity Analysis - WPSC performed all sensitivities with guidance from Westinghouse except for the sensitivity run that lowered the cutset cutoff. This was performed by Westinghouse on their RS6000 machines as it could not be performed using the software configuration setup on the WPSC 486 PC's.

10. Internal Flooding Analysis - The internal flooding analysis was primarily performed by Westinghouse. WPSC participated in the plant walkdowns, provided the engineering and operation support, and reviewed all aspects of the analysis.

5.2 Composition of Independent Review Team

The Kewaunee PRA received several reviews over the course of the project. An independent internal review by senior WPSC staff was conducted in accordance with the requirements of Generic Letter 88-20[1]. In addition, an in-depth independent review by outside experts was conducted to further ensure the quality of the PRA. Finally, one of the WPSC PRA staff members was on the independent review team for the Point Beach PRA which indirectly served as a further review of the Kewaunee PRA. Each of these reviews is further discussed below.

Independent Internal Review

An experienced group of Kewaunee plant staff members performed a review of the Kewaunee PRA. The group was given 2½ days of PRA training on each of the different tasks by the WPSC PRA group, Westinghouse and FAI. It was decided to have the reviewers review areas that they were most familiar with. The titles and operations background of each of the review team members are contained in Table 5.2-1.

Independent External Review

Because WPSC staff had little PRA experience prior to performing the Kewaunee PRA, they relied heavily on training from Westinghouse and FAI in order to perform the work in-house. This presented a possible problem as WPSC had little expertise to question the methodology and techniques recommended by Westinghouse and FAI. Therefore, WPSC chose to have a six person team of experts from other companies spend one week in Green Bay reviewing each of the tasks and the methodology behind them. Table 5.2-2 contains a list of the reviewers. One of the reviewers was from nearby Wisconsin Electric Power Company which owns and operates the Point Beach Nuclear Plant. Point Beach Units 1 and 2 are of similar design and vintage as Kewaunee and it proved invaluable to have someone familiar with the operation of a 2 loop Westinghouse PWR on the external review team.

Point Beach Independent Review

In addition to having a Point Beach engineer on the Kewaunee independent external review, WPSC supported the Point Beach independent review with one of the Kewaunee PRA staff members. Serving as a member of the Point Beach review provided WPSC with additional insights and ways to improve the Kewaunee PRA. WPSC has maintained close contact with the PRA staff at Wisconsin Electric throughout the IPE process which has also helped in the quality of the Kewaunee PRA.

TABLE 5.2-1

KEWAUNEE PRA INTERNAL INDEPENDENT REVIEW TEAM

TITLE	OPERATIONS EXPERIENCE
Shift Supervisor	SRO
Control Room Supervisor	SRO
Plant Operations Supervisor	SRO
Plant Operations Engineer	SRO
Reactor Engineering Supervisor	STA
I&C Engineering Supervisor	STA
Plant Maintenance Superintendent	STA
Plant Maintenance Engineer (Electrical)	STA
Senior Simulator Instructor	SRO

TABLE 5.2-2

KEWAUNEE PRA EXTERNAL INDEPENDENT REVIEW TEAM

COMPANY	TITLE	AREA OF EXPERTISE
Battelle	Research Leader	Level 2 PRA, Definition of Plant Damage States, Severe Accident Phenomena
Battelle	Senior Research Scientist	Human Reliability Analysis
Safety Management, Inc.	Principal	Level 1 PRA
Sargent & Lundy	Structural Engineer	Containment Capability & Failure Modes
Sargent & Lundy	Manager, Analysis	Level 2 PRA, Severe Accident Analysis, Thermal-Hydraulic Analysis, Containment Analysis
Wisconsin Electric	Nuclear Engineer	Level 1 PRA, Plant Configuration & Operation

5.3 Areas of Review and Major Comments

Independent Internal Review

The internal review group worked on its review of the Kewaunee PRA on-and-off for 6 months and generated over 450 separate comments. The reviewers categorized their comments into one of three categories: technical comments, suggestions, and editorial comments. The larger percentage of comments were technical. Many of the comments identified areas that they felt the Kewaunee PRA was too conservative, i.e. that there was more that could be done to prevent core damage given a particular sequence. Those activities that were proceduralized were incorporated, those that were not proceduralized were deferred until WPSC implements an accident management program. In almost all cases, WPSC chose not to model recovery actions that were beyond the scope of procedures and the operator training program. A couple of examples of technical comments received from the internal review are:

- The event tree for a steam line break event required the injection of one boric acid storage tank for reactivity control. Two reviewers questioned the need for this. The PRA group requested support from the nuclear fuels group who determined via DYNODE simulation that BAT injection was only needed for power Levels below 10%. The PRA modified the event tree to incorporate this recommendation.

- A fault tree used for RCS depressurization in the steam generator tube rupture event only included the use of the pressurizer PORVs. This conservative modeling was questioned by a reviewer and the change was made to include depressurization via normal and auxiliary pressurizer spray as they are also proceduralized.

The PRA group will maintain the records from the internal review which are available for NRC review if necessary.

Independent External Review

The external review group performed the majority of their review at the WPSC corporate office during the week of April 6-10, 1992. The review carried over into subsequent weeks in certain areas especially the review of the human reliability analysis. As opposed to the internal review which was an in-depth review of each notebook, fault tree, etc., the external review was more concerned with the methodologies used and overall project quality. In-depth reviews were performed on selected portions of each area and those areas that WPSC staff does not have a great deal of experience. These areas were Level 2 PRA, containment analysis, human reliability analysis, and common cause. Some of the specific improvements made as a result of the external review were: re-performing the common cause evaluation, using recent guidance for determining initiating event frequencies such as that for the interfacing systems LOCA event, and many others.

The PRA group will maintain the records from the external review which are available for NRC review if necessary.

Point Beach Independent Review

By serving on the Point Beach independent review team, WPSC gained insights that helped improve the Kewaunee PRA. Two examples of comments in the Point Beach review that were incorporated into the Kewaunee PRA are: a more complete model of the service water system intake piping and components, and consideration of the subtle interactions listed in NUREG/CR-4550 Volume 1[9].

5.4 Resolution of Comments

All of the comments made by the three review efforts have either been incorporated, deferred until accident management, or dispositioned.

6.0 PLANT IMPROVEMENTS AND UNIQUE SAFETY FEATURES

6.1 Level 1 Unique Safety Features

Based on performance of the level 1 PRA analysis, several features of the Kewaunee design have been identified that reduce the likelihood of core damage. These include:

- High head safety injection pumps inject at 2200 psig which is significantly higher than typical Westinghouse plants designated as low pressure plants.
- Containment sump recirculation can be aligned to the high head safety injection, low head safety injection and containment spray pumps from the control room.
- Three auxiliary feedwater (AFW) pumps (two motor-driven and one turbine driven for diversity) which are independent of cooling water systems. The service water system services as a backup suction supply to the three AFW pumps.
- Separate eight hour batteries for safeguards and non-safeguards equipment.
- Four safety related service water pumps for a single unit site.
- The chemical volume and control system has three positive displacement charging pumps which are independent of cooling water systems. One of the pumps is driven by a variable speed DC motor for speed control and is not dependent on instrument air for attaining maximum pump output.
- Two independent methods for maintaining reactor coolant pump seal integrity, seal injection from the charging pumps and thermal barrier cooling via the component cooling water system.

6.2 Level 2 Unique Safety Features

Based on performance of the Level 2 PRA analysis, several features of the Kewaunee design have been identified that reduce the likelihood of containment failure.

The first of these features is the Kewaunee containment heat removal capability. The plant is designed with four containment fan cooling units (FCUs) and two internal containment spray (ICS) trains, each equipped with heat exchangers. Only one FCU or ICS is needed to preclude containment failure on overpressure. The FCU discharge piping is at high enough elevation to preclude the discharge from being submerged following RWST injection.

The Kewaunee containment free volume is such that complete oxidation of the fuel cladding does not produce enough hydrogen to challenge the containment structure. The open design promotes good communication between compartments, precluding hydrogen pocketing.

The geometry of the cavity and instrument tunnel is such that deentrainment of debris following high pressure melt ejections will occur, precluding direct containment heating (DCH) as a concern. The cavity floor is large enough to allow the debris to spread into a thin layer, allowing coolability through an ablated vessel. This minimizes the likelihood of non-volatile fission product release.

In general the containment design was found to be capable of handling severe accidents. The conservative 95% confidence containment ultimate pressure is 2.7 times the design pressure. The containment penetrations are capable of withstanding high temperature conditions for extended periods of time. These safety features, inherent in the design of containment, allow the containment structure to respond to severe accidents.

6.3 Level 1 Plant Improvements

The Kewaunee PRA identified some vulnerabilities that were significant enough to warrant improvements. Table 6-1 lists the improvements that have been made or are scheduled to be made and the initiating event associated with the improvement. Although some of these improvements have not been made as of this date, they are reflected in the Kewaunee PRA because they have been approved by plant management, and are scheduled to be completed in the near future. One other plant improvement that has yet to be implemented is reflected in the Kewaunee PRA. This is the design modification that will be implemented during the 1993 refueling outage at Kewaunee to meet the NRC station blackout rule. This modification is described in section 3.2 of this report.

In reviewing the sensitivity and importance analyses, a few other possible plant vulnerabilities were identified. These are described in section 3.4.3 of this report. WPSC staff is currently evaluating each of the following areas to determine what actions are required:

1. Valve MU-3A, normal makeup from the condensate storage tank to the condenser hotwell, currently fails open on loss of instrument air and/or loss of DC control power. The emergency makeup valve, MU-3B, fails closed in both cases and this also appears to be the preferred position for MU-3A. WPSC staff is reviewing design information to determine the basis for the current fail safe position of MU-3A.

2. WPSC has been working on improving the reliability of the turbine driven auxiliary feedwater (TDAFW) pump for several years. Plant engineering staffs have determined what actions need to be taken and modifications are currently scheduled.

3. The decrease in reliability of two of the older instrument air compressors has led WPSC to initiate a design modification to remove the two air compressors and replace them with two air cooled air compressors. One air cooled air compressor has already been installed; the other has been purchased and will be installed in the near future.

4. There have been several instances where the relief valves at the discharge of the charging pumps have lifted when a pump has started or sped up. Therefore, WPSC is investigating what actions are necessary to correct this problem in order to prevent diversion of chemical and volume control system water.

6.4 Level 2 Plant Improvements

A Level 2 vulnerability would be indicative of unusually poor containment performance, these include events resulting in containment impairment (failure to isolate), containment bypass, and early containment failure. With an impairment or bypass, the containment is placed in a state where fission product retention within containment cannot be achieved. Early containment failure would allow for releases prior to deposition within containment.

The frequency of impairment and early overpressure failure are well below the reporting requirement of 10^{-7} (Appendix 2 to Generic Letter 88-20). The bypass frequency of 5.28×10^{-6} is dominated by steam generator tube ruptures. These cases are consistent with industry experience and easily remedied with procedural enhancements to refill the RWST and maintain water to the secondary of the ruptured steam generator. These types of enhancements will be considered in the Kewaunee severe accident management program. Kewaunee, it is concluded, does not exhibit any Level 2 vulnerabilities.

TABLE 6-1 **PLANT IMPROVEMENTS INITIATED BY THE IPE**		
IMPROVEMENT	INITIATING EVENT	SCHEDULE
Performed leak testing of an additional four valves serving as a boundary between the reactor coolant system and a low pressure system	Interfacing Systems LOCA	End of Refueling Outage 1993
Modify the normal position of two motor operated valves located on the low pressure safety injection line from open to closed	Interfacing Systems LOCA	End of Refueling Outage 1994
Modify emergency operating procedure ECA 1.2 to improve guidance to the operators in identifying and mitigating an interfacing systems LOCA	Interfacing Systems LOCA	Summer 1993
Modify the swing direction of three doors separating the turbine building basement with areas containing safeguards equipment in order to reduce the likelihood of a turbine building basement flood propagation into these other areas.	Internal Flooding	2 Doors in 1992 Refueling Outage 1 Door in 1993 Refueling Outage
Improved the inspection method for rubber expansion joints to identify possible flooding problems before they occur	Internal Flooding	1992 Refueling Outage
Modify emergency operating procedures to provide instruction for switching the power supply to bus 5262 in the event of the loss of either safeguards bus 5 or 6 in order to have power available to 2 instrument air compressors	Loss of Offsite Power, Station Blackout	Summer 1993

7.0 SUMMARY AND CONCLUSIONS

The objectives for the Kewaunee PRA encompassed those presented in Generic Letter 88-20 with the addition of several others. These additional objectives are consistent with the intent of the generic letter and include:

1. Satisfying the requirements of GL 88-20.

2. Developing a living PRA of Kewaunee that can be used as a tool in decision making for the life of the plant.

3. Gain additional insight in the area of the effects, mitigation and prevention of severe accidents at Kewaunee.

4. Identify potential improvements in the plant design and operation that will reduce the overall core damage frequency or the containment failure frequency.

This report completes the activities associated with the first objective and describes the actions taken and results associated with this extensive effort. A Level 1 PRA for the internal initiating events including flooding and a limited scope Level 2 containment performance analysis were performed. The resultant core damage frequency including internal flooding is 6.65E-5/year. Results of the Kewaunee back-end analysis show that, should a core damage event occur, there is a 92% probability of containment success. In other words, there is a 92% probability that the final barrier to fission product release is not breached, impaired, or bypassed within the 48 hours mission time. However, the majority of the core damage events at Kewaunee (51%) require some additional recovery actions not credited in the PRA in order to prevent containment failure at some time beyond 48 hours.

The second objective, although not a requirement from the NRC, provides justification for committing significant financial and personnel resources on this project. WPSC has been and plans to continue using the Kewaunee PRA in making decisions in numerous areas associated with plant safety. WPSC also plans on using the Kewaunee PRA to support its accident management program in response to the forthcoming NRC requirements. In fact, the PRA group has developed a list of possible accident management procedures based on risks determined in the PRA effort.

The results of objectives 3 and 4 are discussed in detail in sections 3.4 and 6. Several important insights were gained in the areas of design, operation and maintenance of the Kewaunee plant. These insights are resulting in several plant improvements that may never have been identified without the use of PRA.

In summary, all four objectives of the Kewaunee PRA have been successfully met, and that WPSC has a new tool for making evaluations and decisions associated with the safe operation of the Kewaunee Nuclear Power Plant.

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