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INTERIM RELIABILITY EVALUATION PROGRAM
BROWNS FERRY 1

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ABSTRACT

Probabilistic risk analysis techniques, i.e., event tree and fault tree analysis, were utilized to provide a risk assessment of the Browns Ferry Nuclear Plant Unit 1. Browns Ferry 1 is a General Electric boiling water reactor of the BWR 4 product line with a Mark 1 (drywell and torus) containment. Within the guidelines of the IREP Procedure and Schedule Guide,[1] dominant accident sequences that contribute to public health and safety risks were identified and grouped according to release categories.

INTRODUCTION

EG&G Idaho, Inc. was contracted by Sandia National Laboratories to perform a risk assessment of the Browns Ferry Nuclear Plant Unit 1 in support of the Nuclear Regulatory Commission's (NRC) Interim Reliability Evaluation Program (IREP). The analysis includes accident identification, event tree/fault tree construction and quantification, and accident sequence evaluation. The analysis was limited in scope to identifying only those risk-significant accident sequences which lead to core damage and provide the mechanism for release of radionuclides to the environment.

ANALYSIS TEAM STAFFING

Figure 1 shows the make-up of the analysis team. Jack E. Trainer served as team leader for the project. The principal analysts responsible for conducting the risk assessment were Steve Mays, Walt Sullivan,

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DISCLAIMER

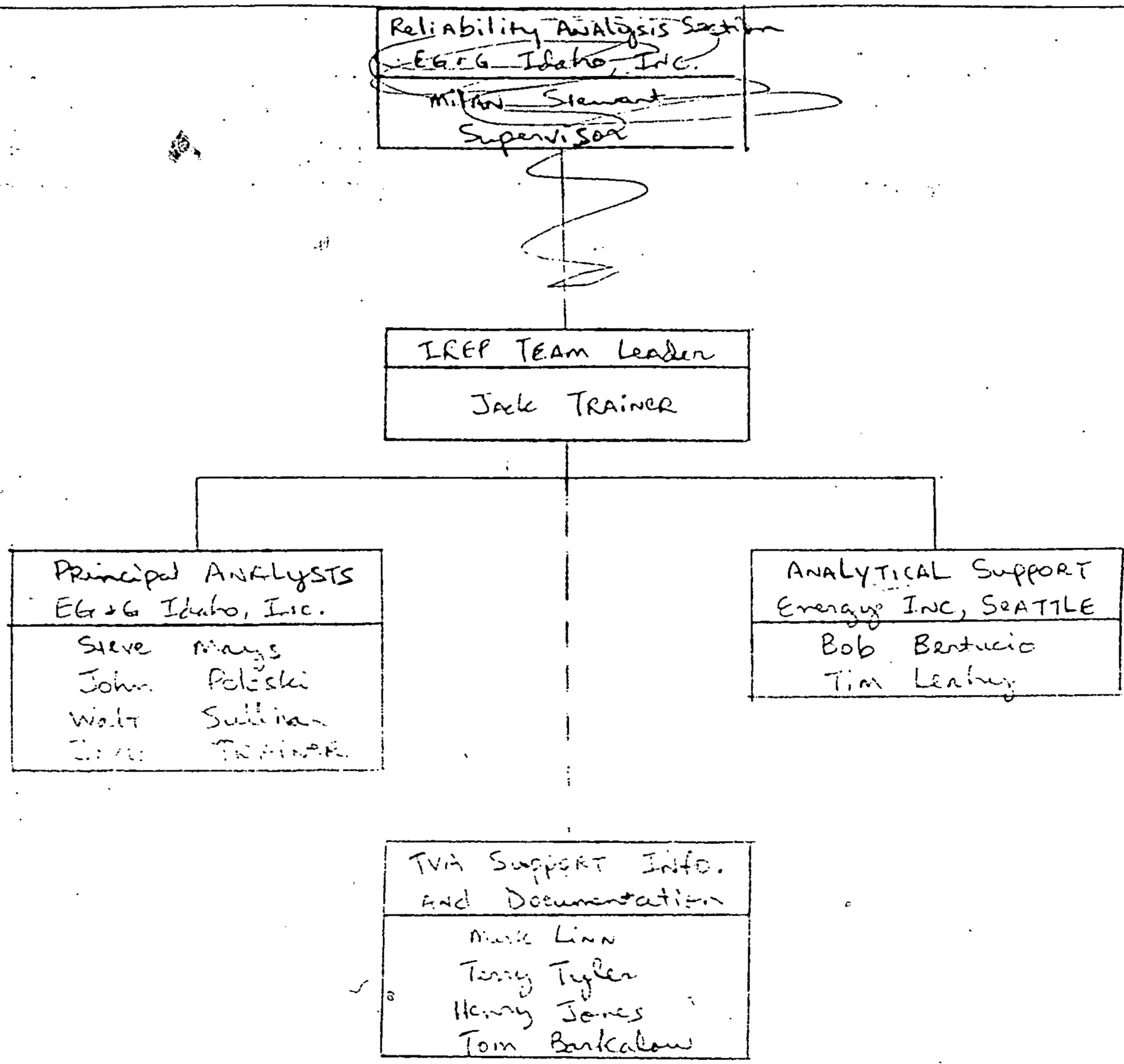


Figure 1 - Analysis Team

John Poloski, and Jack Trainer of EG&G Idaho, Inc. Bob Bertucio and Tim Leahy of the Seattle Office of Energy Incorporated provided analytical support to assist EG&G in the study.

Utility support from TVA was coordinated by Mark Linn with assistance from Terry Tyler, Henry Jones, and Tom Barkalow. Unlike other IREP teams which had a full-time participant from the utility, the Browns Ferry IREP team had to rely on telephone calls, mail, and occasional meetings with TVA personnel for information exchange. The TVA support included documentation of plant design, analyses beyond those found in the FSAR, and verification of system operating characteristics.

PLANT DESCRIPTION

Browns Ferry Nuclear Plant, Unit 1 is a General Electric-designed boiling water reactor of the BWR-4 class with a Mark 1 containment. The plant is rated to produce 1100 MWe of power. The primary differences in the reactor systems of this plant as opposed to earlier BWR plant designs include

- Variable speed recirculation pumps which discharge into jet pumps that are arranged around the periphery of the reactor vessel.
- An integrated Core Standby Cooling System including High Pressure Coolant Injection (HPCI), Low Pressure Core Spray (CS), Automatic Depressurization (ADS), and Residual Heat Removal (RHR) systems.
- An integrated RHR system providing Low Pressure Coolant Injection (LPCI), Shutdown Cooling, and Containment Cooling modes of operation.
- A Reactor Core Isolation Cooling (RCIC) system instead of an Isolation Condenser for mitigating transients where the reactor is isolated from the main condenser.
- LPCI Loop Selection Logic has been disabled and the LPCI discharge header cross-connection valve closed.

The containment design features include

- A drywell enclosing the reactor coolant system.
- A wetwell (or torus) connected to the drywell and designed to provide energy suppression in the event of a Loss of Coolant Accident and to provide a source of water for injection into the reactor.
- A reactor building surrounding the drywell and torus, that houses the Core Standby Cooling Systems and provides a second barrier between the reactor and the plant environment.

Figure 2 provides a simplified diagram of the safety-related design features.

The Standby AC power system at the Browns Ferry Station is different from most other commercial nuclear power plants. There are 8 diesel generators located at the Browns Ferry Station. Four of these diesel generators provide emergency AC power to four independent shutdown buses when normal AC power to the buses is lost. These shutdown buses support emergency loads for both Units 1 and 2 as depicted by Figure 3. The other four diesels serve Unit 3 but can be manually cross-connected to the Unit 1 and 2 shutdown buses.

The 250V DC system necessary for transient and accident mitigation is completely shared among the three units. Three 250V DC battery boards provide power the Class 1E loads in all three units. A fourth battery board provides power for other non-Class 1E loads.

The RHR Service Water System (RHRSW) and Emergency Equipment Cooling Water System (EECW) are support systems which serve all three units.

One unusual dependency of this arrangement of electric power and cooling water systems is that the pumps of the RHRSW and EECW systems which are necessary for successful operation of accident mitigating systems in Unit 1 require AC power from the standby AC power system of Unit 3. Figure 4 shows the RHRSW pumps and their corresponding diesel generators.

ANALYSIS

EPRI document NP-801[2] and the Browns Ferry Nuclear Plant FSAR[3] served to identify transient events and loss-of-coolant accidents (LOCAs) that could result in core damage if the reactor were not shutdown and decay heat removed. Functional event trees were used initially to develop the mitigating requirements for the accident and transient events identified. From the functional development evolved the systemic event trees depicting the dependencies and requirements of the mitigating systems. Three ranges of LOCAs were identified (large, intermediate, and small) in terms of the phase of coolant lost, i.e., steam or liquid. The large LOCAs were further classified into recirculation system discharge-side, and recirculation system suction-side liquid breaks, and steam breaks. The various classes of LOCAs required separate event trees to account for the differences in success criteria of the LOCA mitigating systems. A functional event tree and resulting systemic event tree for a large break on the discharge-side of a recirculation pump are shown as Figures 5 and 6 respectively. The transients were divided into three groups; transients where the Power Conversion System (PCS) remained available, transients where the PCS was not available due to the transient initiator, and the loss of offsite power (LOSP) transient. Additionally, trees were developed for transient-induced LOCAs caused by stuck open relief valves (SORV). The total number of event trees evaluated was eleven.

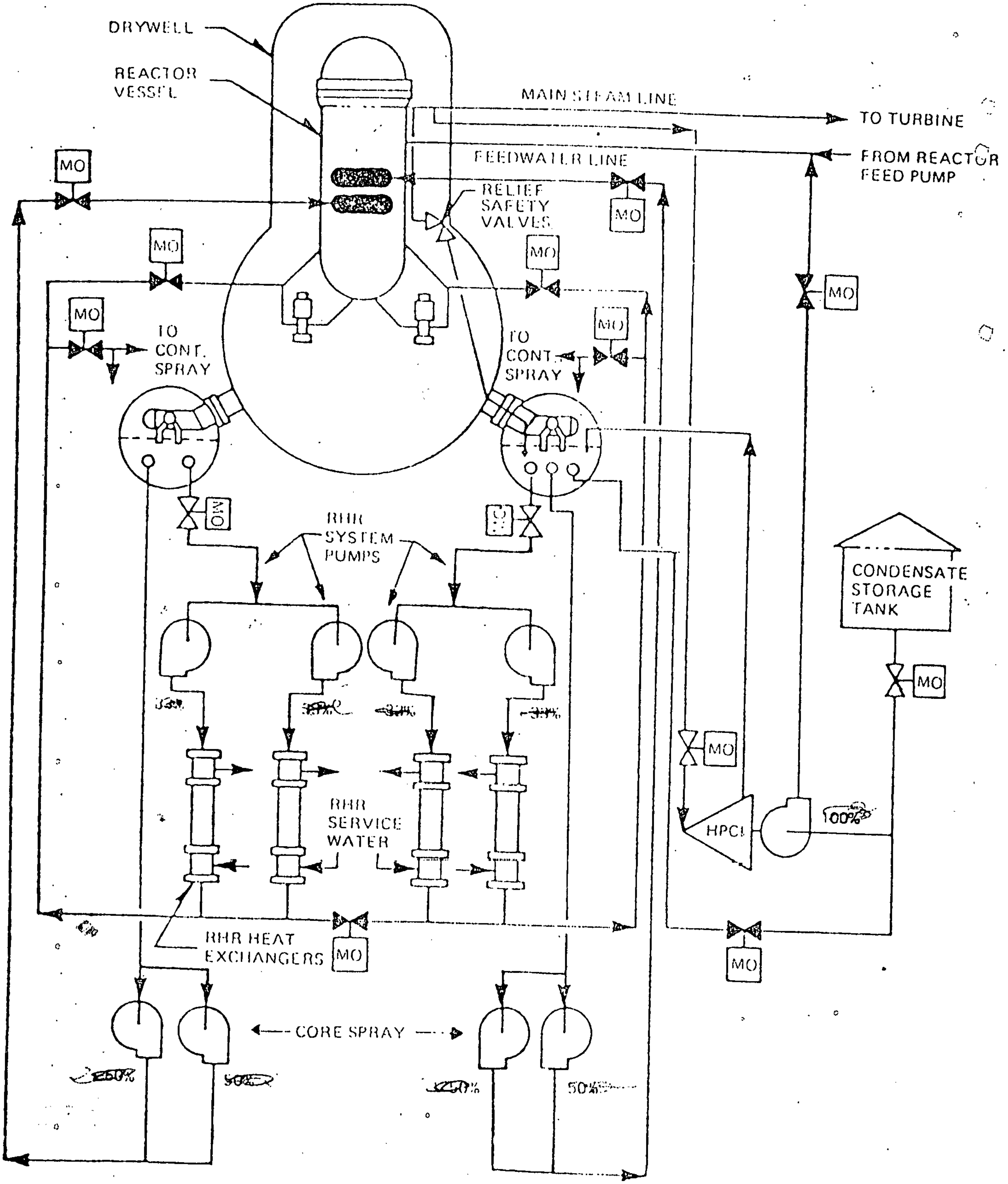


Figure ~~II.(?)~~ 2 Emergency Core Cooling Systems

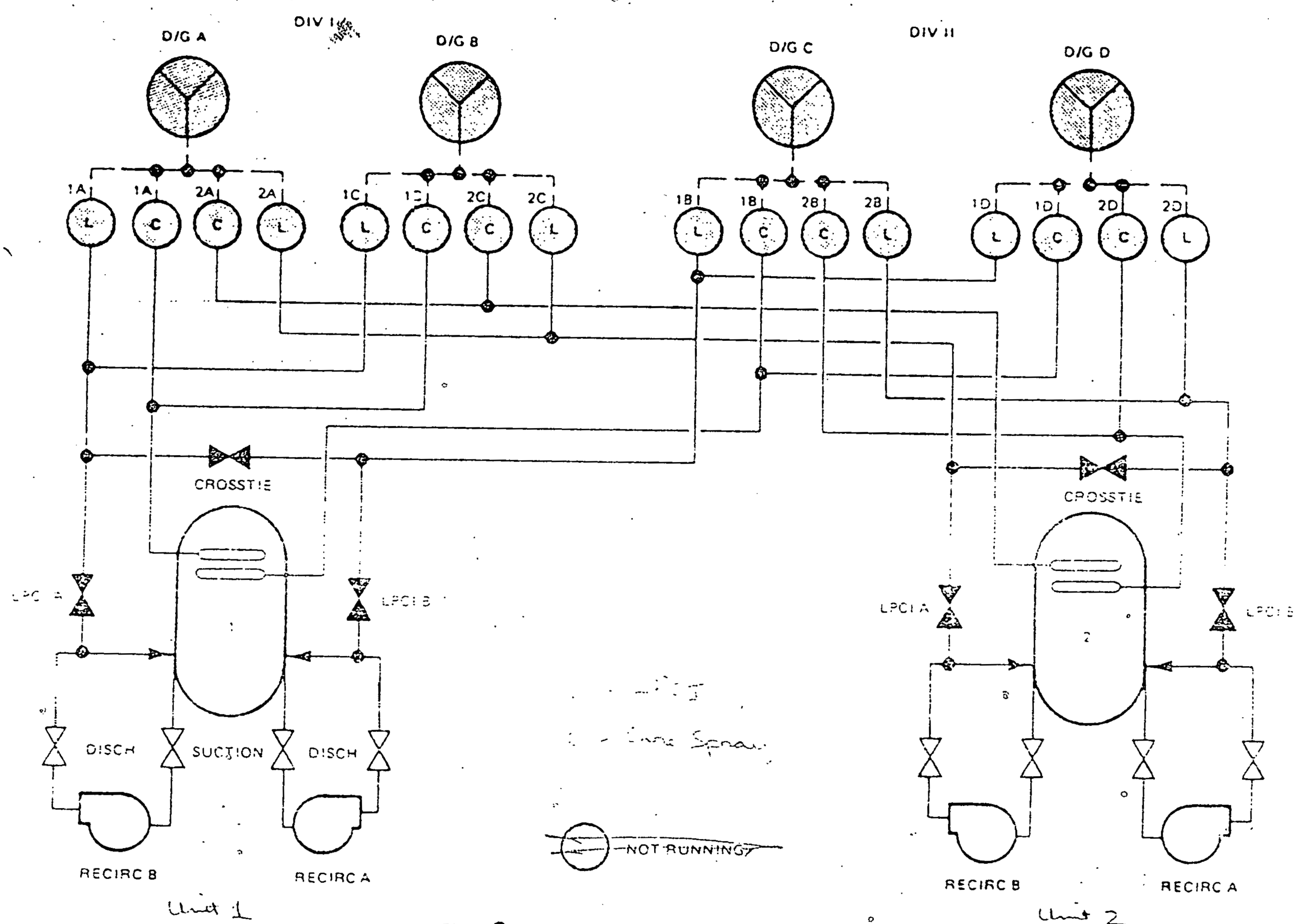
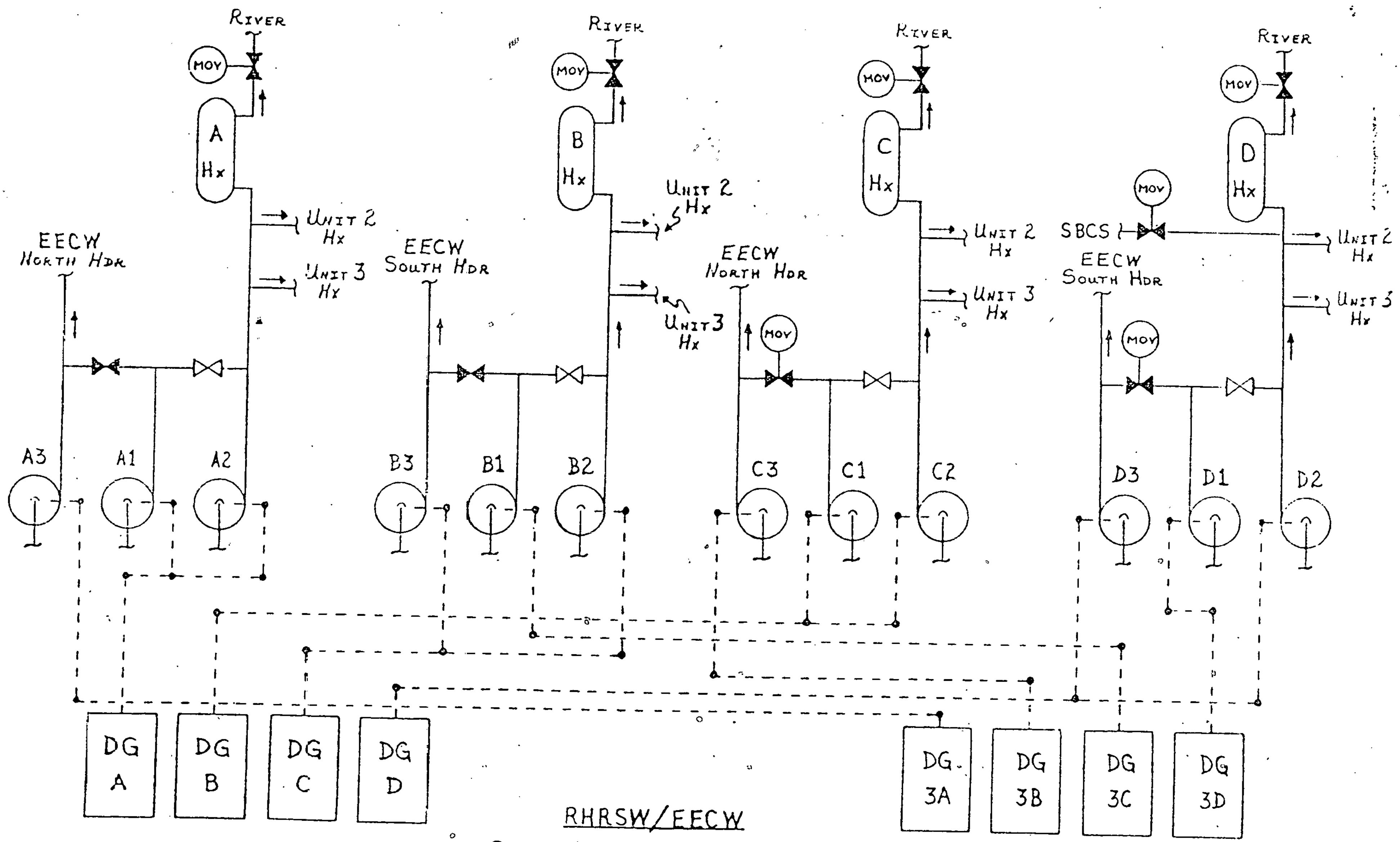


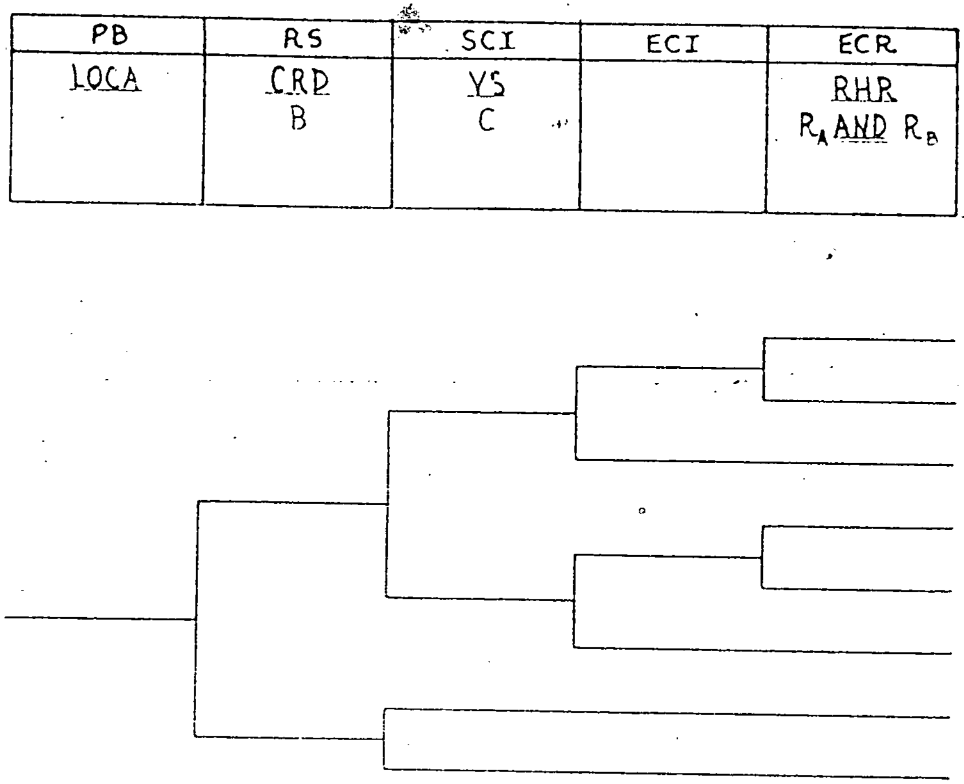
Figure 3 *System Normal Operation*



RHRSW/EECW

SYSTEM POWER DEPENDENCIES

Figure 4



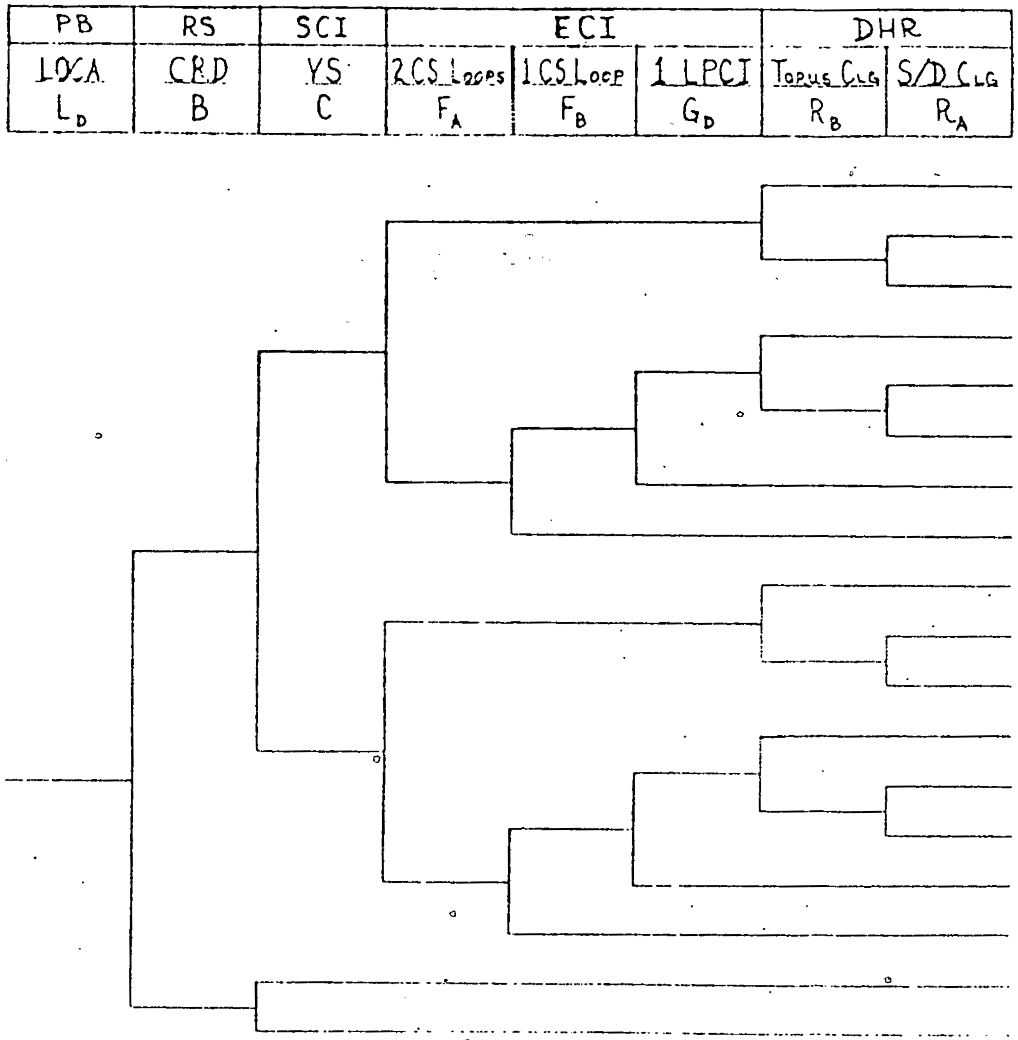
Sequence Probability	Seq. No.	RS	SCI	ECI	ECR	Remarks
	1					Core cooled
	2				X	Slow melt
	3			X	N/A	Melt
	4		X			Core cooled
	5		X		X	Slow melt
	6		X	X	N/A	Melt
	7	X		N/A	N/A	Melt
	8	X	X	N/A	N/A	Melt

X = Function failure

Figure 5

BROWNS FERRY UNIT 1 LOCA SYSTEMIC EVENT TREE - L_D

[LARGE (DISCHARGE-SIDE) LIQUID BREAK] [BREAK SIZE (ft^2): 0.3 TO 4.3]



SEQ. NO.	SEQUENCE FREQUENCY	RS	SCI	ECI	DHR	REMARKS
1						CORE COOLED
2						CORE COOLED
3					X	SLOW MELT
4						CORE COOLED
5						CORE COOLED
6					X	SLOW MELT
7				X	N/A	MELT
8				X	N/A	MELT
9			X			CORE COOLED
10			X			CORE COOLED
11			X		X	SLOW MELT
12			X			CORE COOLED
13			X			CORE COOLED
14			X		X	SLOW MELT
15			X	X	N/A	MELT
16			X	X	N/A	MELT
17		X		N/A	N/A	MELT
18		X	X	N/A	N/A	MELT

X = FUNCTION FAILURE

Figure 6

A front-line system is a system whose function is necessary to successfully mitigate the effects of a loss-of-coolant accident or operational transient at Browns Ferry Unit 1. A support system is a system that affects the course of an accident or transient only by way of its effect on the operation of a front-line system. Fault trees were constructed using an abbreviated fault tree approach[4] for each front-line system appearing in the systemic event trees, with the exception of the reactor protection system (RPS). The support systems required for front-line system success were also analyzed using the fault tree methodology.

SEQUENCE QUANTIFICATION

The Reliability Analysis System (RAS) computer code[5] calculated the unavailabilities for the front-line and support system models. Due to limitations in computer core space and processing time, sequence frequencies were manually calculated using the system unavailabilities calculated by the RAS code. The Boolean combination of system successes and failures was derived and the system unavailabilities from the RAS code were substituted into the derived expression. The Common Cause Analysis (COMCAN) computer code[6] was used to identify the commonalities between systems. Any commonalities identified were evaluated manually or by using the RAS code and were included where appropriate. In some cases, bounding analyses were performed to determine if additional partial dependencies in the sequences could be significant where direct quantification was not feasible. COMCAN also evaluated commonalities between systems in order to account for complement or success paths in each sequence.

The effect of LOCA initiators on the mitigating systems was also considered. For large LOCAs this was accomplished by generating distinct event trees. For other LOCAs the effect was accounted for during sequence quantification by considering the probability of the break occurring in a location which renders LOCA mitigation systems partially or completely inoperable.

Employing the techniques briefly discussed above led to the identification of candidate dominant accident sequences. These potential dominant sequences were then re-examined in view of the possible recovery actions which could influence the course or likelihood of these sequences. Consideration of these effects resulted in the dominant accident sequences that contribute to public health and safety risks. The approximate timing and magnitude of atmospheric releases associated with these sequences were qualitatively classified into release categories similar to those made in prior risk assessments of comparable plants where formal release category analysis was performed.

INSIGHTS

In general, the failures leading to core melt can be classified into three functional categories; failure to remove long-term decay heat, failure to keep the core covered, and failure to achieve subcriticality (ATWS events). Preliminary quantification indicates that transients contribute to core melt sequences with higher frequencies than LOCA sequences. Furthermore, those sequences with failure of the long-term decay heat removal function have the highest frequencies followed by sequences with the failure to achieve subcriticality function and sequences with failure to keep the core covered function.

With regard to LOCA sequences leading to core melt, the loss of offsite power induced SORV and transient-induced SORV sequences have the highest frequency. The frequencies of the remaining LOCA sequences that lead to core melt are approximately equal and are several orders of magnitude less than the frequencies of the transient-induced LOCA sequences.

The loss of offsite power (LOSP) transient is interesting due to the dependency of all three units on one common support system during this transient. The Emergency Equipment Cooling Water (EECW) System is a shared system among the units. Under LOSP conditions, once started, all the diesel generators must receive cooling water from this system in order to continue to run. Failure of this system would then result in a loss of cooling water to the diesels and ultimately result in the loss of all AC power at all three units. Fortunately, there are DC powered systems that can operate under these conditions and keep the reactors cool for three to four hours before AC power must be restored. However, it seems that such a dependency is neither necessary nor desirable and adds considerably to the frequency of the core melt sequences for the LOSP initiator.

REFERENCES

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5. N. H. Marshall et al., "User's Guide for the Reliability Analysis System (RAS)", TREE-1168, (1977).

6. N. H. Marshall et al., "COMCAN II: A Computer Program For Common Cause Failure Analysis," TREE-1289, (1978).