ENCLOSURE 1

LETTER REPORT ON THE REVIEW OF THE SEQUENCES FOLLOWING A RELEASE OF EXCESSIVE WATER IN ELEVATION 8 OF THE REACTOR BUILDING IN THE SHOREHAM NUCLEAR POWER STATION

K. Shiu
Y. Sun
E. Anavim
I. A. Papazoglou

Risk Evaluation Group
Department of Nuclear Energy
Brookhaven National Laboratory
Upton, New York 11973

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ABSTRACT

The core vulnerable risk resulted from Reactor Building flooding events is addressed as a part of the SNPS PRA.[1] The analysis was reviewed and re-evaluated at BNL and the results are presented in this report. The BNL review includes both qualitative and quantitative analyses of flooding initiators, operator errors, and accident sequences which result in a vulnerable core state. An estimate of the uncertainty for the core vulnerable risk is also included.

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1.0 INTRODUCTION

At the Shoreham Nuclear Power Station (SNPS) the majority of safety-related equipment are located in the Reactor Building (RB). The Shoreham Reactor Building is a cylindrical building surrounding the MARK II containment structure. Water leakage from equipment in the reactor building will drain to Elevation 8 (the lowest level of the RB) via openings and stairwells since there is no structural separation between safety systems. Flooding of the Elevation 8 compartment may potentially disable all the ECCS because they are located in the Elevation 8 compartment.

The SNPS-PRA(1) has included flooding as a common-mode event which may disable the ECCS equipment. The SNPS PRA assumes that a critical flooding depth of 3'-10" from the RB floor will disable all the ECCS equipment. Operator diagnosis and isolation of the flooding before it reaches 3'-10" depth is considered in SNPS-PRA.

Because of the potentially significant impact, the SNPS's evaluation of the core melt risk due to RB flooding warrants a special review. A field trip to the Shoreham plant has been made by BNL personnel for obtaining detailed information on the equipment and power control layouts in the RB, especially in the Elevation 8 compartment. BNL has determined that there are three flooding depths (1'-3", 1'-10", and 3'-10") that are critical to the availability of various ECCS equipment. The initiator event trees are thus revised accordingly.

BNL also identified that the random failure of a equipment protection circuit breaker coinsiding with the RB flood event may cause the propagation of failures to equipment powered by separated Motor Control Centers (MCC). This potential common mode failure event has also been modeled in BNL event trees.

Shoreham Plant Procedure Guides relevant to the RB flooding have been reviewed by BNL. BNL found that these procedure guides fail to require a systematic check of system parameter indicators in the control room following a RB Flooding Alarm annunciation. This may cause the operator to ignore an abnormal system parameter, especially under a multiple alarm situation (such as a turbine trip).

BNL's revised event trees, quantitative evaluation of core vulnerable risk due to RB flooding events, and an uncertainty estimate for the core vulnerable risk are presented in this report.

The report is organized as follows: Section 2 summarizes the SNPS-PRA approach to the flood sequence identifications and quantification. Section 3 presents the BNL revision both in the methodology and in the quantification. Finally, Section 4.0 summarizes the results.

2.0 SNPS METHODOLOGY AND ANALYSIS

2.1 Overview

The SNPS methodology for determining the contribution to the risk of the internal floods can be divided into three steps.

1. Identification of water sources and pathways to Elevation 8 compartment.
2. Evaluation of operators responses and assessment of likelihood of arresting the flood.
3. Evaluation of system responses and identification of the sequences leading to a core vulnerable state given a flood.

In the Shoreham PRA approach it was determined that flooding at locations other than Elevation 8 would be bounded by the analysis of flooding at the lowest level of the reactor building Elevation 8, since the flood water will drain and cascade down to that level through stairwells and openings. All the evaluations of flood are hence focused on equipment at the Elevation 8 level.

The volume of water required to flood the reactor building Elevation 8 compartment, with all equipment and piping installed, is estimated to be 41,600 gallons in SNPS-PRA for each foot of depth. The following drainage systems are available to receive the initial volume of flood water:

- Reactor Building Floor Sumps
- Reactor Building Equipment Sumps
- Reactor Building Porous Concrete Sumps.

These systems have total sump capacity of 4,650 gallons, and total sump pump capacity of 640 gallons per minute, however, they are not included in the analysis.

The potential water sources which may release excessive water in Elevation 8 are summarized in Table 2.1.1. For each of these sources, a pathway investigation has been performed in the SNPS-PRA, to define the potential for flood at Elevation 8. Table 2.1.2 summarizes the water sources as evaluated in the Shoreham PRA. For each water source the largest possible flow rate has been determined and the time required for the flood to reach the 3'-10" level in Elevation 8, have been estimated. These times are also given in Table 2.1.2. These times provide the basis for estimating the probability of successful prevention of flood at the 3'-10" level by operator actions.

A survey of all vital equipment by Shoreham identified a number of components for the various accident mitigation systems which could potentially be submerged in the event of an internal flood. Based on this information, the critical height of 3'-10" was defined. It was assumed that if flood water exceeds the 3'-10" level, all ECCS equipment would be disabled. Flooding scenarios which are arrested before reaching the 3'-10" level, have been found to contribute negligibly in the core damage frequency.

Functional event trees were used in the Shoreham internal flood PRA to model the plant response given an internal flood initiator. The flood initiator frequency was calculated based on two types of internal flood precursors: online maintenance and rupture of piping, valves or pumps. These precursor frequencies are described in Section 2.2. Given the occurrence of these flood precursors, the progression of events was modeled using initiator event trees. Details of the initiator event trees are presented in Section 2.3.

Since all the ECCS systems are assumed lost given a 3'-10" flood, the only available means for cooling the core are the feedwater and the condensate pump injection. The availability of these two systems depends on the state of the MSIVs and on the ultimate source of the flood (condensate storage tank or suppression pool).

Because of these dependences, the end states of the initiator event trees were classified into six categories each of which becomes the entry condition for the functional event trees. Table 2.1.3 summarizes the information in a matrix form. Each row of the matrix depicts one of the 17 types of internal flood precursors, the columns represent the six entry conditions to the functional event trees. The six entry conditions can be grouped into manual shutdown, turbine trip and MSIV closure. Two possible entry conditions are considered for each of these three initiators: flooding due to water from the condensate storage tank (CST) and flooding due to water from other sources.

Based on these six entry conditions, six functional event trees were developed. An example is given in Figure 2.1.1.

2.2 SNPS-PRA Quantification of the Frequency of Flood Initiators

Two types of flood initiators were considered in the SNPS-PRA.

1. Floods initiated by an accidental loss of isolation (valve opening) while a component in the Elevation 8 area is dismantled for maintenance.

2. Floods initiated by a rupture in the pressurized or the non-pressurized part of the piping.

2.2.1 Maintenance-induced Flood Initiators

The frequency of the first type of initiator was calculated by estimating the frequency of maintenance of various components based on operating experience data. The LER data base in Ref.2 identifies the observed failures from turbine-driven and motor-driven pump failures. The data used in the SNPS-PRA are summarized in Table 2.2.1. There are four failure modes for pumps, i.e., leakage/rupture, does not start, loss of function, and does not continue to run. The hourly LER failure rates characterize the leakage/rupture failure mode, while demand failure rates consider other failure modes.

The following LER rates are found for the four failure modes in motor-driven and turbine-driven standby pumps.

Motor Driven Pumps

- Leakage/rupture: 6 events/6,777,627 hrs. = 8.9×10^{-7}/hr.
- Does not start, loss of function, and does not continue to run: (5+4+6) events/(13,644 demands) = 1.1×10^{-3}/demand

SNPS-PRA assumed that these pumps are in standby status until there is a demand. The number of demand used in SNPS-PRA are 12 on the average per year (four scheduled tests plus eight other occurrences). Hence, the maintenance frequency for motor driven standby pumps per year is calculated as

$(8.9\text{x}10^{-7}$ failure/hr)*(24 hr/day)*(365 day/yr) +
$(1.1\text{x}10^{-3}$/demand)*(12 demands/yr) = $2.0\text{x}10^{-2}$ failure/year.

Turbine Driven Pump

Similiarly, the maintenance frequency for turbine driven standby pumps per year is calculated as 0.079 failure/year.

There are two motor driven pumps associated with the Core Spray System, four motor driven pumps with the LPCI System, and four motor driven pumps associated with the Service Water System in which two are linked as a pair to the RHR Heat Exchanger System. There is only one turbine driven pump associated with the HPCI System and one with the RCIC System. Table 2.2.2 summarizes the SNPS-PRA frequencies associated with major maintenance operations based upon the above evaluation and a conservative estimate of heat exchanger online maintenance.

2.2.2 Rupture-Induced Flood Initiators

The frequencies of the initiators caused by loss of system integrity from breaks or ruptures were derived from WASH-1400 failure rates of major components involving external leak and external ruptures, based on assumptions made in NUREG/CR-1363 (Reference 3). This information has been summarized in Table 2.2.3.

The calculation of each initiator is done by identifying the appropriate type and length of piping and number of components susceptible to rupture and summing the estimated yearly rupture rates. As an example, the total number of valves involved in the HPCI discharge system are 3 (2 MOV's and 1 Check Valve); there is no pump involved (Table 2.2.5) and the total length of piping is 76'. Referring to Table 2.2.3, the rupture failure rate for 100' of pipe section is $4.3\text{x}10^{-11}$/hr, and for external failure of a valve is

$1.3x10^{-9}$/hr. The total length of pipe in the HPCI Discharge System is estimated to be 76' (Table 2.2.5).

$$(3 \text{ valves})*(1.3x10^{-9}/hr) + 76'/100' \ (4.3x10^{-11}/hr)$$
$$= 3.9x10^{-9}/hr \text{ or } 3.5x10^{-5}/yr.$$

Since the flow rates through suction line breaks are time dependent (i.e., a function of the varying water head in the source) and a strong function of the break shape and size, a simplified model based on historical experience and engineering judgement is used in the Shoreham PRA to describe the conditional probability of break size. Table 2.2.4 summarizes the classes of break size examined.

These probabilities, are combined with the frequencies estimated for initiators associated with core spray, HPCI, RCIC, LPCI, and Service Water Rupture/Leak Suction System failure to obtain the initiating event frequencies for non-pressurized piping. Table 2.2.6 summarizes the frequencies of initiators due to the loss of system integrity from breaks or ruptures.

2.3 Initiator Event Trees

The probability of causing a flood due to component under maintenance or the probability of not arresting the flood is calculated with the help of initiator Event Trees. These trees are shown in Figures 2.3.1 through 2.3.17. A discussion of the P, D, E, I, and A events in the event trees follows.

a. Event P - Operator removes power from equipment and valves. The removal of power from equipment and its isolation valves is a required procedure during a maintenance in both fossil and nuclear power stations. The equipment and isolation valves are electrically disconnected from their associated power supply by pulling and tagging the appropriate breaker at the MCC. A second qualified person verifies the correct implementation of the tagging order and placement of the clearance tags.

   A human error probability (HEP) of 0.01 is assigned for this operator action. This value is determined using the probability data given in NUREG/CR-1278[4] (p.20-23).

b. Event D - System not demanded.
During the maintenance process there is a possibility that the safety systems will be demanded because of a transient challenge. Isolation valves will automatically open if the operator has failed to remove power from the isolation valves (Event P).

c. Event E - Operator maintains isolation.
During on-line maintenance with the equipment disassembled, the isolation valves need to be maintained in closed position throughout the duration of the maintenance process. However, an operator error could inadvertently open isolation valves.

SNPS concludes that it is unlikely that the operator will manually open these valves locally in the RB and fail to notice the flood. Opening of the isolation valves at the MCC is also concluded by SNPS to be unlikely.

The remaining possibility is that the valve is opened from the control room (given Event P). The panel switch could be activated by three events. These events are: the operator mistakenly operates the switch; a command fault to the valve; or the operator inadvertently operates the switch. The probabilities for these events are 10^{-3}, 10^{-4}, and 10^{-2}, respectively.

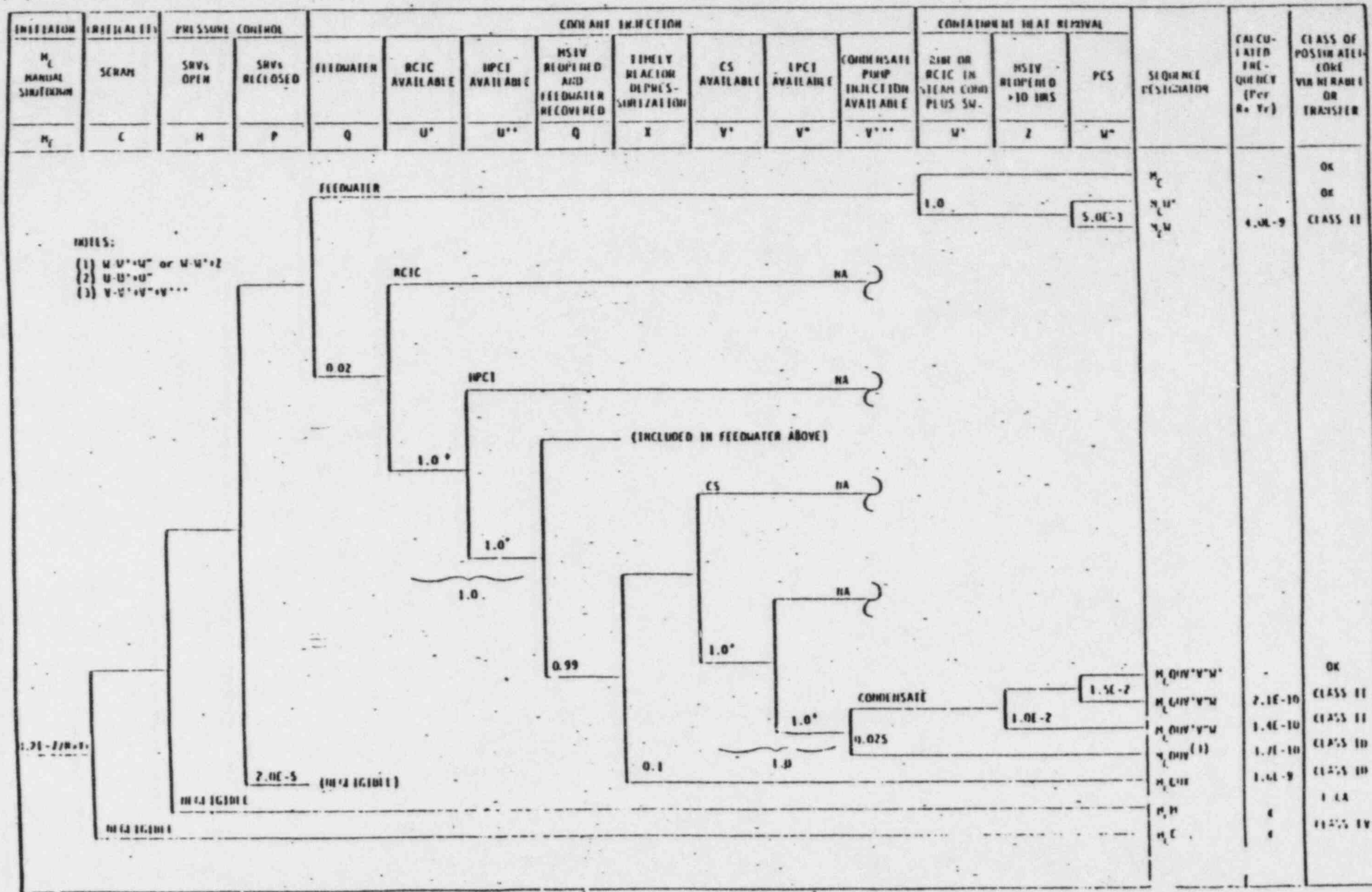
d. Event I - Flood annunciation.
The excessive water in reactor building is annunciated by alarms in the control room. The probability of the operator to fail to notice the alarm (the light is in a "back" panel) is assessed at 10^{-3}.

e. Event A - Operator diagnoses and responds to isolate the flood.
The operator must identify the source of and isolate the flood before it reaches the 3"-10" level. This event is considered by SNPS under two conditions as follows.

1. Operator isolates flood after auto occurrence, e.g., turbine trip or MSIV closure (Event A_A). Multiple alarms will occur in the control room at the same time as the flood alarm.

2. Operator isolates flood after manual occurrence, e.g., power operation or manual shutdown (Event A_M). Only the flood related alarms will annunciate in the control room.

The HEP data provided in NUREG/CR-1278[4] (1982 Edition, Chapter 12) are applied by SNPS for their evaluation. Figure 2.3.18 and Table 2.3.1 show the time varying cumulative HEP for both the single and the multiple occurrence conditions.



*In maintenance or flooded.

Figure 2.1.1 System Event Tree for Manual Shutdowns With Greater Than 3'-10" of Water in the Reactor Building (Source = CST).

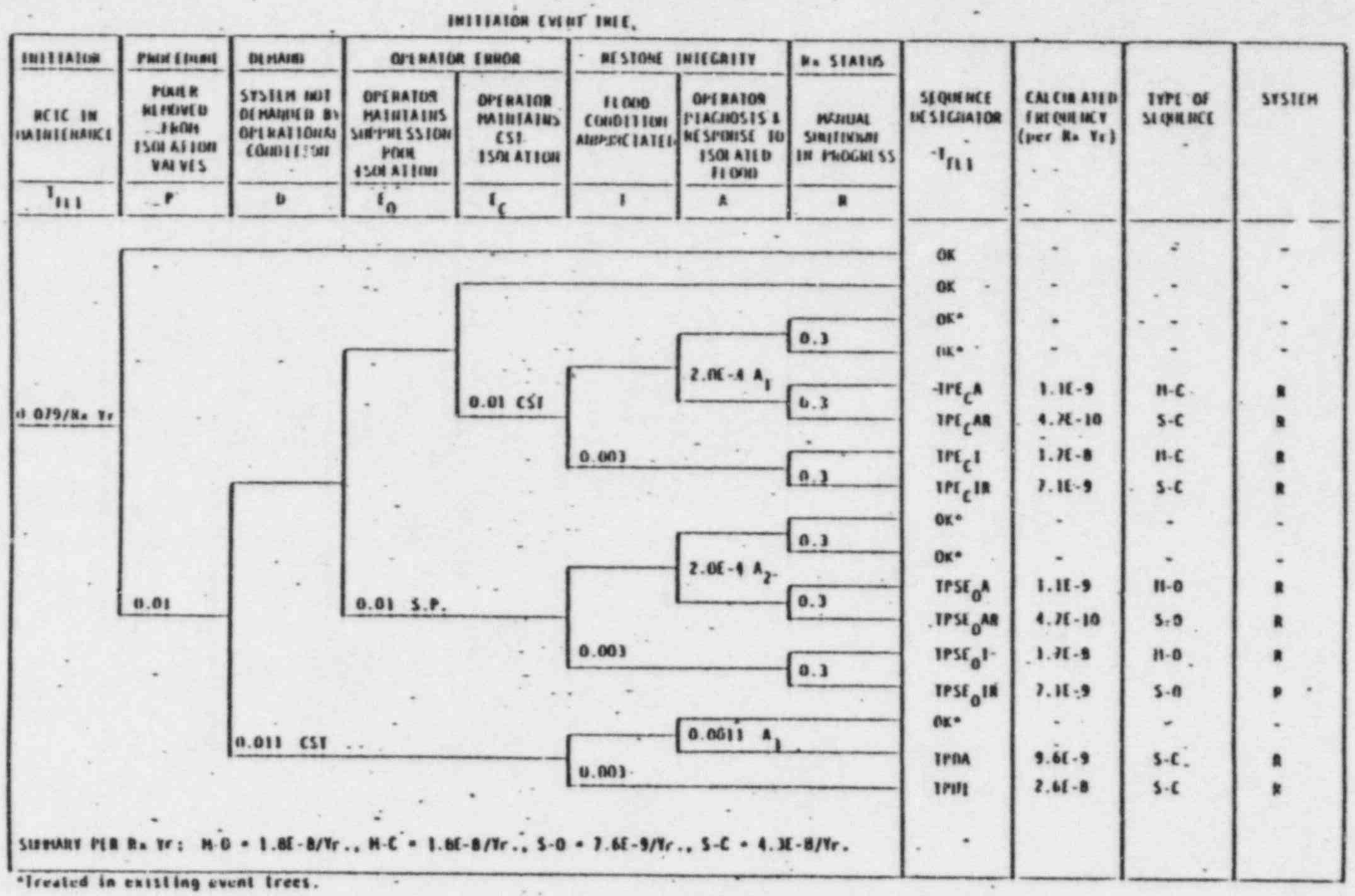


Figure 2.3.1 T_{FL1}: Initiator Event Tree for Postulated Flooding Sequences Initiated During RCIC Maintenance

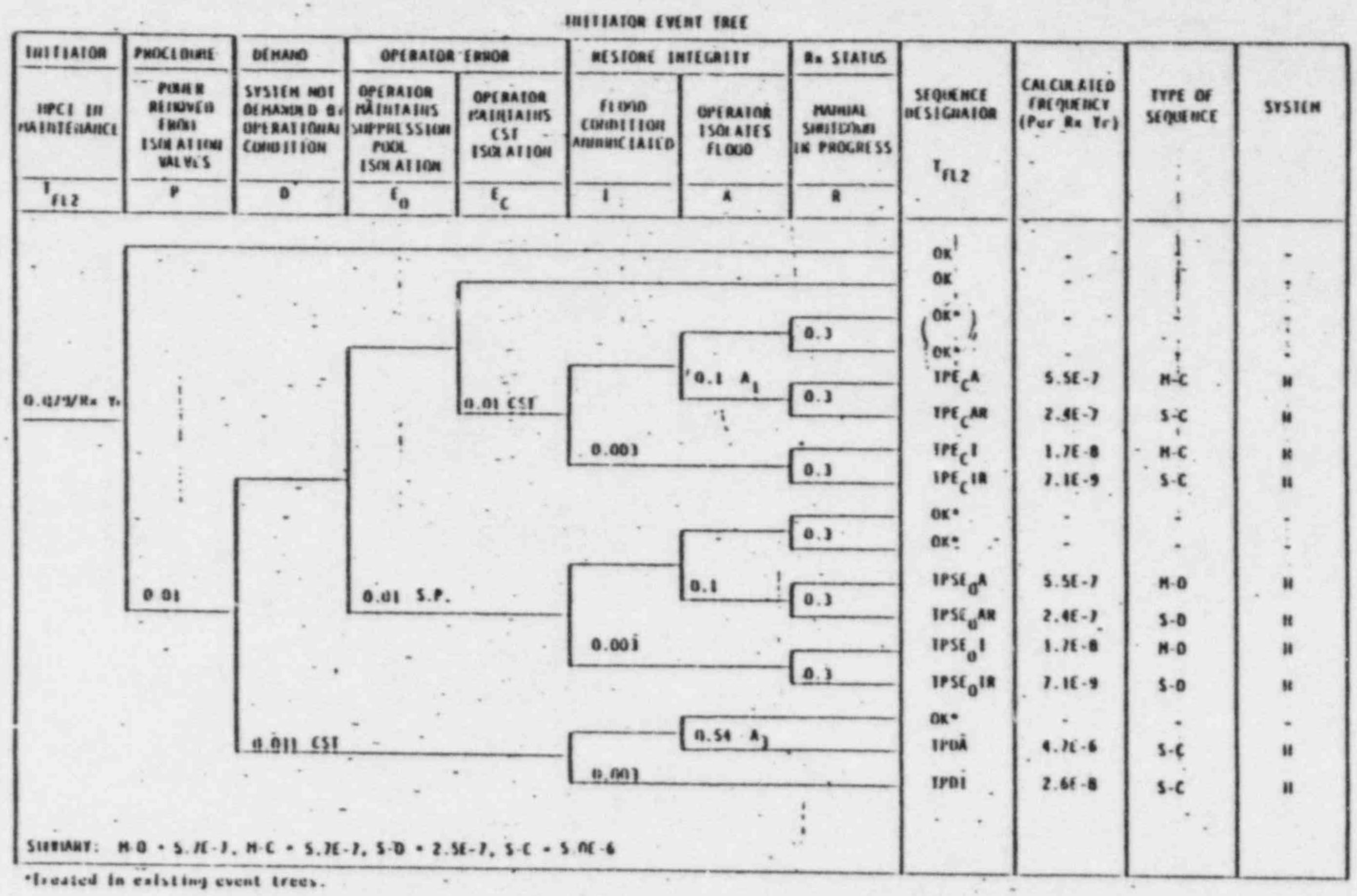
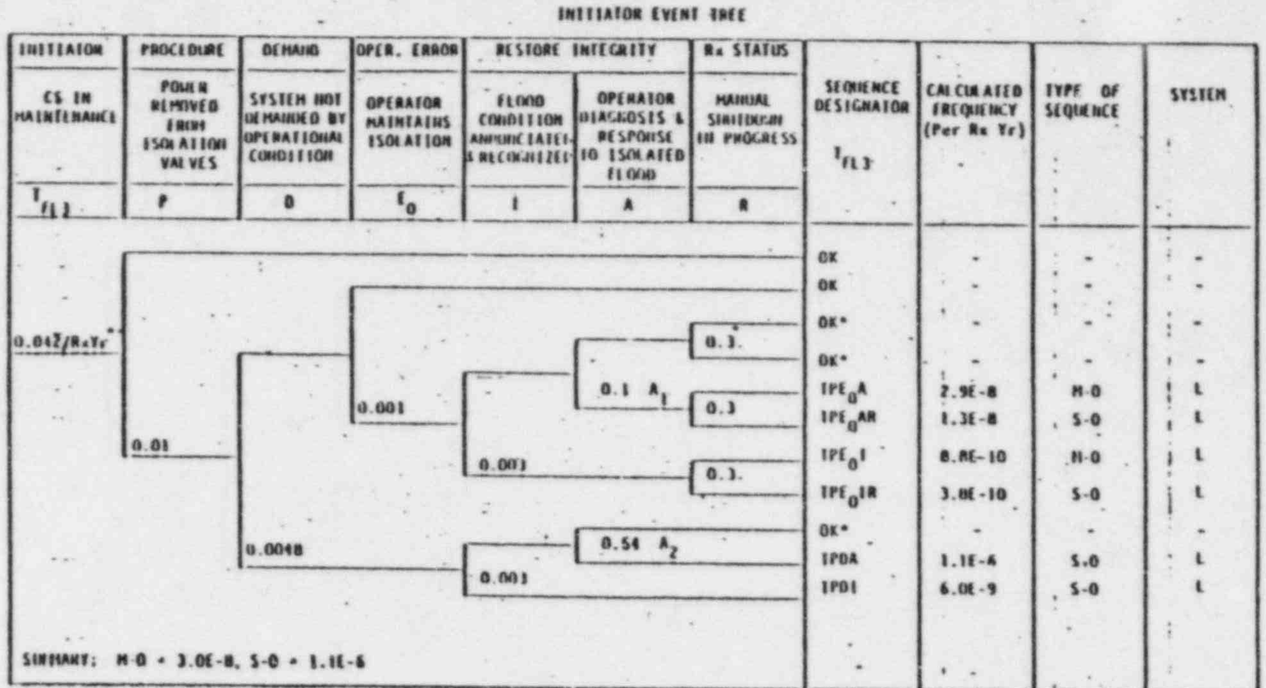
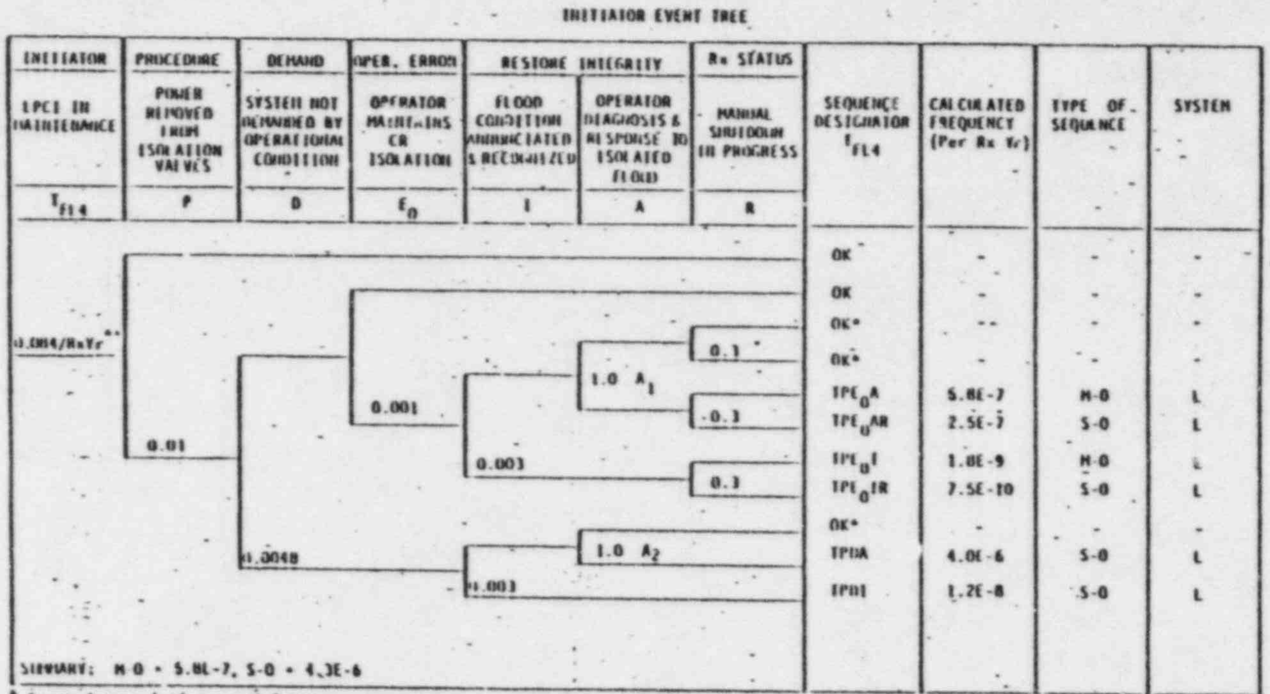


Figure 2.3.2. T_{FL2} - Initiator Event Tree for Postulated Flooding Sequences Initiated by an Error During HPCI Major Maintenance



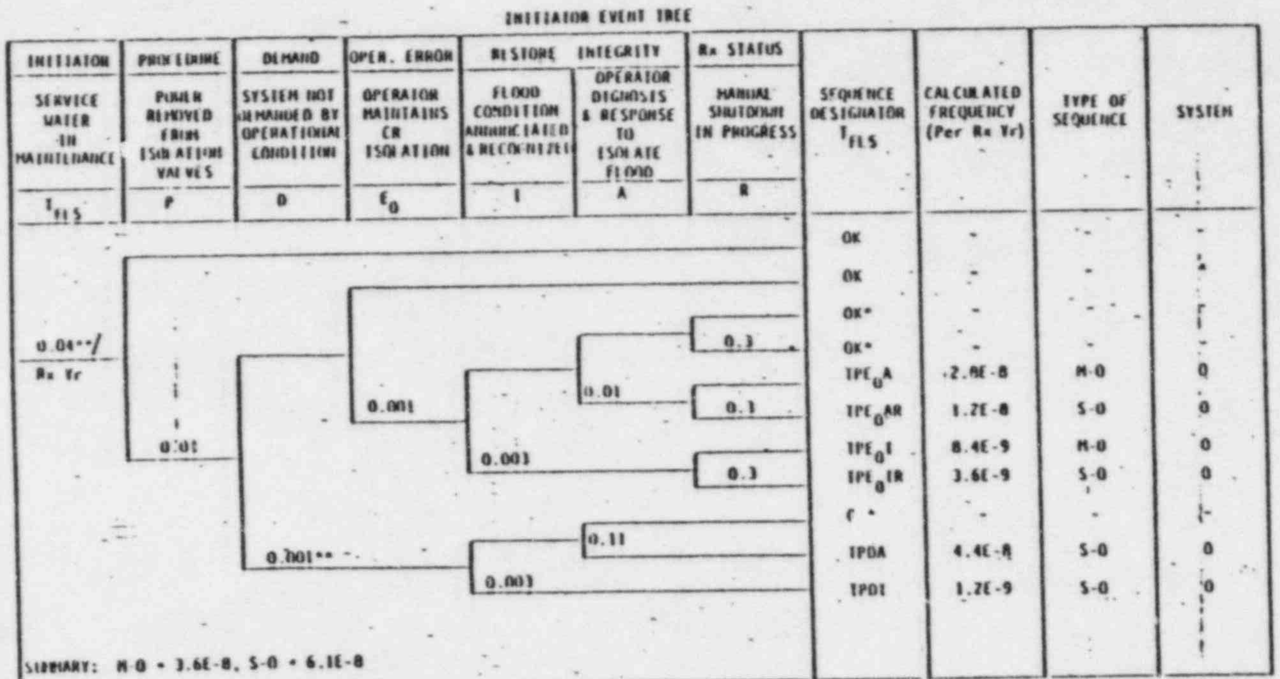
* Treated in existing event trees.
**Includes both CS pumps.

Figure 2.3.3 T_{FL3} - Initiator Event Tree for Postulated Flooding Sequences Initiated by an Error During Core Spray Major Maintenance



* Treated in existing event trees.
**Includes four LPCI pumps.

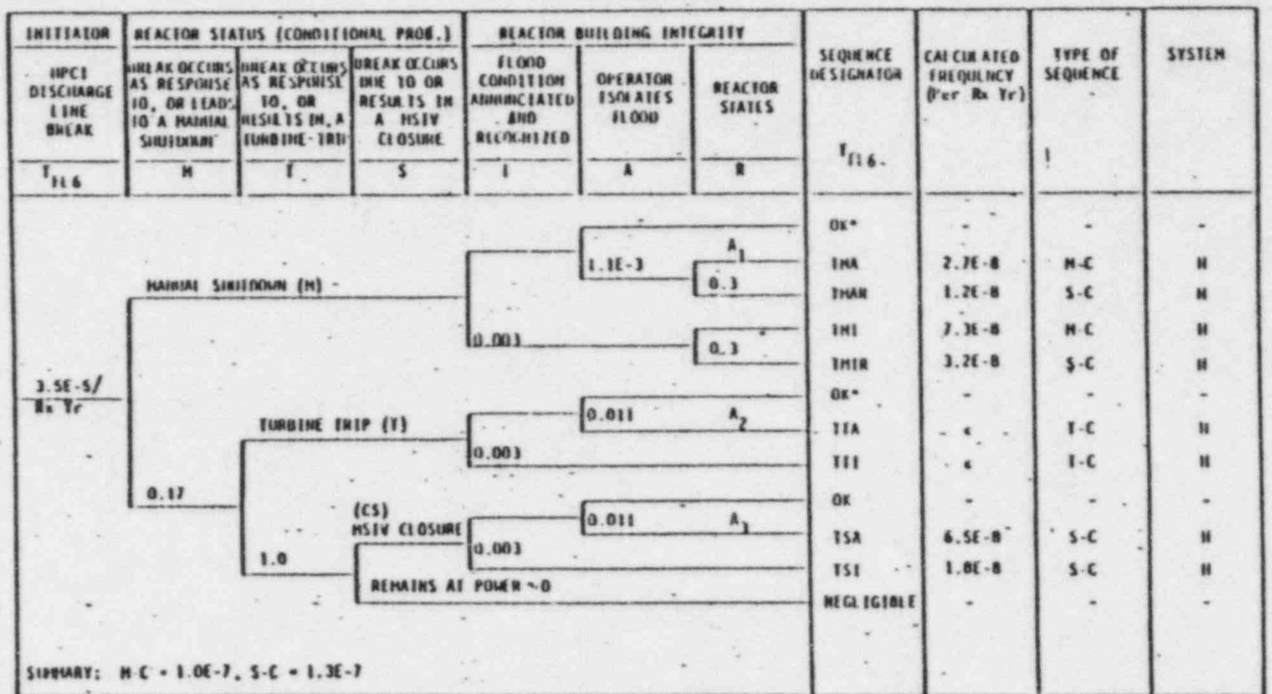
Figure 2.3.4 T_{FL4} - Initiator Event Tree for Postulated Flooding Sequences Initiated by an Error During LPCI Major Maintenance



* Included in existing event trees.
**Included both service water loops.

Figure 2.3.5 T_{FL5} - Initiator Event Tree for Postulated Flooding Sequences Initiated by an Error During Service Water Major Maintenance (i.e., Heat Exchangers)

INITIATOR EVENT TREE



*Included in the previously evaluated event tree.

Figure 2.3.6 Initiator Event Tree for Postulated Flooding Sequences Initiated by a HPCI Discharge Pipe Break

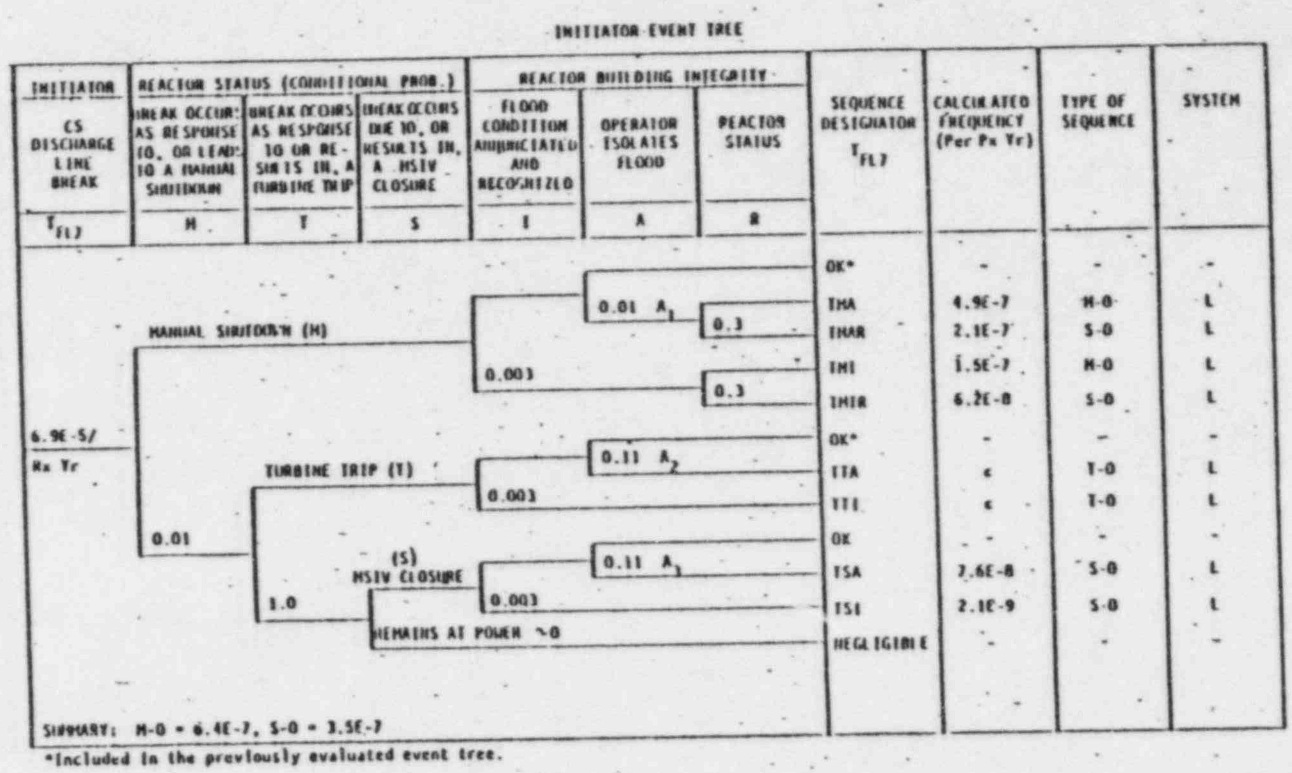


Figure 2.3.7 Initiator Event Tree for Postulated Flooding Sequences Initiated by a CS Discharge Pipe Break

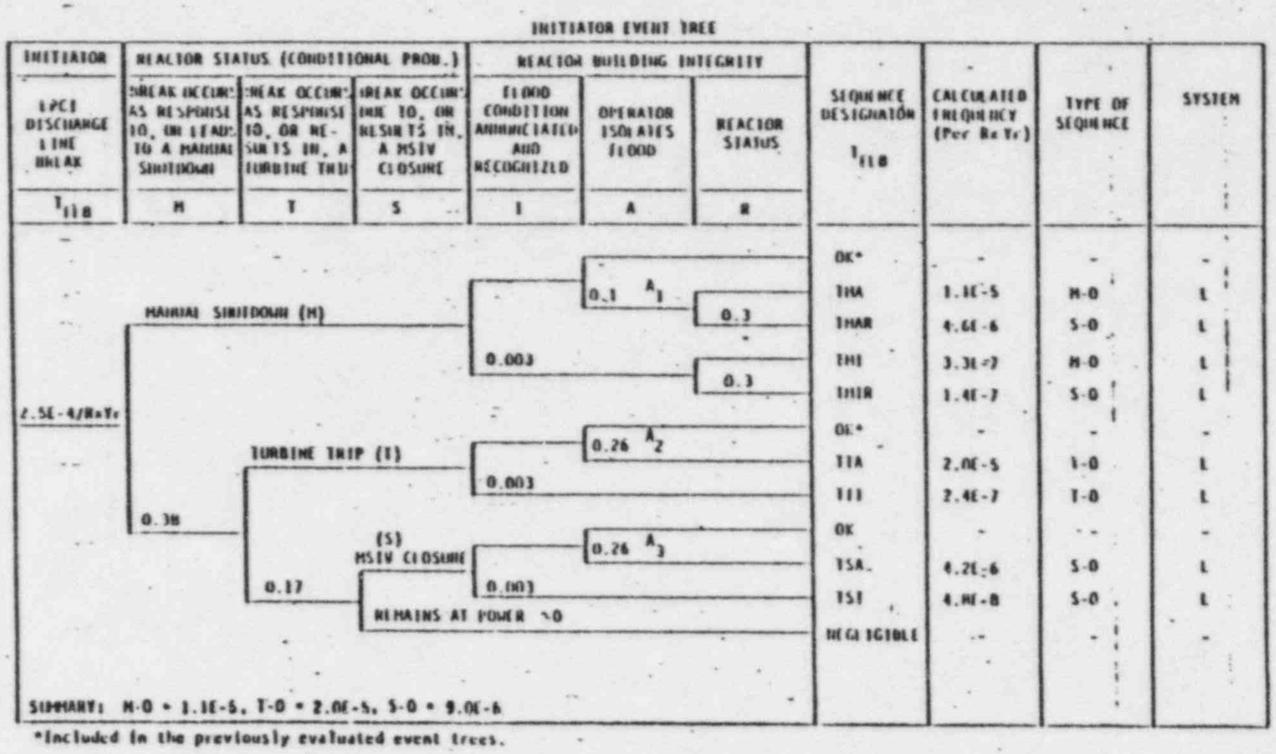


Figure 2.3.8 Initiator Event Trees for Postulated Flooding Sequences Initiated by a LPCI Discharge Pipe Break

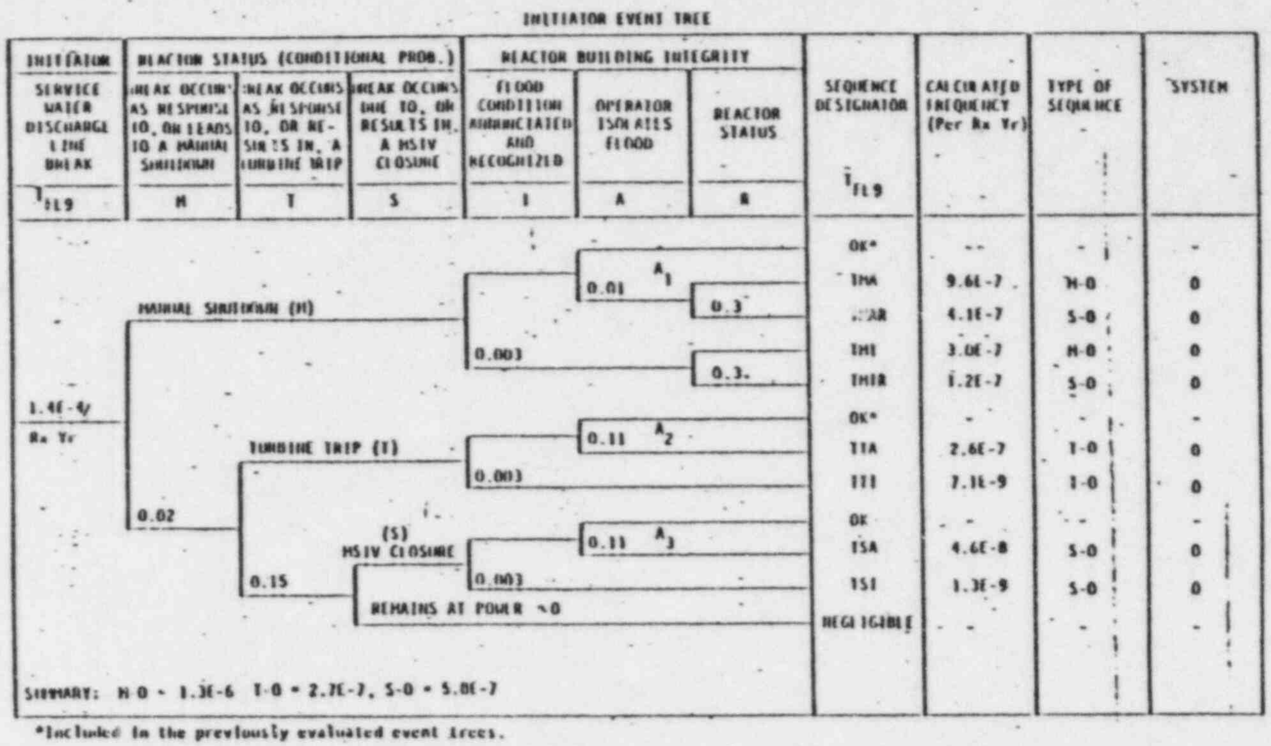


Figure 2.3.9 Initiator Event Tree for Postulated Flooding Sequences Initiated by a Service Water Line Break

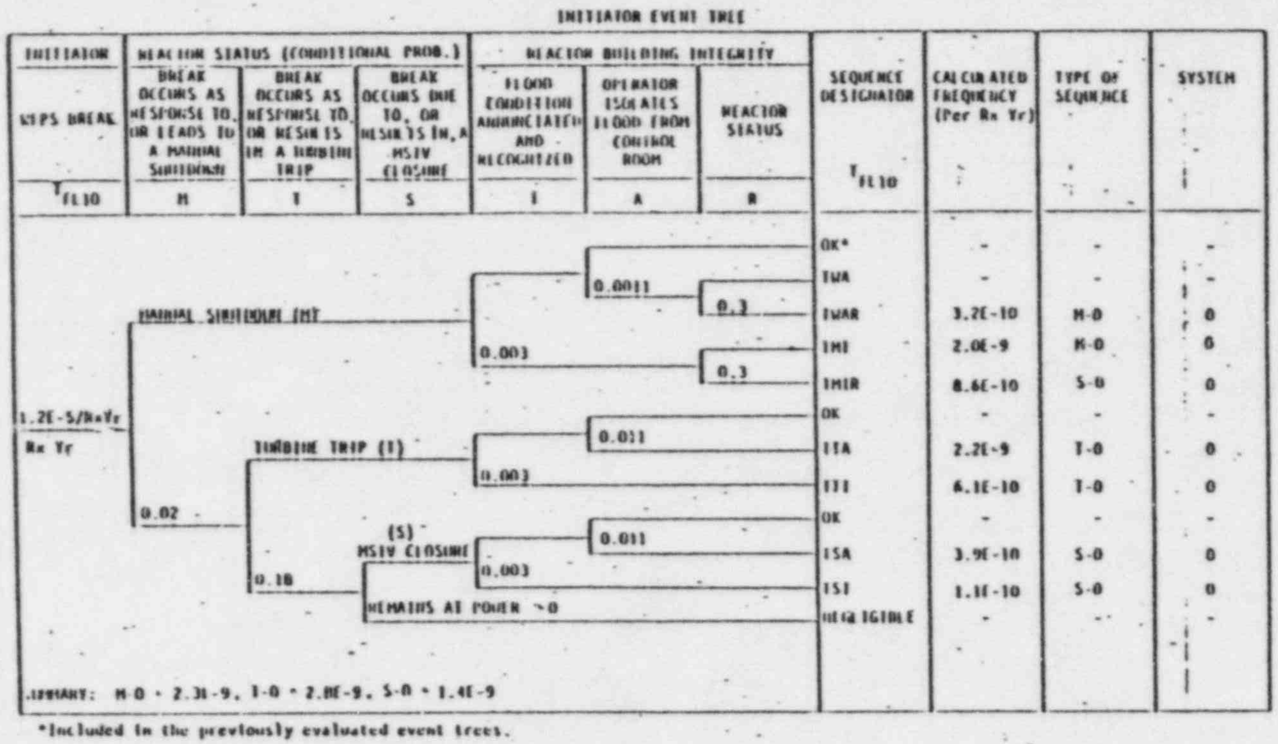
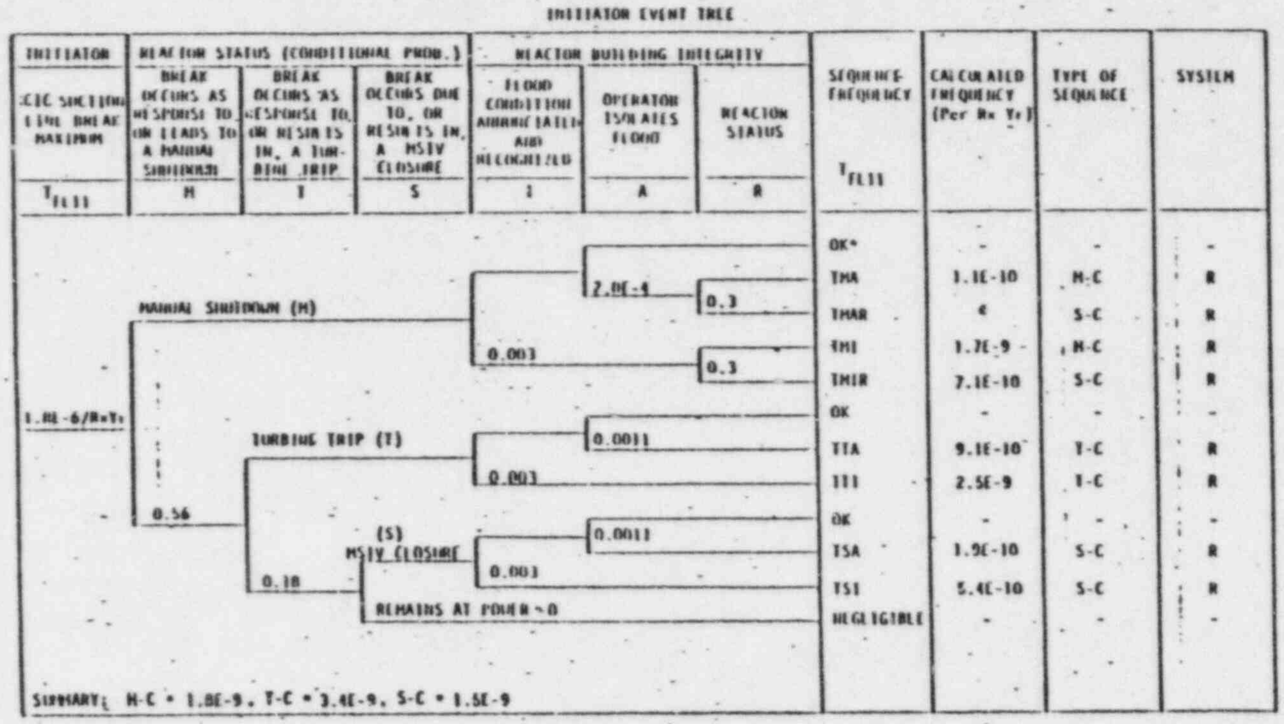
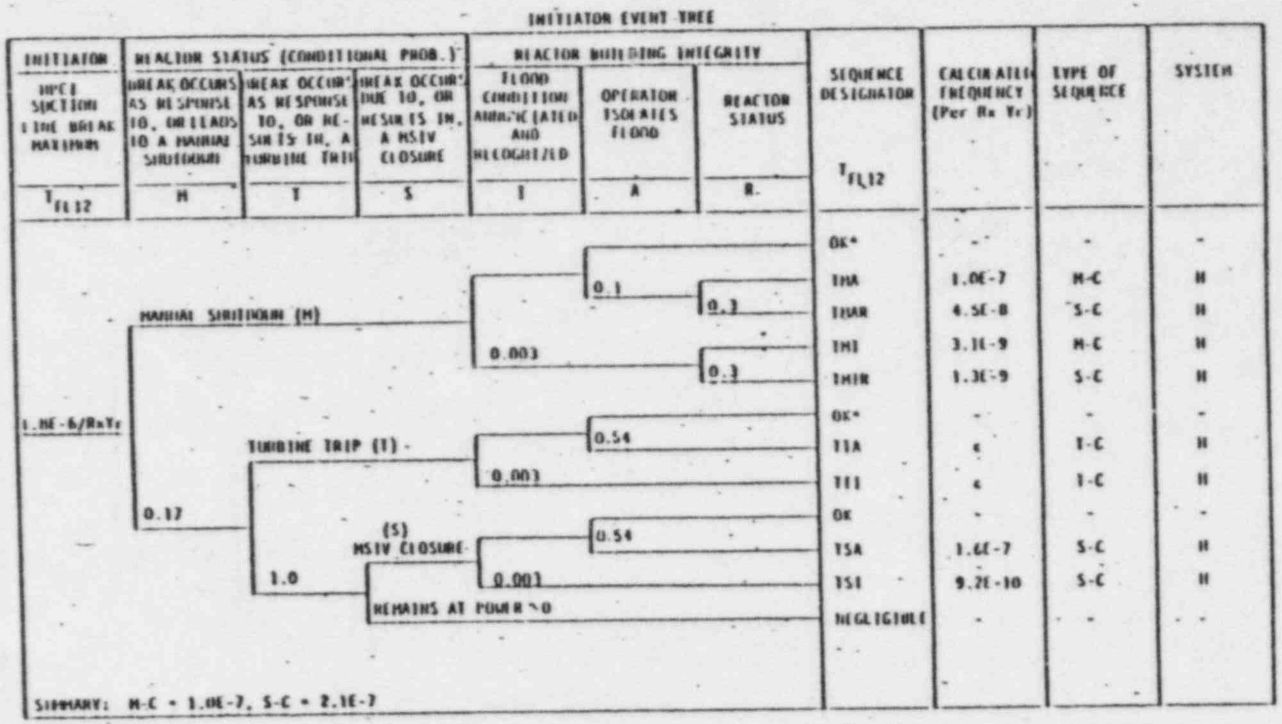


Figure 2.3.10 Initiator Event Tree for Postulated Flooding Sequences Initiated by a WFPS Break.



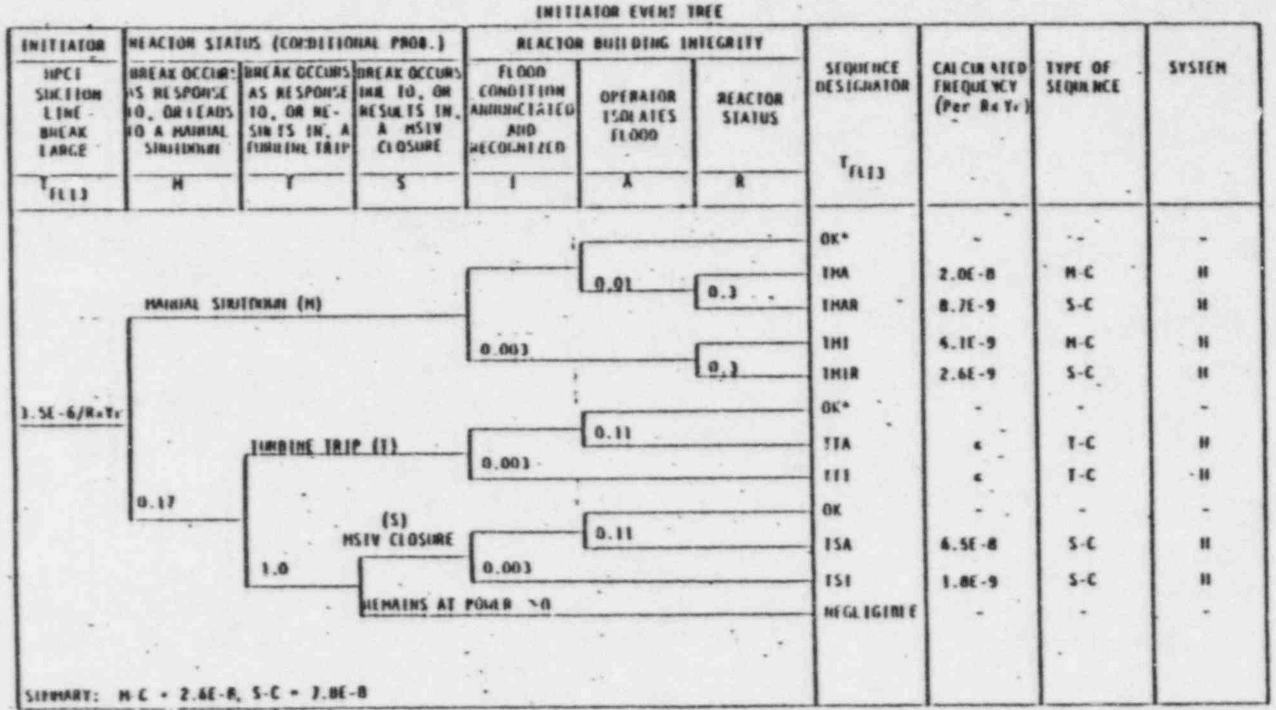
*Included in the previously evaluated event trees.

Figure 2.3.11 Initiator Event Tree for Postulated Flooding Sequences Initiated by a Maximum RCIC Suction Line Break



*Included in the previously evaluated event trees.

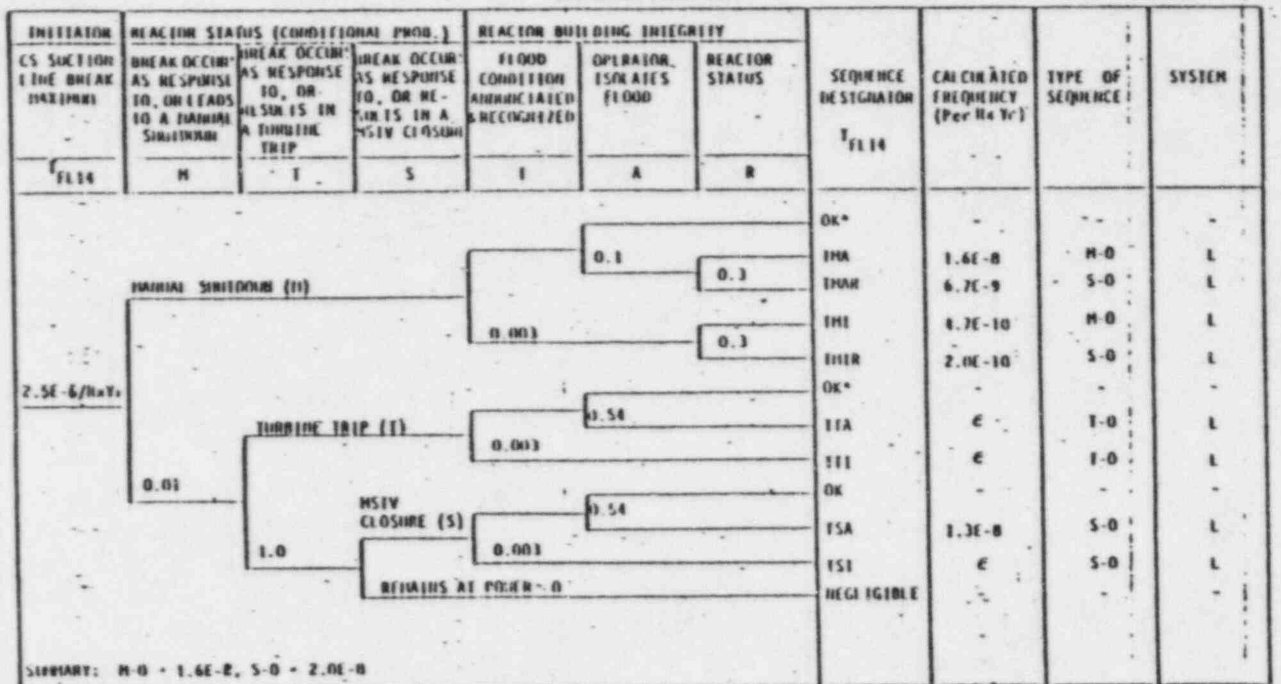
Figure 2.3.12 Initiator Event Tree for Postulated Flooding Sequences Initiated by a Maximum HPCI Suction Line Break



*Included in the previously evaluated event trees.

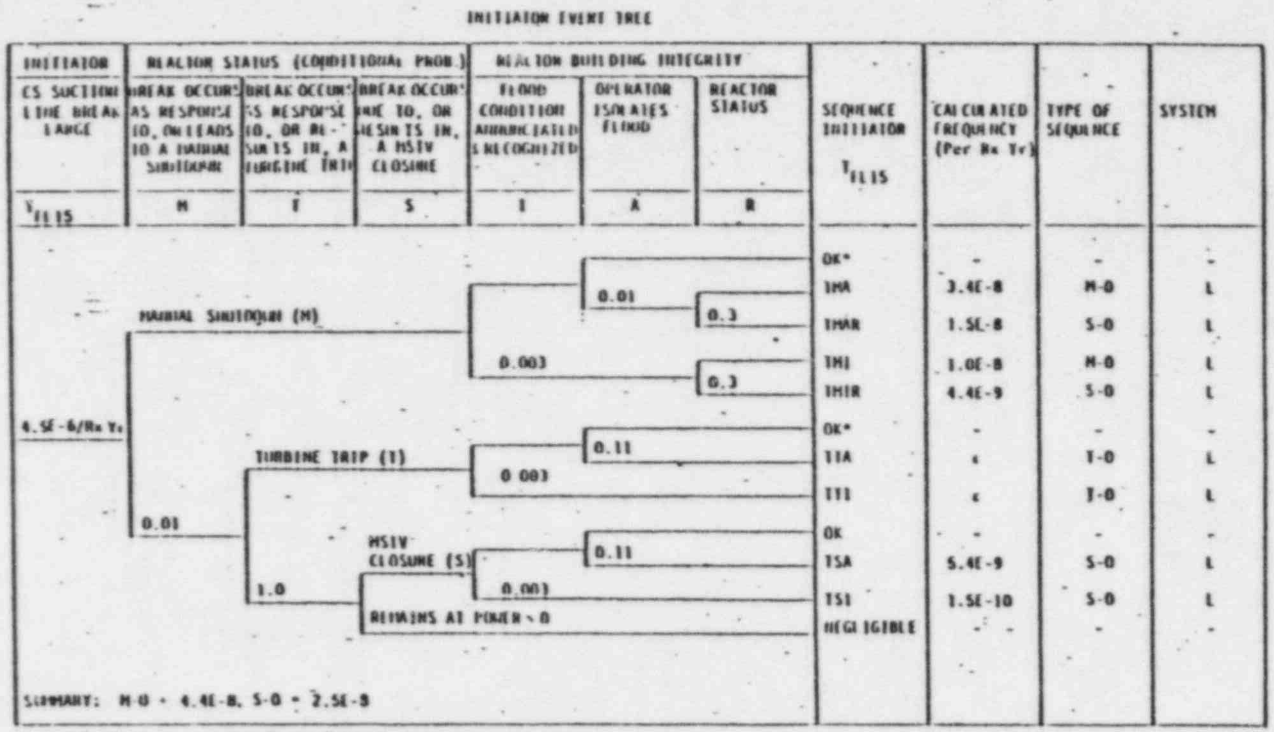
Figure 2.3.13 Initiator Event Tree for Postulated Flooding Sequences Initiated by a Large HPCI Suction Line Break

INITIATOR EVENT TREE



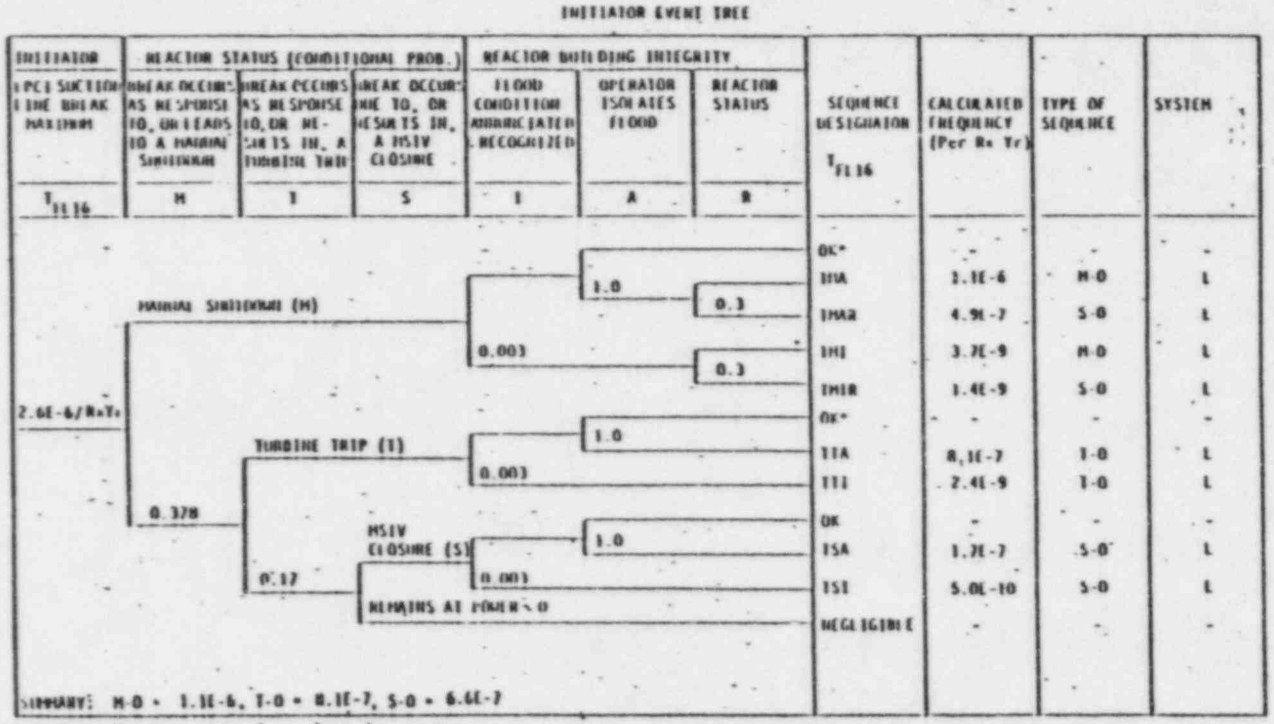
*Included in the previously evaluated event trees.

Figure 2.3.14 Initiator Event Tree for Postulated Flooding Sequences Initiated by a Maximum Core Spray Suction Line Break



*Included in the previously evaluated event trees.

Figure 2.3.15 Initiator Event Tree for Postulated Flooding Sequences Initiated by a Large Core Spray Suction Line Failure



*Included in the previously evaluated event trees.

Figure 2.3.16 Initiator Event Tree for Postulated Flooding Sequences Initiated by a Maximum LPCI Suction Line Break

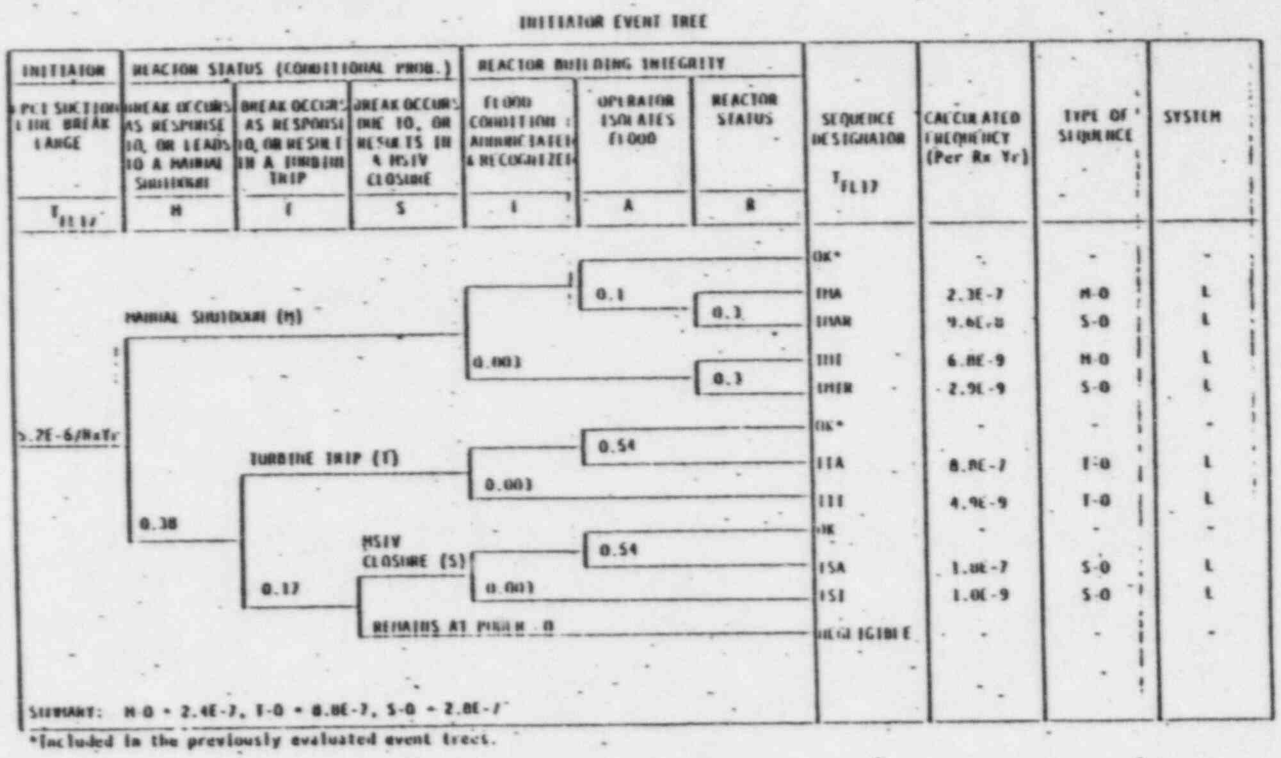


Figure 2.3.17 Initiator Event Tree for Postulated Flooding Sequences Initiated by a Large LPCI Suction Line Break

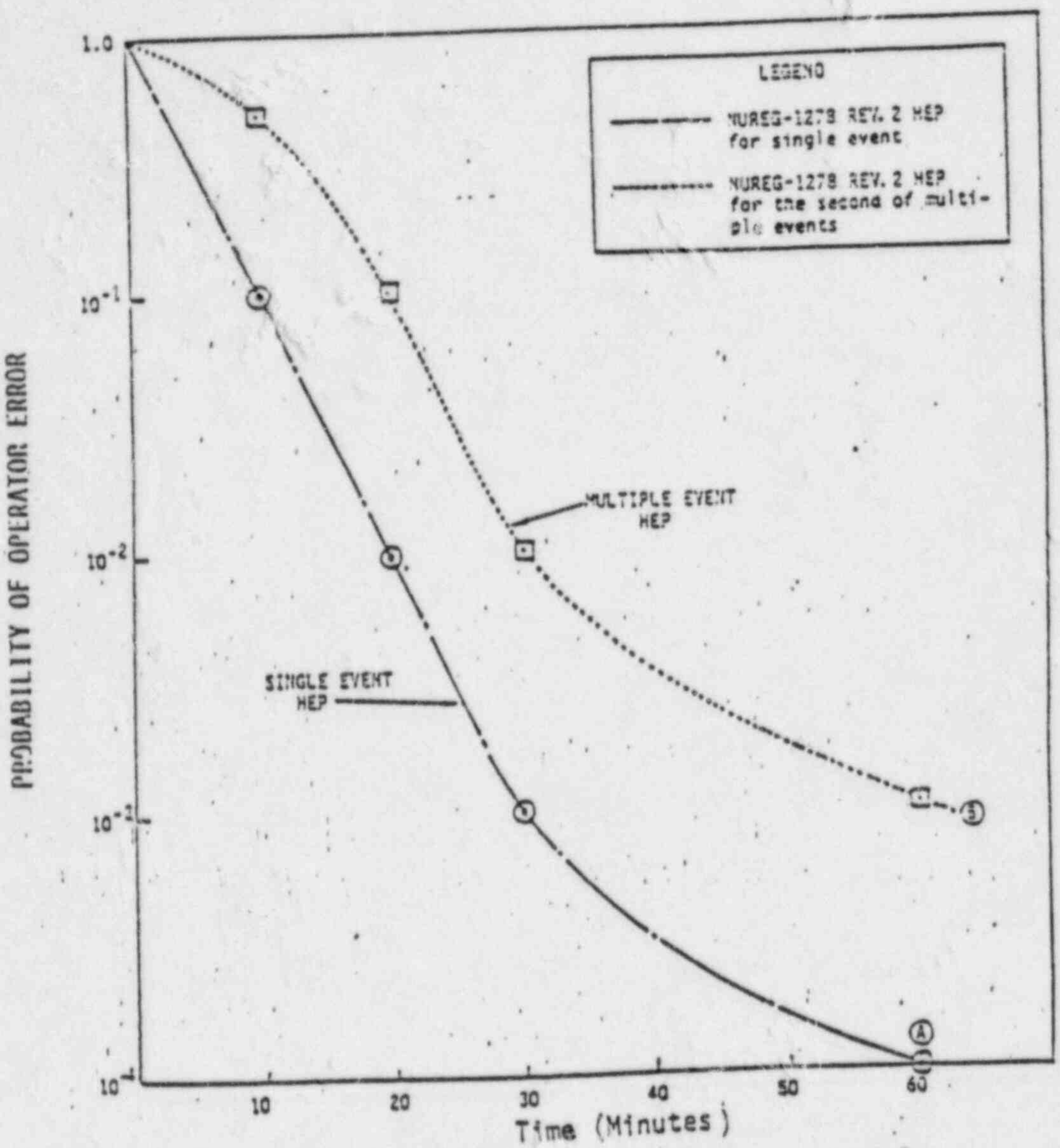


Figure 2.3.18 Comparison of the HEPs Associated with Operator Actions for Singular Events and Coincident Multiple Events

Table 2.1.1 Summary of Potential Water Sources and Types of Initiators Which may Lead to Release of Excessive Water in the Elevation 8 Compartment

Source	Quantity (Gallons)	No. of Lines	Systems Involved
Suppression Pool	160,000*	8	CS,LPCI,RCIC,HPCI
Condensate Storage Tank (CST)	550,000	4	CS,HPCI,RCIC
Reactor Primary System**	a) 42,928 b) 152,928		
Screenwell (Long Island Sound)	Unlimited	4	Service Water
Water Fire Protection System Storage Tank	600,000	Many	Fire Main

*Total water volume in the suppression pool at the high water level mark is 608,500 gallons. However, only a portion of the water can be drained through ECCS pump suction piping.

**Figure (a) includes water from the bottom of the core to normal water level in the RPV. Figure (b) includes (a) plus condenser hotwell water.

Table 2.1.2 Summary of Internal Flooding Initiator Types: Source, Pathway, Flowrates, and Time to Critical Flooding Depth

Source	Location	Flow Rate gpm*	Elevation 8 Flooding Time (Minutes*) 3'-10"
Suppression Pool	HPCI Pump Suction	9600	17.6
	RCIC Pump Suction	1500	10.6
	LPCI Pump Suction (Max/Large)**	17000/8500	9.4/19.0
	CS Pump Suction	13000	12.0
	LPCI Pump Suction	10500 (1 Pump Runout)	15.0
	CS Pump Discharge	6850 (1 Pump Runout)	23.0
Condensate Storage Tank (CST)	HPCI Pump Suction (Max/Large)**	1200/6000	13.0/27.0
	RCIC Pump Suction	2100	76.0
	CS Pump Suction (Max/Large)**	1200/6000	13.0/27.0
	HPCI Pump Discharge	4350 (Design)	37.0
Service Water	RHR Heat Exchanger	8000 (Pump Runout)	20.0
WFPS	Rupture of 8" Pipe	4000	40.0

*These flood times were calculated based on a failure of the sump pumps to successfully operate and a 41,600 gallon per foot depth in the reactor building given in the Shoreham FSAR.

**Large flow rates assumed to be 1/2 maximum flow.

Table 2.1.3 Summary of System Event Tree Entry States by Initiator Type

	SYSTEM EVENT TREE ENTRY CONDITION FREQUENCY (Per Rx Yr)					
INITIATOR	H-O	H-C	T-O	T-C	S-O	S-C
T_{FL1}	1.8×10^{-8}	1.8×10^{-8}			7.6×10^{-9}	4.3×10^{-8}
T_{FL2}	5.7×10^{-7}	5.7×10^{-7}			2.5×10^{-7}	5.0×10^{-6}
T_{FL3}	3.0×10^{-8}				1.1×10^{-6}	
T_{FL4}	5.0×10^{-7}				4.3×10^{-6}	
T_{FL5}	3.6×10^{-8}				6.1×10^{-8}	
T_{FL6}		1.0×10^{-7}				1.3×10^{-7}
T_{FL7}	6.4×10^{-7}				3.5×10^{-7}	
T_{FL8}	1.1×10^{-5}		2.0×10^{-5}		9.0×10^{-6}	
T_{FL9}	1.3×10^{-6}		2.7×10^{-7}		5.8×10^{-7}	
T_{FL10}	2.3×10^{-9}		2.8×10^{-9}		1.4×10^{-9}	
T_{FL11}		1.8×10^{-9}		3.4×10^{-9}		1.5×10^{-9}
T_{FL12}		1.0×10^{-7}				2.1×10^{-7}
T_{FL13}		2.6×10^{-8}				7.8×10^{-8}
T_{FL14}	1.6×10^{-8}				2.0×10^{-8}	
T_{FL15}	4.4×10^{-8}				2.5×10^{-8}	
T_{FL16}	1.1×10^{-6}		8.1×10^{-7}		6.6×10^{-7}	
T_{FL17}	2.4×10^{-7}		8.8×10^{-7}		2.8×10^{-7}	
TOTALS	1.6×10^{-5}	8.2×10^{-7}	2.2×10^{-5}	3.4×10^{-9}	1.7×10^{-5}	5.5×10^{-6}

Table 2.2.1 LER Data for BWR Standby Pumps for the Period of January 1972 Through April 1978

Standby Pumps	Demands	Standby Hours	Leakage Rupture	Does Not Start	Loss of Function	Does Not Continue To Run
Motor Driven	13,644	6,777,627	6	5	4	6
Turbine Driven	1,820	868,033	-	1	6	5

Table 2.2.2 Frequency of Online Major Maintenance System in the Reactor Building

System	Frequency (Per Year) SNSP-PRA	Initiator Event Tree
Core Spray (Motor Driven)	0.042	TFL3
LPCI (Motor Driven)	0.084	TFL4
HPCI (Turbine Driven)	0.079	TFL2
RCIC (Turbine Driven)	0.079	TFL1
Service Water (RHR or RBCLCW HX) (Motor Driven)	0.042	TFL5

Table 2.2.3 Summary of Failure Rates for Major Components Involving External Leak and External Rupture

Parameter Rate	Total Failure Rate/Hr (Mean)	Reference	Rupture* Failure Rate/Hr
Pipe Failure Section (100')	8.5E-10	WASH-1400	4.3E-11
External Failure of a Valve	2.7E-8	WASH-1400	1.3E-9
External Failure of a Pump	3.0E-9	WASH-1400	1.5E-10

*Based upon the operating experience to date, given that a failure occurs, the ratio of external leaks to complete failures appears to be in the range of 20 to 1. This is substantiated by the specific data review cited in the text for values (18 to 1) and data published by Bush[5] on pipes (4 to 1 up to 30 to 1). Because the internal flood evaluation is based upon initiators with substantial flooding rates, i.e., short operator response times, only the catastrophic or large external rupture failures are treated in this evaluation.

Table 2.2.4 Conditional Probability of Pipe Break Size

Break Size	Characterization	Flow Rate	Conditional Probability
Maximum	Guillotine Break	100%	0.05
Large	Substantial Rupture	50%	0.10
Small*	Localized Rupture in Ductile Material	13%	0.85

*Remainder of the conditional probability was allocated to small breaks.

Table 2.2.5 Initiating Event Frequency Estimates Involving Component Leak/Ruptures

INITIATOR	SOURCE	VALVES				PIPING LENGTH (FT)/ SECT/DIA (IN)	ESTIMATED FREQUENCY/ YR
		MOV	MAN	CHK	PUMPS		
HPCI Discharge T_{FL6}	CST/SUPP	2	0	1	0	76/1/14	3.5E-5
CS Discharge T_{FL7}	SUPP	4	0	2	0	128/2/12	6.9E-5
LPCI Discharge T_{FL8}	SUPP	14	4	4	0	240/6/16	2.5E-4
Service Water T_{FL9}	Service Water	4	4	4	0	715/8/10-20	1.4E-4
WFPS T_{FL10}	WFPS	1				157/2/6-8	1.1E-5
RCIC** Suction T_{FL11}	CST	1	1	1	1	70/1/6	3.5E-5
HPCI** Suction T_{FL12}, T_{FL13}	CST**	1	1	1	1	87/1/16	3.5E-5
CS** Suction T_{FL14}, T_{FL15}	CST*	2	2		2	120/2/12	4.9E-5
LPCI** Suction T_{FL16}, T_{FL17}	SUPP	4			4	120/2/20	5.2E-5

*CST is assumed to be the source.
**Suction failures are also classified by flow rate.

Table 2.2.6 Calculated Frequencies for Initiating Events Resulting from System Ruptures (SNPS-PRA)

Initiator	Frequency (Per RX Yr)
Pressurized Piping	
HPCI Discharge Break, TFL6	3.5×10^{-5}
CS Discharge Break, TFL7	6.9×10^{-5}
LPCI Discharge Break, TFL8	2.5×10^{-4}
SW Discharge Break, TFL9	1.4×10^{-4}
WFPS Discharge Break, TFL10	1.1×10^{-5}
Non-Pressurized Piping	
RCIC Suction Failure, TF11 (max)	1.75×10^{-6}*
HPCI Suction Failure, TF12 (max)	1.75×10^{-6}*
HPCI Suction Failure, TF13 (large)	3.5×10^{-6}*
CS Suction Failure, TF14 (max)	2.5×10^{-6}*
CS Suction Failure, TF15 (large)	4.9×10^{-5}*
LPCI Suction Failure, TF16 (max)	2.6×10^{-6}*
LPCI Suction Failure, TF17 (large)	5.2×10^{-6}*

*Modified based upon engineering judgement made on the size of low pressure suction line breaks.

Table 2.3.1

THE PROBABILITY THAT FLOOD REMAINS UNISOLATED FOR X MINUTES AFTER AUTOMATIC PLANT ACTION: E.G., TURBINE TRIP OR MSIV CLOSURE

X	P(for multiple event)	P(for single event)
1	1	1.0
10	1st + 2nd = 0.54	0.1
20	0.11	0.01
30	0.011	1.1E-3
60	0.0011	2.0E-4
1500	1.1E-4	1.1E-4

3.0 BNL ACCIDENT REVIEW AND SEQUENCE QUANTIFICATION

This section discusses the quantification and review of the internal flooding accident sequences in the SNPS-PRA due to system maintenance and pipe ruptures. The section is organized as follows. Subsection 3.1 presents a summary of the approach used by BNL to calculate the initiator frequencies. Subsection 3.2 discusses BNL quantitative review of the initiator event trees, and Subsection 3.3 presents the functional event tree analysis and evaluation.

3.1 Flood Precursor Frequency

This review revised the assessment of the frequency of the flood initiators in two ways. First the experiential data for the estimation of the various failure rates were revised to include recent events. Second, the models for calculating the frequency of floods (or probability per year of reactor operation) have been improved by removing unnecessary conservatisms. As it was already discussed in Section 2.2, two types of initiators were considered: a) maintenance-induced initiators; and b) rupture-induced initiators. The revised frequencies for these types of initiators are presented in the following two subsections.

3.1.1 Maintenance-Induced Flood Initiators

A flood can be initiated during the maintenance of a component of the ECCS or of another system in the Elevation 8 area if the maintenance process requires dismantling of the component and one of the isolation valves opens inadvertently while the component is being maintained.

The components that contribute to these initiators are the pumps and the heat exchangers in the Elevation 8 area. These are standby components that can fail in a time-dependent fashion while on standby. Periodic tests are performed to check their operability and if found failed they are put under repair.

A Markov model that describes the stochastic behavior of these components has been developed and quantified. The important characteristics of this model are as follows:

i) The component can be in six states (see Figure 3.1.1).

ii) In state 1 the component (pump, heat exchanger) is available, that is ready to start operating if asked to do so.

iii) The component while on standby can fail with exponentially distributed times to failure. A failure brings the component into state 2 (see Figure 3.1.1).

iv) The failure remains undetectable until a test is performed or a real challenge is posed to the component. A test that will find the component in state 2 will initiate a repair action. The same will happen following a real demand for the component.

v) There are three repair states. States 3 and 3' in which the component is under repair while the reactor is online, and State 4 where the component is under repair with the reactor shutdown.

vi) Following a test that finds the component failed and before the dismantling of the component, all the appropriate motor operated valves must be closed and their breakers racked out from the corresponding MCCs. There is, however, a chance that the operator will not remove the breakers from the MCCs leaving then the MOVs able to open following a signal to do so. If the probability of such an error is P, then a test brings the component from State 2, to State 3 with probability 1-P (breaker removed, and to State 3' with probability P.

vii) The component remains in States 3 or 3' until the repair is completed and then it returns to State 1, or until the allowable outage time is exhausted and then the component transit to State 4 where the repair continues with the reactor shutdown. When the repair is completed, the reactor is brought back online and the component returns to State 1 (Transition 4 to 1).

Quantification

The solution of the model requires the quantification of the following parameters.

i) The catastrophic failure rate λ. This failure mode implies such failures that require major maintenance (dismantling) of the component. The SNPS-PRA used the data presented in Table 2.2.1 from Ref. 2. BNL has updated this table using additional data included in an updated version of Ref. 2 (Ref.6). The new data are summarized in Table 3.1.1.

Maximum likelihood estimators for the failure rates

$$\lambda=\left(\frac{\text{number of failures}}{\text{total operating time}}\right) \text{ yield}$$

$$\lambda=5.7\times10^{-5}/\text{hr for Turbine Driven Pumps}$$

and

$$\lambda=3.3\times10^{-6}/\text{hr for Motor Driven Pumps}$$

ii) The mean times to repair were assumed 100 hrs and 50 hrs for the turbine driven and the reactor driven pumps, respectively. Thus

$$\mu=10^{-2}/\text{hr for Turbine Driven Pumps}$$

and

$$\mu=2\times10^{-2}/\text{hr for Motor Driven Pumps.}$$

iii) In the BNL revision of the SNPS-PRA, the frequency of transients involving MSIV closure has been assessed at 4.42/yr. Thus, the frequency of transients on an hourly basis is

$$\lambda_D=5.0\times10^{-4}/\text{hr}$$

iv) Tests are performed every 3 months (4 times a year) for both motor-driven and turbine-driven pumps. The allowable outage times are 14 and 7 days for turbine-driven and motor-driven pumps, respectively.

v) The probability of not racking out the breakers of the isolation valves (P) is assessed in the SNPS-PRA as 10^{-2}. The same value is used in these requantifications.

vi) The mean time for inadvertently activating a particular switch in the control room has been assumed equal to 10,000 hrs. This implies a rate of

$$\lambda_0 = 10^{-4}/\text{hr}.$$

Quantification of the Markovian model with the numerical values of the parameters mentioned above yields the probabilities per year for the various maintenance induced floods. The results are tabulated in Table 3.1.2. Additional assumptions are: the Core Spray System consists of two motor driven pumps, the LPCI consists of four motor driven pumps and that RBCLCW heat exchangers are equivalent to motor driven pumps.

3.1.2 Rupture-Induced Flood Initiators

A flood can be initiated if a rupture occurs at any point in the pressure boundary of the various systems in the Elevation 8 area. Such a rupture will involve one of the following three types of components: 1) piping; 2) valve; and 3) pump. The model assumes that catastrophic ruptures occur in the following way. A component fails in such a way that if it is demanded to operate then a catastrophic rupture (large enough to allow the flow rates necessary for the flood sizes of interest to this analysis) will occur. That is, the component transits first in a rupture-vulnerable state and then, when a demand occurs, it ruptures.

A Markov model that decribes this stochastic behavior has been developed and quantified. The model is graphically depicted in Figure 3.1.2. The basic characteristics of the model are as follows:

(i) The system in question (HPCI, RCIC, LPCI, CS, RHR, RBCLCWHX) is in state where it is available to perform its function.

(ii) The system transits to State 2, which is a rupture vulnerable state with failure rate λ_R.

(iii) If a demand occurs while in State 2 a flood is initiated. A demand occurs whenever a transient, a manual shutdown or a test occurs. We distinguish three flood states: State 3, which is a rupture triggered by a transient involving an MSIV closure; State 4, which is a rupture triggered by a turbine-trip transient; and State 5 which is rupture triggered by a manual shutdown or an equipment test.

The solution of this model yields the probabilities that the system will occupy States 3, 4 and 5 denoted by P_S, P_T, P_M, respectively. These probabilities at the end of one-year period provide the frequency of rupture-initiated flood precursors. The expression for these probabilities is

$$P_i(t) = F \frac{\lambda_i\lambda_R}{\lambda-\lambda_R} [(1-e^{-\lambda_R t})/\lambda_R-(1-e^{-\lambda t})/\lambda] \tag{1}$$

where i = S, T

F is the number of tests per year.

λ_i is the rate of arrival of a transient of type i (i=S,T)

λ_R is the rate of catastrophic rutpure failure in the system

and

λ is the rate of arrival of any transient ($\lambda=\lambda_S+\lambda_T+\lambda_M$)

For the manual shutdown the corresponding expression is

$$P_M(t) = F[\frac{\lambda_M\lambda_R}{\lambda-\lambda_R}(1-e^{-\lambda_R t})/\lambda_R-(1-e^{-\lambda t})/\lambda + \frac{\lambda_R}{\lambda-\lambda_R}(e^{-\lambda_R T}-e^{\lambda T})] \tag{2}$$

Quantification

For a given system having piping of length L, n_v valves and n_p pumps the failure rate λ_R is equal to

$$\lambda_R = L\lambda'+n_v\lambda_v+n_p\lambda_p \tag{3}$$

where λ_V, λ_P are the catastrophic rupture failure rates for valves and pump and λ' the same failure rate per unit of piping length.

A search of the LER, has indicated that at least one pipe rupture (welding failure) has occurred in the ECCS piping in the 215 accumulated BWR years (see Ref.8).

This provides a maximum likelihood estimator for the rupture failure rate of ($1/215yr = 5.31\times10^{-7}/hr$). Assuming, as in the SNPS-PRA, that only one out of every twenty ruptures will create a break large enough to generate floods of the sizes of concern to this analysis, the catastrophic piping rupture rate becomes $\lambda = 2.7\times10^{-8}$. This of course is applicable for the total length of safety related piping (denoted by L).

For a particular system with a total of piping length ℓ, then the catastrophic rupture rate for piping becomes

$$\lambda'' = \left(\frac{\ell}{L}\right)\times 2.7\times10^{-8}/hr \tag{4}$$

where ℓ/L denotes the fraction of the total length of the piping that belongs to the particular system.

For the rupture rates of the valves and the pumps, the WASH-1400 values were used (see Table G.4-4 in SNPS-PRA). Using the length of piping, number of valves and pumps provided in Table G.4-5 of the SNPS-PRA, and by virtue of Eqs.1-3. The total failure rate λ_R for the various systems along with the probabilities P_S, P_T and P_M were calculated. The results are tabulated in Table 3.1.3.

A total of 13.51 transients per year were assumed (4.42 MSIV closures, 4.89 turbine trips and 4.2 manual shutdowns).

The splitting between maximum and large floods for initiators TFL12-TFL13, TFL14-TFL15, TFL16-TFL17 was done as in the SNPS-PRA, that is, 1 to 2. The additional factor of 20 used in the SNPS-PRA to account for non-pressurized piping is not assumed in the BNL quantification.

3.2 BNL Quantitative Review of the Initiator Event Tree

The quantitative review of the initiator event trees is discussed in the following subsections.

3.2.1 Review of Flooding Alarm Related Procedures

The RB water level is detected by two RB water level monitors installed on the RB floor. The flood alarms are activated by the monitors when the water level is more than 0.5 in. above the floor. The sump alarms will be activated when water level reaches the sump alarm setpoints installed at a level right below the level that activates the RB flood alarms. Sump alarm sensors are installed at various locations in the RB.

The immediate operator action specified in the Alarm Response Procedure (ARP5671) is to initiate the Suppression Pool Leakage Return System. The required subsequent actions are:

1. Monitor RB water level to determine approximate leak rate. Use sump alarms to supplement the information obtained from the above instruments to ascertain the approximate location of the leak.
2. Monitor parameters (such as line pressure and flow rate) of the safety systems as a leak would affect the system parameters. Isolate the source of leakage per procedure listed below in Step 3.
3. If required and plant condition permit, dispatch an operator to the RB floor to visually locate the source of leakage. Isolate using the appropriate system procedure listed below.

<u>System</u>

HPCI, Procedure No.SP23.202.01

Leakage indication:
- . Abnormal suction or discharge piping pressure.
- . Excessive HPCI Loop Level Pump Flow or low discharge pressure.

. Reactor building sump high water levels in vicinity of leak.
. Reactor building flooding alarm.

Leakage isolation:
. If in standby, isolate the HPCI system by securing the HPCI Loop Level Pump and then closing CST Suction Valve (MOV-031).
. If the system is operating, secure per shutdown procedure and then isolate as described above.

RCIC, Procedure No.SP23.119.01

Leakage indication:
. Abnormal suction or discharge piping pressure.
. Excessive HPCI Loop Level Pump.
. Reactor building sump high water levels.
. Reactor building flooding alarm.

Leakage isolation:
. If in standby, isolate the RCIC system by securing the RCIC Loop Level Pump and then closing CST Suction Valve (MOV-031).
. If the system is operating, secure per shutdown procedure and then isolate as described above.

RHR, Procedure No.SP23.121.01

Leakage indication:
. Heat exchanger service water side temperature inconsistencies.
. Abnormal RHR system flow for mode of operation.
. Abnormal RHR system pressures for mode of operation.
. Reactor water level inconsistencies for mode of operation.
. Sump high level alarms.
. Reactor building flooding alarm.

Leakage isolation:
. Isolate the leakage by shutting down the affected loop in accordance with the appropriate procedure

for the mode in which it was operating and then systematically shutting valves to isolate areas of the system found above to be possible sources of leakage.

- The above isolation procedure may require intermittent operation of the leakage return system to observe the effects on water buildup.
- When the leakage has been isolated return the unaffected portions (as required) to service.

BNL has found that SNPS alarm response procedures specify general guidelines for monitoring system parameters for determining the leakage location and for initiating the leakage isolation. However, the procedures fail to include specific requirements for operators to systematically check the operation parameters of relevant systems. Since there are many system parameter indicators in the control room, the operators may possibly fail to observe the indication of an abnormal system parameter.

When the abnormal condition is severe enough to actuate the alarm of a particular system parameter, the corresponding Alarm Response Procedure will then be followed by operators. However, BNL has reviewed the relevant Alarm Response Procedures for abnormal system parameters, and found that these procedures do not contain steps that should be followed under RB flood conditions. These procedures provide guidelines for conditions other than RB flood, such as water source abnormal or isolation valves abnormal, etc. The operator responses to the flood could be delayed or confused when these Alarm Response Procedures are followed.

3.2.2 Requantification

The revised initiator frequencies are applied for evaluating the sequence frequencies of the initiator event tree. In addition to the critical flood depth of 3'-10" used by SNPS, BNL also evaluated the sequence frequencies corresponding to flood depth of 1'-10" and 1'-3". This is because, as indicated in Table 3.2.1, flood heights of 1'-10" and 1'-3" will disable several vital systems such as HPCI and RCIC. The times for the flood to reach 3'-10", 1'-10", and 1'-3" depth were calculated based on the leakage flow rates determined in SNPS PRA. The calculated times are shown in Table 3.2.2.

The HEP values used by SNPS are identical to the nominal HEP values provided in the Probabilistic Risk Analysis Procedure Guide(7) (see Figure 3.2.1 and Table 3.2.3). BNL feels that the HEP could be higher than the nominal HEP values because the flooding alarm related procedures fail to provide specific guidelines to identify and to isolate the flood source (see Section 3.2.1).

The HEPs under the multiple alarm and the single alarm conditions are listed in Tables 3.2.4 and 3.2.5.

3.3 BNL Review of Functional Event Tree

This section is divided into three subsections. Section 3.3.1 provides a qualitative review of the Shoreham Internal Flood event tree analysis and Section 3.3.2 presents the BNL revised time phased event trees. Section 3.3.3 describes the results obtained from the quantification of the BNL event trees.

3.3.1 Qualitative Review

In general, BNL is of the opinion that the methodology used in the Shoreham Internal Flood Analysis is consistent with that of the state-of-the-art and the approach is reasonable. The analysis for the internal flood postulated much severe scenarios than those of the Shoreham FSAR.

The Shoreham Internal Flood functional event tree analysis is based predominantly on the event trees developed for the internal event initiators, namely, turbine trip, MSIV closure and manual shutdown. These internal flood functional event trees only model flood scenarios where the flood water height at Elevation 8 exceeds 3'-10". While it appears that the Shoreham functional event trees do provide a representative modeling of the plant response, it is not well substantiated that floods that are arrested before reaching 3'-10" will result in negligible core vulnerable frequency.

Table 3.3.1 enumerates the vital equipment that has been identified in the Shoreham analysis. The components that are located at the lowest elevation are presented first. It can be seen that at the 1' level, both the RCIC and HPCI vacuum pumps and condensate pumps are expected to be disabled. However, it is judged that their failures do not lead to the failure of the respective high pressure systems. Similar arguments apply to the loop level pumps of the low pressure core spray, HPCI and the RCIC systems as well. At approximately 2', instrumentation for both high pressure injection systems are submerged and hence resulting in failure of both systems. At 3'-10" instrumentation for both LPCS and RHR is submerged leading to the failure of those low pressure systems. In the Shoreham analysis the critical height of 3'-10" is selected. However, since both HPCI and RCIC have failed at about the 2' level, these scenarios with termination of the flood prior to 3'-10" may not contribute an insignificant amount to the core vulnerable frequency. In the BNL revised event trees, a time-phased approach is used to include the contribution from flooding below the 3'-10" level.

Another area of concern stems from the treatment of propagation of failures in the Shoreham analysis. As noted in Table 3.3.1, at the 1' level, 4-480V pumps are expected to experience electrical shorts. The Shoreham analysis did not investigate any cascading failure which may result from the electrical shorts. BNL reviewed the electrical drawings and elementary drawings for some of the systems. It appears that for each pump there is only one electrical breaker which separates it from the rest of the loads in the same motor control center (MCC). Random failure of this breaker to open could result in the propagation of the short circuit fault upstream to the MCC, other MCCs and the load center. BNL's review of the electrical diagrams indicates that failure of the breaker to open will result in tripping the breaker at the load center. Discussions with Shoreham engineers suggested that there may possibly be an additional breaker per pump that is in series with the first breaker. However, this was not confirmed by BNL. In the BNL revised event trees, only one breaker is assumed and its failure is modeled explicitly.

BNL did not review the spraying effects due to water cascades from higher elevations.

3.3.2 BNL Time Phase Event Tree

The determination of the time periods which are critical to the consideration of the progression of the flood is based on the vital equipment location list (Table 3.3.1). Three heights were selected for the BNL analysis: at the 1'-3" level, at the 1'-10" level, and at the 3'-10" level. If the flood is terminated prior to reaching the 1'-3" level, no impact is assumed for any equipment and the plant will be shutdown, this is Phase 1. However, if the flood water exceeds the 1'-3" level but is terminated before the 1'-10" level, this is Phase II. Phase III entails the failures of both HPCI and RCIC system as well as the loss of power to the MG set recirculation pump fluid coupler before arresting the flood below the 3'-10" level. Any flood level which exceeds the 3'-10" level, it is treated in Phase IV.

The event trees of these four phases are presented in Figures 3.3.1 through 3.3.4. Given that the flood is terminated in Phase I, BNL assumed that the reactor has a high probability (0.9) that it will be manually shutdown. Ten percent of the time, it may result in a MSIV closure event. These two branches of the Phase I event trees are transferred to the respective internal event tree, Figure 3.3.1.

Figure 3.3.2 depicts the Phase II functional event tree which considers the various accident mitigation systems. Moreover, owing to the fact that a number of the 480V pumps will be flooded, the possibility of a breaker failure to isolate the fault is also evaluated. It is assumed that the breaker failure to open probability is $1x10^{-3}$ and there are a total of five pumps in Division 1 and two pumps in Division II that will be short circuited. A probability of 0.5 is also assumed that failure of a load center in a division would lead to failure of other equipment connected to that division. In the event of a MSIV closure, the feedwater system is considered to be unavailable. The probability that the reactor will be manually shutdown is also assumed to be 0.9 for the maintenance induced flood events.

Figure 3.3.3 illustrates the functional event tree used to describe the Phase III events. The major difference between this event tree and the Phase

II tree is the high pressure systems. In the Phase III events, both the RCIC and the HPCI systems are unavailable due to the failure of respective instrumentation. The probability that the reactor will be manually shutdown is assumed to be 0.5 for the maintenance induced flood events.

The Phase IV event tree is presented in Figure 3.3.4. This tree is drastically different from the other ones in that it only considers the feedwater system, the depressurization function and the PCS. All the other systems are disabled due to flooding. The likelihood that the reactor will be manually shutdown is the same as in Phase III for maintenance-induced floods.

3.3.3 Quantitative Analysis

Based on the development of the revised flood initiator frequency, the BNL time-phased event tree and the modified human response to arrest flood, quantitative results are obtained. In the BNL analysis, there are 17 different flood precursors. Similar to the Shoreham classification, the first five precursors are online maintenance related; the remaining twelve of them are rupture related. A detailed discussion on the BNL flood precursors is given in Section 3.1.

Owing to the ways that these flood precursors are calculated, the initiator event trees have been modified to include only three functions: the flood alarm annunication, I; operator action to isolate flood, A; and reactor status. The entry condition to the different time phase event trees is determined by the A function (see Section 3.2 for details).

Each of the 17 flood precursors were evaluated with the initiator event tree and the four time phase event trees. The unavailability values for the various event trees are the same as those used in the Shoreham analysis except as noted in the last section.

When the time phase event trees were quantified for the 17 flood precursors, the results are the conditional frequency of core vulnerable given the particular flood precursor. These frequencies are summarized in Table

3.3.2. The seventeen precursors are listed as rows while the four phases are shown as columns. Within each precursor, contributions from manual shutdown, MSIV closure or turbine trip are also shown. For instance, the conditional frequency of core vulnerable with operator arresting the flood prior to 3'-10" but after 1'-10" - Phase III, for TFL1 is 2.0(-5) given the reactor is manually shutdown. However, if instead of a manual shutdown, the plant experiences a MSIV closure, then the conditional frequency is 8.5(-4).

As expected, the conditional frequency consistently increases as the flood progresses to higher elevations. In other words, the conditional frequency of Phase IV is always larger than any of the other phases. Another noteworthy observation is the unusually large conditional frequency of core vulnerable for the LPCI system induced flood, i.e., TFL4 and TFL8. The TFL9 and TFL5 values are also large since they disabled the LPCI systems as well.

The core vulnerable frequency given the BNL revised flood precursors, initiator event trees and time phase event trees is shown in Table 3.3.3. In this table, the 17 precursors are depicted on the left with the 4 phases depicted as columns. Each precursor also identifies the contributions from the various plant states. Core vulnerable frequency contributions from Phase I and II are very small, in the order of 10^{-9}. Contributions from Phase III are not insignificant but not substantial, approximately 10^{-6}. Seventy percent of the total core vulnerable frequency (70% of 2.0(-5)) is attributable to LPCI system maintenance or rupture induced flood. The maintenance contribution to flood is about 37% while the balance is due to rupture.

It appears also that failure to properly model the fault propagation of the short circuits through the breakers does not have a significant effect on core vulnerable frequency.

3.4 Uncertainty Estimates

This section presents a limited uncertainty assessment on the BNL quantitative analysis for the core vulnerable frequency due to reactor building flooding.

A rigorous propagation of the uncertainties is outside the scope of the present review. The BNL approach for the uncertainty evaluation consisted of the following general steps.

1. The uncertainties in the human errors as well as the split ratio between the manual shutdown and the MSIV closure event were quantified by fitting lognormal distributions to evaluate uncertainty measures (mean and variance). An error factor of 10 was applied to human errors and the split ratio.

2. Human errors of the following operator actions were included for the uncertainty evaluation:

   - Operator maintains isolation valves in closed position during the online maintenance (Event E, see Section 2.3).

   - Operator diagnoses and responds to isolate the flood (Event A, see Section 2.3).

   - Operator depressurizes the Reactor Pressure Vessel (Event X, Figures 3.3.2-3.3.4).

3. The uncertainties in the core vulnerable frequency were evaluated using the major accident sequences and the distributions assessed in Step 1.

The SAMPLE code was used for the estimaton of uncertainties. The mean, median, 5% and 95% probability intervals for the core vulnerable frequency are shown as follows.

Mean	=	1.91E-5
Median	=	1.90E-6
5% Confidence	=	2.19E-7
95% Confidence	=	7.51E-5

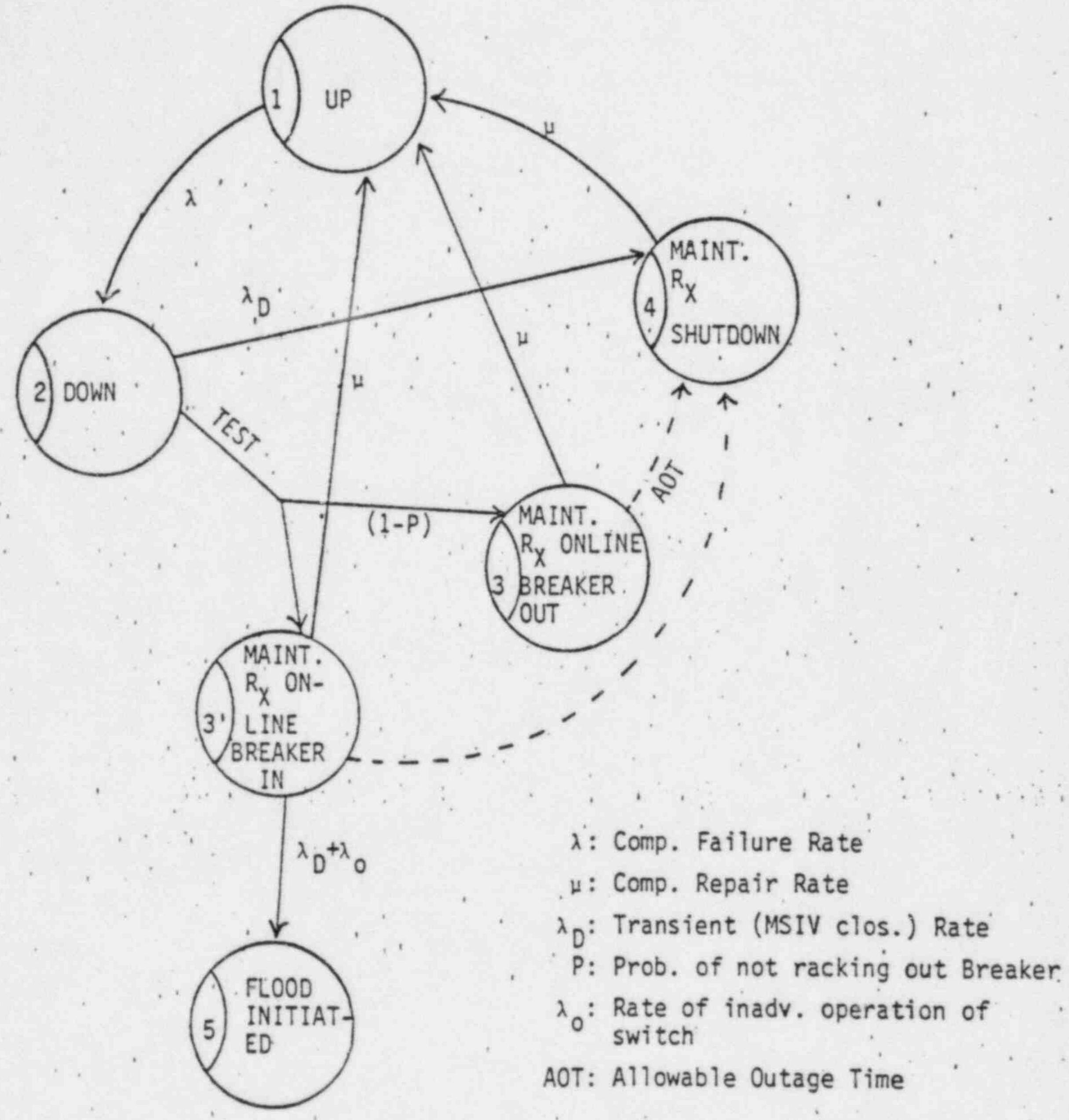


Figure 3.1.1 State Transition Diagram for Component-Maintenance Induced Floods.

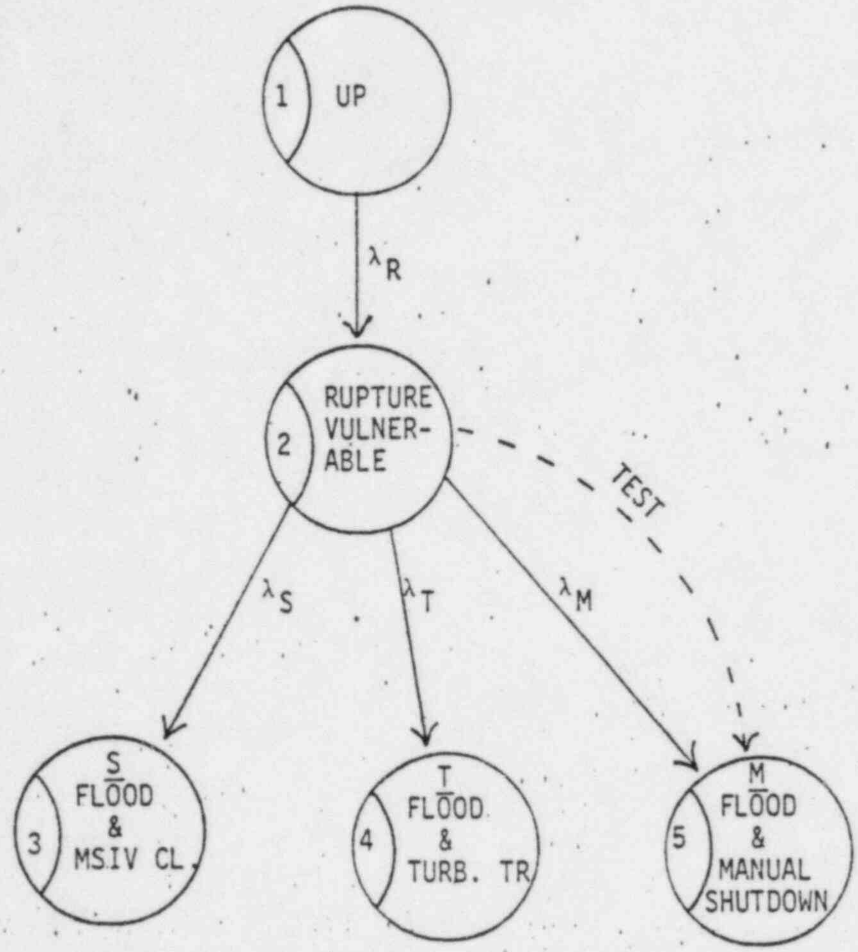


Figure 3.1.2 State Transition Diagram for Rupture-Induced Floods.

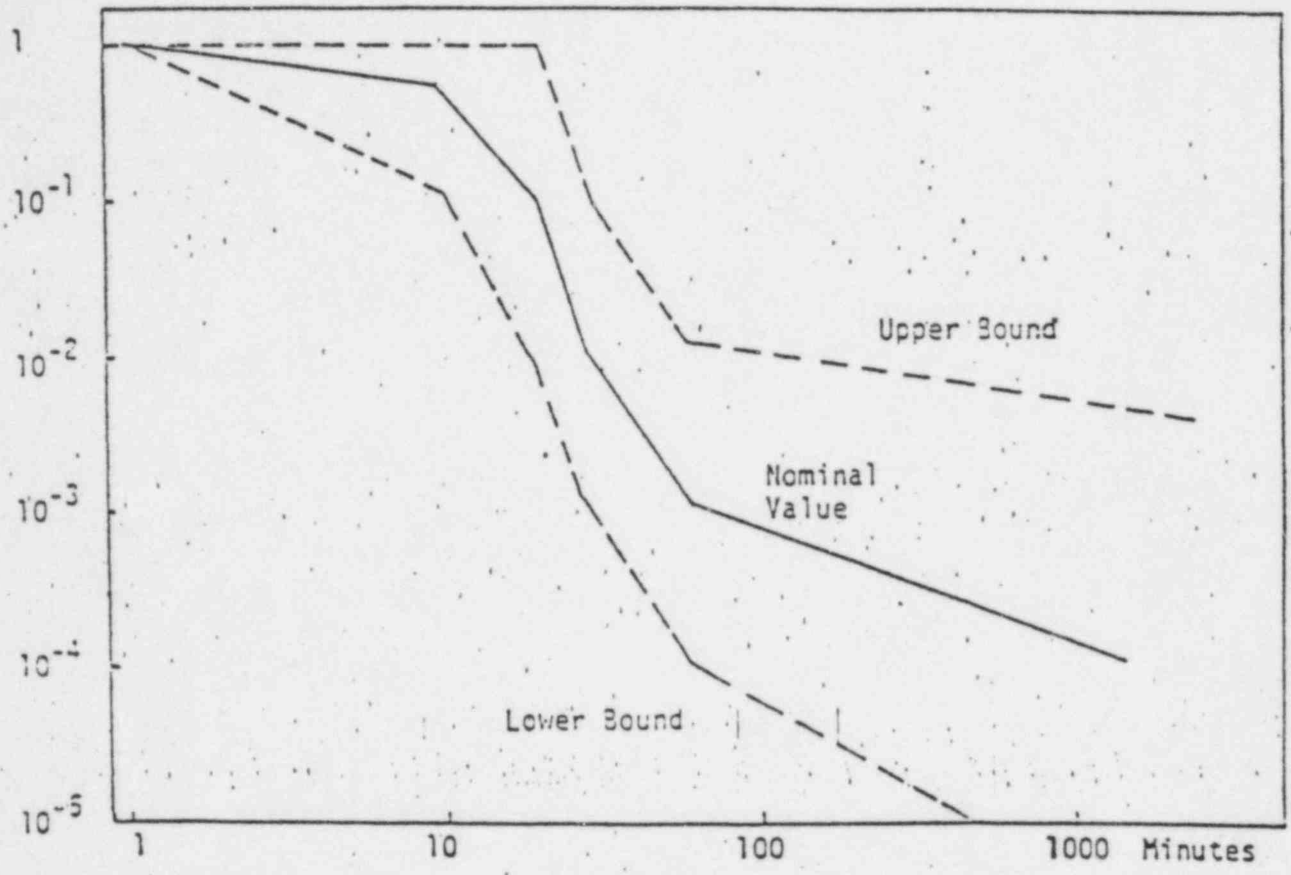


Figure 3.2.1 Problem-solving human error probability vs time - screening values.

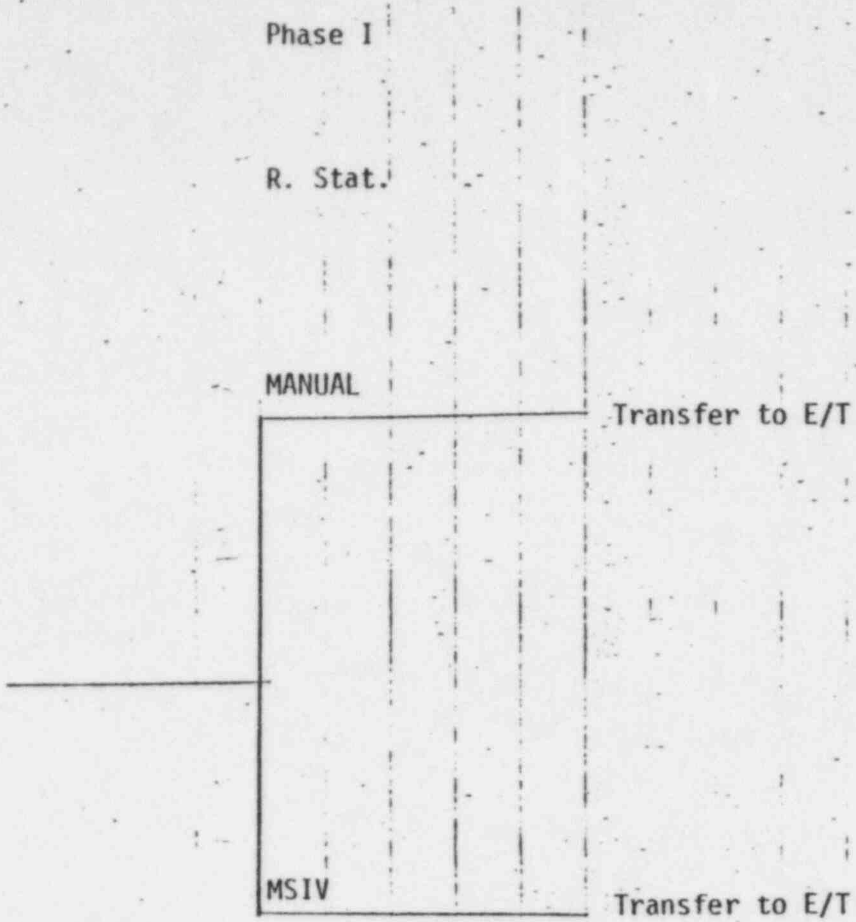


Figure 3.3.1 Phase I of Internal Flood Functional Event Tree

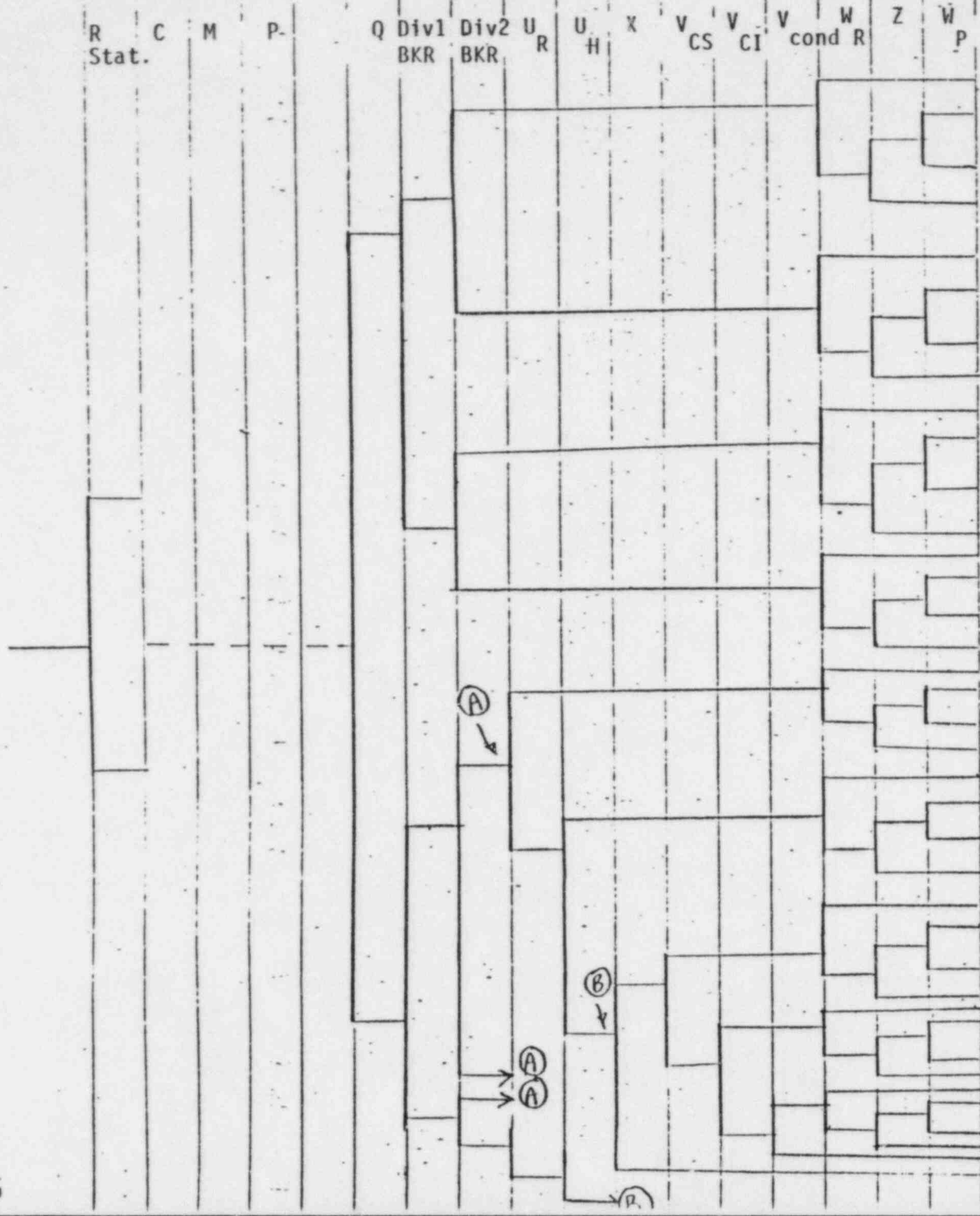


Figure 3.3.2 Phase II of Internal Flood Functional Event Tree

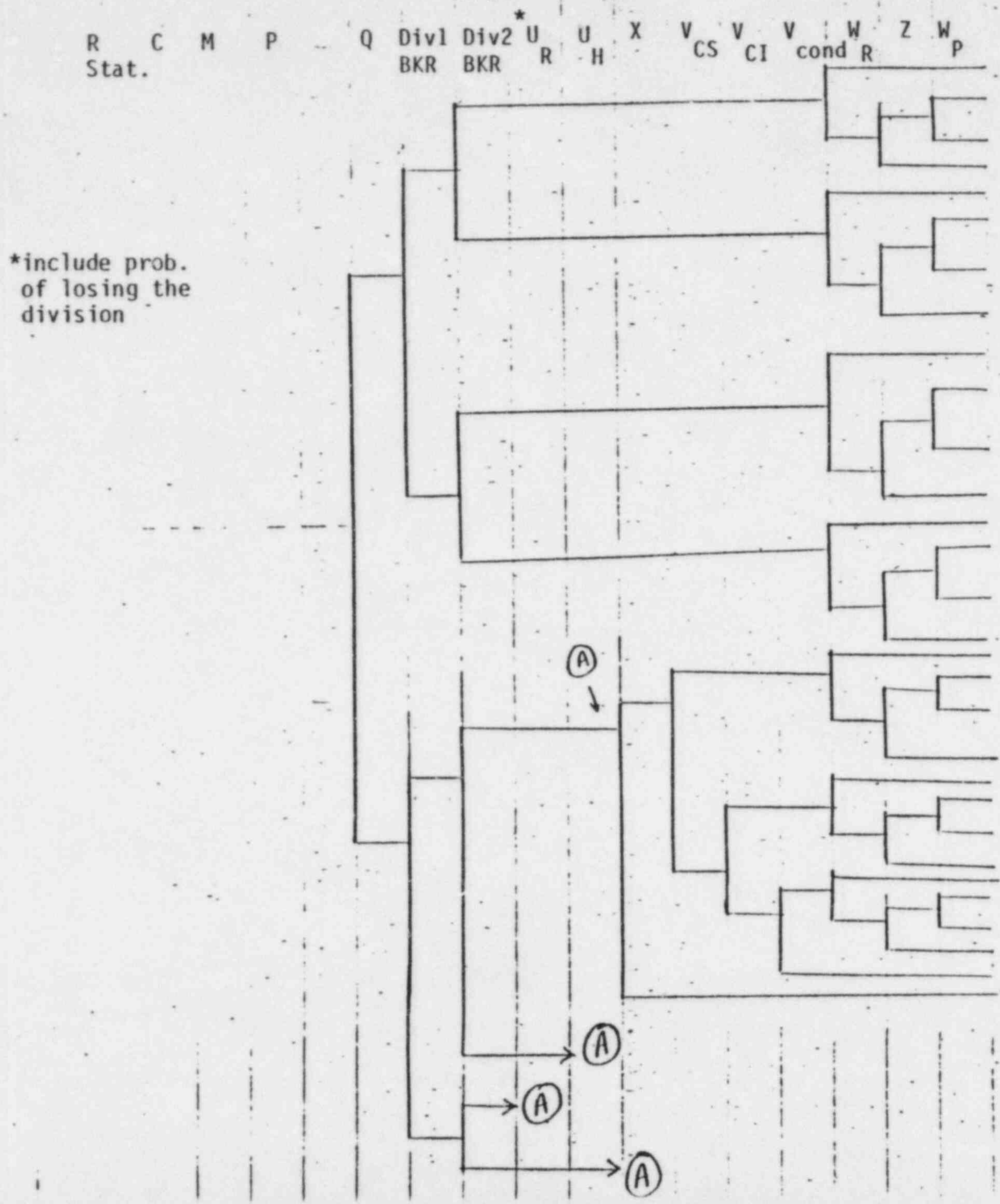


Figure 3.3.3 Phase III of Internal Flood Functional Event Tree

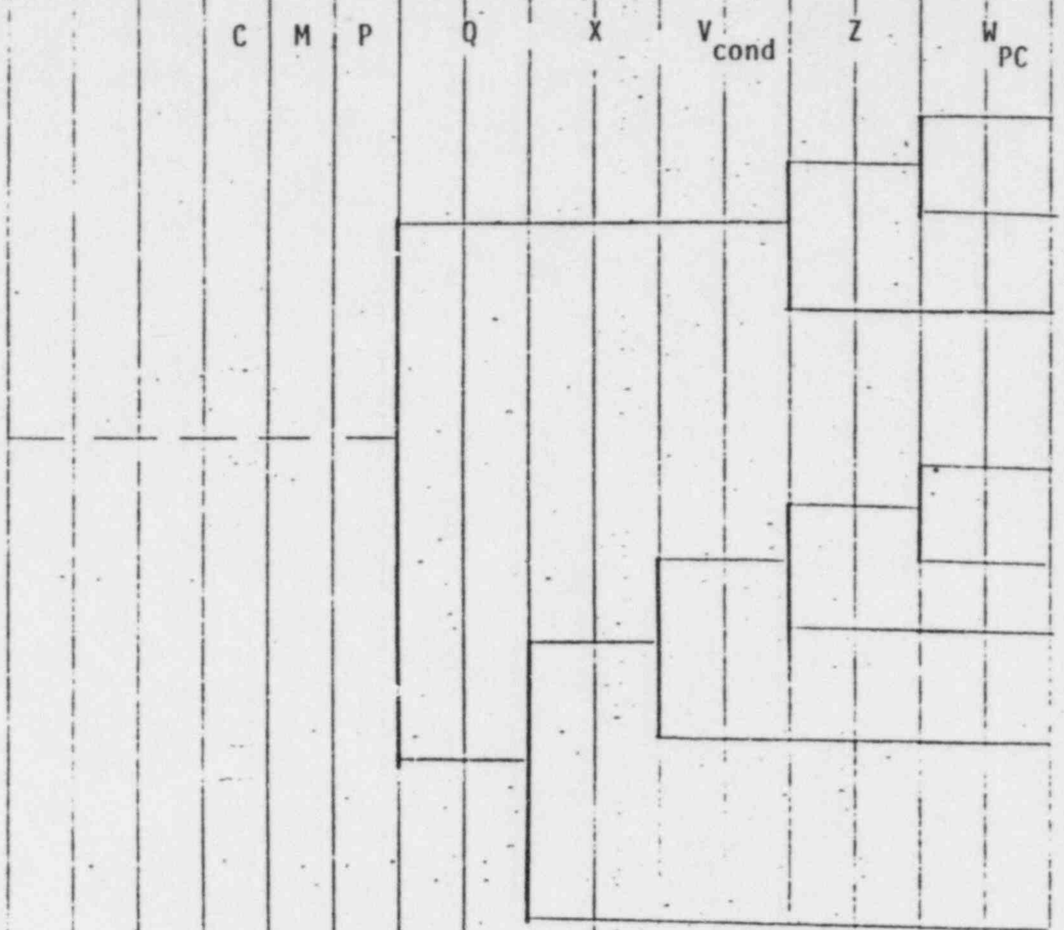


Figure 3.3.4 Phase IV of Internal Flood Functional Event Tree

Table 3.1.1 LER Data for BWR Standby Pumps for the Period of January 1972 Through September 1980

Standby Pumps	Demands	Standby Hours	Leakage Rupture	Does Not Start	Loss of Function	Does Not Continue To Run
Motor Driven	20,321	10,453,806	9	8	8	9
Turbine Driven	2,860	1,439,491	-	34	25	23

Table 3.1.2 Frequency of Maintenance - Induced Flood Precursors

System	Initiator Event Trees	Probability per Year
1. RCIC	TFL1 P.D	1.05×10^{-4}
	TFLI P.E_0	2.10×10^{-5}
	TFL1 P.E_L	2.10×10^{-5}
2. HPCI	TFL2 P.D	1.05×10^{-4}
	TFL2 P.E_0	2.10×10^{-5}
	TFL2 P.E_L	2.10×10^{-5}
3. Core Spray	TFL3 P.D	1.89×10^{-5}
(2 motor driven pumps)	TFL3 P.E_0	1.87×10^{-6}
4. LPCI	TFL4 P.D	3.78×10^{-5}
(4 motor driven)	TFL4 P.E_0	3.74×10^{-6}
5. Service Water	TFL5 P.D	1.89×10^{-5}
(RHR or RB(LW HX)	TFL4 P.E_0	1.88×10^{-6}
2 motor driven pumps		

Table 3.1.3 Flood Precursor Frequency

	Pipe	Valves	Pump	Total λ_R	P_S	P_T	P_M
TFL6	1.2(-9)	6.5(-9)	0	7.7(-9)	1.6(-5)	1.7(-5)	1.5(-5)
TFL7	2.0(-9)	1.3(-8)	0	1.5(-8)	3.1(-5)	3.4(-5)	2.9(-5)
TFL8	3.7(-9)	2.9(-8)	0	3.2(-8)	6.5(-5)	7.3(-5)	6.2(-5)
TFL9	1.1(-8)	2.3(-8)	6.0(-10)	1.3(-8)	2.6(-5)	2.9(-5)	2.5(-5)
TFL10	2.4(-9)	1.3(-9)	0	3.7(-9)	7.5(-6)	8.4(-6)	7.2(-6)
TFL11	1.1(-9)	9.1(-9)	1.5(-10)	1.0(-8)	2.1(-5)	2.4(-5)	2.0(-5)
TFL12	1.4(-9)	3.9(-9)	1.5(-10)	5.5(-9)	3.7(-6)	4.0(-6)	3.6(-6)
TFL13	-	-	-	-	7.3(-6)	8.0(-6)	7.1(-6)
TFL14	1.9(-9)	5.2(-9)	3.0(-10)	7.4(-9)	5.0(-6)	5.6(-6)	4.8(-6)
TFL15	-	-	-	-	1.0(-5)	1.1(-5)	9.6(-6)
TFL16	1.9(-9)	5.2(-9)	6.0(-10)	7.7(-9)	5.2(-6)	5.8(-6)	5.0(-6)
TFL17	-	-	-	-	1.0(-5)	1.2(-5)	1.0(-5)

Table 3.2.1

MAJOR ELEVATION 8 EQUIPMENT LIST

EQUIP. TYPE	EQUIPMENT DESCRIPTION	PART NO.	POSTULATED DISABLED HEIGHT*
PUMPS			
	Floor Drain Sump Pumps	1G11*P-035A-D 1G11*P-036A-F	1'-0"
	Dry Floor Drain Tank Pumps	1G11*P-161A,B	1'-0"
	Radwaste Equip. Drain Sump & Pump to Porous Sump	1G11*P-224A,B	1'-1"
	** HPCI Pump	1E41*P-016	-----
	HPCI Vacuum Pump	1E41*P-075	1'-0"
	HPCI Con. Pump	1E51*P-076	1'-0"
	** RCIC Pump	1E51*P-015	-----
	RCIC Vacuum Pump	1E51*P-076	1'-0"
	RCIC Con. Pump	1E51*P-077	1'-0"
	** RHR Pump Motors	1E11*P-014A-D	5'-4"
	** Leakage Return Pump	G11-*P-270	3'-9"
	** Core Spray Loop Level Pumps	1E21*P-049A,B	1'-3"
	Drywell Equip. Drain Tank Pumps	1G11*P-0332A,B	1'-2"
	RCIC Loop Level Pump	1E51*P-051	1'-4"
	** HPCI Loop Level Pump	1E41*P-050	2'-3"
TURBINES			
	** HPCI Turbine	1E41*-TU-002	6'-0"
	** RCIC Turbine	1E41*-TU-005	4'-0"

Table 3.2.1 (Continued)

MAJOR ELEVATION 8 EQUIPMENT LIST

EQUIP. TYPE	EQUIPMENT DESCRIPTION	PART NO.	POSTULATED DISABLED HEIGHT*
MOTOR CONTROL CENTERS			
	Sump Pumps and Cooling Water Pumps to Recirc. Pump MG-Set Fluid Coupler	1R24-11D1 1R24-12D1	1'-6" 1'-6"
TANKS			
	Floor Drain Sump Tank	1G11*TK-050A,B 1G11*TK-056A-C	----- -----
	Drywell Floor Drain Receiver	1G11*TK-057	-----
	Salt Water Drain Tank	1G11*TK-190	-----
	Drywell Equip. Drain Receiver	1G11*TK-049	-----
HEAT EXCHANGER			
	HPCI Barometric Con. Vacuum Tank	1E41*E-036	-----
	RCIC Barometric Con. Tank	1E51*E-038	-----
	RHR Heat Exchanger	1E114*E-034A,B	-----
	RBCLCW Heat Exchangers	1P42*-011A,B	-----
	Drywell Equip. Drain Cooler	1G11*E-094	-----

Table 3.2.1 (Continued)
MAJOR ELEVATION 8 EQUIPMENT LIST

EQUIP. TYPE	EQUIPMENT DESCRIPTION	PART NO.	POSTULATED DISABLED HEIGHT+
ELEC. PANELS			
	** RCIC Instr. Rack	1H21*PNL-017	2'-0"
	** RCIC Instr. Rack	1H21*PNL-037	2'-0"
	** Core Spray Rack	1H21*PNL-01	3'-10"
	** Core Spray Rack	1H21*PNL-019	3'-10"
ELEC. PANELS			
	** RHR Inst. Rack A	1H21*PNL-018	3'-10"
	** RHR Inst. Rack B	1H21*PNL-021	3'-10"
	** HPCI Inst. Rack A	1H21*PNL-036	1'-10"
	** HPCI Inst. Rack B	1H21*PNL-14	1'-10"

** Equipment required for operation of the identified system.
+ Heights are taken from a physical survey measurement from the bottom of the component to floor level.
---- Non-electrical component

Table 3.2.2 Times to Flood Depth of 3'-10", 1'-10", and 1'-3" in Reactor Building

System	Water Source	Leakage Location	Time (min.) to Flood Depth of 3'-10"	1'-10"	1'-3"
HPCI	S.P.	pump suction (max.)	17	7.9	5.4
	S.P.	pump suction (large	34	15.8	10.8
	CST	pump suction (max.)	13	6.4	4.4
	CST	pump suction (large)	27	12.8	8.7
	---	pump discharge	37	17.5	11.9
RCIC	S.P.	pump suction (max.)	110.0	50.8	34.6
	S.P.	pump suction (large)	220.0	101.6	69.3
	CST	pump suction (max.)	76.0	36.3	24.8
	CST	pump suction (large)	152.0	72.6	49.5
LPCI	S.P.	pump suction (max.)	9.4	4.5	3.1
	S.P.	pump suction (large)	19.0	9.0	6.1
	---	pump discharge	15	7.3	5.0
CS	S.P.	pump suction (max.)	12	5.9	4.0
	S.P.	pump suction (large)	24	11.8	8.1
	CST	pump suction (max.)	13	6.4	4.4
	CST	pump suction (large)	27	12.8	8.7
	---	pump discharge	23	11.1	7.6
SW	SW	RHR heat exchanger	20	9.5	6.5
WFPS	WFPS	rupture of 8" pipe	40	19.1	13.0

Note: 1. Large flow rates is 1/2 of maximum flow rates.
2. Flood times were calulated based on a 41,600 gallons per foot depth in the reactor building.
3. S.P. = Suppression Pool
CST = Condensate Storage Tank
SW = Service Water System
WFPS = Water Fire Protection System Tanks

Table 3.2.3 Human Error Probability: Screening Values

Problem-solving

Time	Nominal Value	Error Factor
<1 min.	1	---
10 min.	5E-1	5
20 min.	1E-1	10
30 min.	1E-2	10
60 min.	1E-3	10
1500 min.	1E-4	30

Procedural Errors

Nominal Value	Error Factor
1E-3 (With Recovery)	3
1E-2 (Without Recovery)	3

Table 3.2.4 HEP (Event A) Single Alarm Condition Manual Shutdown (NUREG/CR-1278)

	1'-3"	1'-10"	3'-10"
TFL1	10^{-3}	10^{-3}	2.0×10^{-4}
TFL2	1	1	0.1
TFL3	1	1	0.1
TFL4	1	1	1
TFL5	1	1	10^{-2}
TFL6	0.1	0.1	10^{-3}
TFL7	1	0.1	10^{-2}
TFL8	1	1	0.1
TFL9	1	1	10^{-2}
TFL10	0.1	0.1	10^{-3}
TFL11	10^{-3}	10^{-3}	2×10^{-4}
TFL12	1	1	0.1
TFL13	0.1	0.1	10^{-3}
TFL14	1	1	0.1
TFL15	1	0.1	10^{-2}
TFL16	1	1	1
TFL17	1	1	0.1

Table 3.2.5 HEP (Event A), Multiple Alarm Condition (Nominal Value, PRA Procedures Guide)

	1'-3"	1'-10"	3'-10"
TFL1	10^{-2}	10^{-2}	10^{-3}
TFL2	1	1	0.5
TFL3	1	1	0.5
TFL4	1	1	1
TFL5	1	1	0.1
TFL6	0.5	0.5	10^{-2}
TFL7	1	0.5	0.1
TFL8	1	1	0.5
TFL9	1	1	0.1
TFL10	0.5	0.5	10^{-2}
TFL11	10^{-2}	10^{-2}	10^{-3}
TFL12	1	1	0.5
TFL13	0.5	0.5	10^{-2}
TFL14	1	1	0.5
TFL15	1	0.5	0.1
TFL16	1	1	1
TFL17	1	1	0.5

Table 3.3.1 Vital Equipment Locations at Elevation 8

1'	HPCI vac. pump cond. pump RCIC vac. pump cond. pump	1'-3"
1'-3"	CS loop level pump	
1'-4"	RCIC loop pump	
1'-6"	recir. pump M-G set	1'-10"
1'-10"	HPCI instrumentation	
2'	RCIC instrumehtation	
2'-3"	HPCI loop level pump	
3'-10"	RHR instrumentation CS instrumentation	3'-10"

Table 3.3.2 Conditional Frequency of Core Vulnerable
(1 of 2)

		Phase I	Phase II	Phase III	Phase IV
TFL1	Manual	5.8(-7)	2.7(-6)	2.0(-5)	7.3(-3)
	MSIV	3.2(-6)	8.7(-5)	8.5(-4)	1.2(-1)
	TT	7.7(-7)	2.2(-5)	2.1(-4)	3.3(-2)
TFL2	Manual	5.8(-7)	2.2(-6)	2.0(-5)	7.3(-3)
	MSIV	3.2(-6)	6.8(-5)	8.5(-4)	1.2(-1)
TFL3	Manual	5.8(-7)	1.1(-6)	2.2(-5)	7.3(-3)
	MSIV	3.2(-6)	1.1(-5)	9.5(-4)	1.2(-1)
TFL4	Manual	5.8(-7)	3.9(-4)	5.2(-4)	7.3(-3)
	MSIV	3.2(-6)	2.0(-2)	2.6(-2)	1.2(-1)
TFL5	Manual	5.8(-7)	3.9(-4)	5.2(-4)	7.3(-3)
	MSIV	3.2(-6)	2.0(-2)	2.6(-2)	1.2(-1)
TFL6	Manual	5.8(-7)	2.2(-6)	2.0(-5)	7.3(-3)
	MSIV	3.2(-6)	6.8(-5)	8.5(-4)	1.2(-1)
	TT	7.7(-7)	1.6(-5)	2.1(-4)	3.3(-2)
TFL7	Manual	5.8(-7)	1.1(-6)	2.2(-5)	7.3(-3)
	MSIV	3.2(-6)	1.1(-5)	9.5(-4)	1.2(-1)
	TT	7.7(-7)	3.2(-6)	2.3(-4)	3.3(-2)
TFL8	Manual	5.8(-7)	3.9(-4)	5.2(-4)	7.3(-3)
	MSIV	3.2(-6)	2.0(-2)	2.6(-2)	1.2(-1)
	TT	7.7(-7)	4.7(-3)	6.2(-3)	3.3(-2)
TFL9	Manual	5.8(-7)	3.9(-4)	5.2(-4)	7.3(-3)
	MSIV	3.2(-6)	2.0(-2)	2.6(-2)	1.2(-1)
	TT	7.7(-7)	4.7(-3)	6.2(-3)	3.3(-2)
TFL10	Manual	5.8(-7)	1.1(-6)	2.2(-5)	7.3(-3)
	MSIV	3.2(-6)	1.0(-5)	9.5(-4)	1.2(-1)
	TT	7.7(-7)	3.2(-6)	2.3(-4)	3.3(-2)
TFL11	Same as TFL1				
TFL12	Same as TFL6				
TFL13	Same as TFL6				

Table 3.3.2 Conditional Frequency of Core Vulnerable
(2 of 2)

		Phase I	Phase II	Phase III	Phase IV
TFL14	Manual	5.8(-7)	1.1(-6)	2.2(-5)	7.3(-3)
	MSIV	3.2(-6)	1.1(-5)	9.5(-4)	1.2(-1)
	TT	7.7(-7)	3.2(-6)	2.3(-4)	3.3(-2)
TFL15	Same as TFL14				
TFL16	Same as TFL8				
TFL17	Same as TFL8				

Table 3.3.3 Core Vulnerable Frequency
(1 of 2)

		P-1	P-2	P-3	P-4	TOTAL
TFL1	Man.	7.3(-11)	0	1.7(-11)	1.2(-9)	
	MSIV	4.5(-11)	0	8.2(-11)	1.3(-8)	
		1.2(-10)	0	9.9(-11)	1.5(-8)	1.4(-8)
TFL2	Man.	0	0	9.1(-10)	2.1(-7)	
	MSIV	0	0	3.9(-8)	3.4(-6)	
		0	0	4.0(-8)	3.6(-6)	3.7(-6)
TFL3	Man.	0	0	1.2(-10)	3.5(-8)	
	MSIV	0	0	5.2(-9)	5.8(-7)	
		0	0	5.3(-9)	6.1(-7)	6.2(-7)
TFL4	Man.	0	0	0	1.5(-7)	
	MSIV	0	0	0	2.5(-6)	
		0	0	0	2.7(-6)	2.7(-6)
TFL5	Man.	0	0	4.9(-9)	6.9(-9)	
	MSIV	0	0	2.5(-7)	1.1(-7)	
		0	0	2.5(-7)	1.1(-7)	3.7(-7)
TFL6	Man.	*	0	*	1.6(-10)	
	MSIV	*	0	1.4(-8)	5.6(-8)	
		*	0	1.4(-8)	5.6(-8)	7.0(-8)
TFL7	Man.	0	*	*	1.4(-9)	
	MSIV	0	3.9(-10)	2.6(-8)	8.3(-7)	
		0	3.9(-10)	2.6(-8)	8.3(-7)	8.6(-7)
TFL8	Man.	0	0	1.5(-8)	2.3(-8)	
	MSIV	0	0	1.6(-6)	4.3(-6)	
	TT	0	0	2.3(-7)	1.2(-6)	
		0	0	1.8(-6)	5.5(-6)	7.3(-6)
TFL9	Man.	0	0	6.5(-9)	1.2(-9)	
	MSIV	0	0	9.2(-7)	3.4(-7)	
	TT	0	0	1.8(-8)	9.9(-8)	
		0	0	9.4(-7)	4.4(-7)	1.4(-6)

*Less than 1.0(-10).

Table 3.3.3 Core Vulnerable Frequency
(2 of 2)

		P-1	P-2	P-3	P-4	TOTAL
TFL10	Man.	*	0	*	*	
	MSIV	*	0	3.9(-9)	1.4(-8)	
	TT	*	0	9.4(-10)	3.6(-9)	
		*	0	4.8(-9)	1.8(-8)	2.2(-8)
TFL11	Man.	*	0	*	*	
	MSIV	*	0	1.6(-10)	1.4(-8)	
	TT	*	0	*	3.2(-9)	
		*	0	1.6(-10)	1.7(-8)	1.7(-8)
TFL12	Man.	0	0	*	1.1(-9)	
	MSIV	0	0	4.5(-9)	4.9(-7)	
		0	0	4.5(-9)	4.9(-7)	5.0(-7)
TFL13	Man.	*	0	*	*	
	MSIV	*	0	6.6(-9)	2.5(-8)	3.1(-8)
TFL14	Man.	0	0	*	1.8(-9)	
	MSIV	0	0	7.3(-9)	6.9(-7)	
		0	0	7.3(-9)	6.9(-7)	7.0(-7)
TFL15	Man.	0	*	*	4.6(-10)	
	MSIV	0	*	8.4(-9)	2.6(-7)	
		0	*	8.4(-9)	2.6(-7)	2.8(-7)
TFL16	Man.	0	0	0	1.8(-8)	
	MSIV	0	0	0	9.2(-7)	
	TT	0	0	0	1.9(-7)	
		0	0	0	1.1(-6)	1.1(-6)
TFL17	Man.	0	0	2.3(-9)	3.8(-9)	
	MSIV	0	0	2.5(-7)	6.6(-7)	
	TT	0	0	3.7(-8)	2.0(-7)	
		0	0	2.9(-7)	8.6(-7)	1.2(-6)

*Less than 1.0(-10).

4.0 SUMMARY

BNL reviewed the internal flood analysis which is a part of the Shoreham PRA and found that assumptions, methodology, and results are reasonable. BNL re-evaluated the flood precursor frequency using recent LER data and a more accurate methodology. This methodology avoids some of the conservatisms in the SNPS-PRA approach. A slight increase in the initiator frequency is calculated because of the revised data.

Similarly, based on the PSA Procedure Guide, the HEP was reviewed and only minimal changes were made to the Shoreham HEP values used in the analysis. As for the functional event trees, a time phase approach was adopted to better model the progression of the flood events.

Results are summarized in Table 4.1. This table can be divided into three parts. Part A provides a comparison between the Shoreham results and those obtained in the BNL review. The BNL value is about 5 times that of the Shoreham frequency, 2.0(-5) vs. 3.9(-6). The contributions from the different plant states are also presented. The major increase in the total core vulnerable frequency in the BNL analysis is attributable to the increase in flood precursor frequencies. Part B compares only the contributions from the BNL Phase IV results with the Shoreham values. It can be inferred that by neglecting the initial three phases, the core vulnerable frequency will be underestimated by 3×10^{-6} or about 18%. Part C shows the contributions of core vulnerable frequency for different plant states due to maintenance and rupture induced floods. In the Shoreham analysis 41% of the core vulnerable frequency is calculated to be caused by maintenance related floods while the BNL analysis shows 37%.

An uncertainty estimation has been carried out assuming lognormal distributions. An error factor of 10 was applied to the operator errors and the split ratio for the manual shutdown and the MSIV closure event following the Reactor Building flooding. The results of the uncertainty assessment for the core vulnerable frequency are as follows.

Mean	=	1.9E-5
Median	=	1.9E-6
5% Confidence	=	2.2E-7
95% Confidence	=	7.5E-5

Table 4.1 Summary of Core Vulnerable Frequency

	Shoreham	BNL
Part A		
Manual	8.5(-8)	4.8(-7)
MSIV	3.0(-6)	1.8(-5)
TT	7.7(-7)	2.0(-6)
Total	3.9(-6)	2.0(-5)

	Shoreham	BNL (only Phase IV
Part B		
Manual	8.5(-8)	4.5(-7)
MSIV	3.0(-6)	1.5(-5)
TT	7.7(-7)	1.7(-6)
Total	3.9(-6)	1.7(-5)

		Shoreham	BNL
Part C			
Manual	Maintenance	3.9(-8)	4.1(-7)
	Rupture	1.6(-7)	7.0(-8)
MSIV	Maintenance	1.5(-6)	6.9(-6)
	Rupture	1.4(-6)	1.1(-5)
TT	Maintenance	0	0
	Rupture	6.7(-7)	2.0(-6)
Total	Maintenance	1.6(-6)	7.3(-6)
	Rupture	2.3(-6)	1.3(-5)

REFERENCES

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7. "Probabilistic Safety Analysis Procedures Guide," NUREG/CR-2815, September 1983.
8. "Nuclear Power Experience," Division of Petroleum Information Corporation, Denver, Colorado, August 1981.

MAY 9 1984

Docket No. 50-322

MEMORANDUM FOR: Albert Schwencer, Chief
Licensing Branch #2
Division of Licensing

FROM: Ashok Thadani, Chief
Reliabiity and Risk Assessment Branch
Division of Safety Technology

SUBJECT: SHOREHAM FLOODING

Reference: (1) Memorandum dated March 30, 1984, from A. Thadani to A. Schwencer, "Shoreham Flooding."

In Reference 1 we transmitted our findings on the Shoreham flooding issue. These findings were based on the Brookhaven National Laboratory (BNL) evaluation provided to us in a draft report. The purpose of this memorandum is to transmit to you the final BNL report on this issue. Our conclusions reported to you in Reference 1 still remain valid.

Ashok Thadani, Chief
Reliability and Risk Assessment Branch
Division of Safety Technology

Enclosure:
BNL Final Report on Shoreham Flooding

cc: H. Denton
R. Mattson
D. Eisenhut
T. Speis
E. Chelliah
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ENCLOSURE 1

LETTER REPORT ON THE REVIEW OF THE SEQUENCES FOLLOWING A RELEASE OF EXCESSIVE WATER IN ELEVATION 8 OF THE REACTOR BUILDING IN THE SHOREHAM NUCLEAR POWER STATION

K. Shiu
Y. Sun
E. Anavim
I. A. Papazoglou

Risk Evaluation Group
Department of Nuclear Energy
Brookhaven National Laboratory
Upton, New York 11973

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ABSTRACT

The core vulnerable risk resulted from Reactor Building flooding events is addressed as a part of the SNPS PRA.[1] The analysis was reviewed and re-evaluated at BNL and the results are presented in this report. The BNL review includes both qualitative and quantitative analyses of flooding initiators, operator errors, and accident sequences which result in a vulnerable core state. An estimate of the uncertainty for the core vulnerable risk is also included.

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1.0 INTRODUCTION

At the Shoreham Nuclear Power Station (SNPS) the majority of safety-related equipment are located in the Reactor Building (RB). The Shoreham Reactor Building is a cylindrical building surrounding the MARK II containment structure. Water leakage from equipment in the reactor building will drain to Elevation 8 (the lowest level of the RB) via openings and stairwells since there is no structural separation between safety systems. Flooding of the Elevation 8 compartment may potentially disable all the ECCS because they are located in the Elevation 8 compartment.

The SNPS-PRA(1) has included flooding as a common-mode event which may disable the ECCS equipment. The SNPS PRA assumes that a critical flooding depth of 3'-10" from the RB floor will disable all the ECCS equipment. Operator diagnosis and isolation of the flooding before it reaches 3'-10" depth is considered in SNPS-PRA.

Because of the potentially significant impact, the SNPS's evaluation of the core melt risk due to RB flooding warrants a special review. A field trip to the Shoreham plant has been made by BNL personnel for obtaining detailed information on the equipment and power control layouts in the RB, especially in the Elevation 8 compartment. BNL has determined that there are three flooding depths (1'-3", 1'-10", and 3'-10") that are critical to the availability of various ECCS equipment. The initiator event trees are thus revised accordingly.

BNL also identified that the random failure of a equipment protection circuit breaker coinsiding with the RB flood event may cause the propagation of failures to equipment powered by separated Motor Control Centers (MCC). This potential common mode failure event has also been modeled in BNL event trees.

Shoreham Plant Procedure Guides relevant to the RB flooding have been reviewed by BNL. BNL found that these procedure guides fail to require a systematic check of system parameter indicators in the control room following a RB Flooding Alarm annunciation. This may cause the operator to ignore an abnormal system parameter, especially under a multiple alarm situation (such as a turbine trip).

BNL's revised event trees, quantitative evaluation of core vulnerable risk due to RB flooding events, and an uncertainty estimate for the core vulnerable risk are presented in this report.

The report is organized as follows: Section 2 summarizes the SNPS-PRA approach to the flood sequence identifications and quantification. Section 3 presents the BNL revision both in the methodology and in the quantification. Finally, Section 4.0 summarizes the results.

2.0 SNPS METHODOLOGY AND ANALYSIS

2.1 Overview

The SNPS methodology for determining the contribution to the risk of the internal floods can be divided into three steps.

1. Identification of water sources and pathways to Elevation 8 compartment.
2. Evaluation of operators responses and assessment of likelihood of arresting the flood.
3. Evaluation of system responses and identification of the sequences leading to a core vulnerable state given a flood.

In the Shoreham PRA approach it was determined that flooding at locations other than Elevation 8 would be bounded by the analysis of flooding at the lowest level of the reactor building Elevation 8, since the flood water will drain and cascade down to that level through stairwells and openings. All the evaluations of flood are hence focused on equipment at the Elevation 8 level.

The volume of water required to flood the reactor building Elevation 8 compartment, with all equipment and piping installed, is estimated to be 41,600 gallons in SNPS-PRA for each foot of depth. The following drainage systems are available to receive the initial volume of flood water:

- Reactor Building Floor Sumps
- Reactor Building Equipment Sumps
- Reactor Building Porous Concrete Sumps.

These systems have total sump capacity of 4,650 gallons, and total sump pump capacity of 640 gallons per minute, however, they are not included in the analysis.

The potential water sources which may release excessive water in Elevation 8 are summarized in Table 2.1.1. For each of these sources, a pathway investigation has been performed in the SNPS-PRA, to define the potential for flood at Elevation 8. Table 2.1.2 summarizes the water sources as evaluated in the Shoreham PRA. For each water source the largest possible flow rate has been determined and the time required for the flood to reach the 3'-10" level in Elevation 8, have been estimated. These times are also given in Table 2.1.2. These times provide the basis for estimating the probability of successful prevention of flood at the 3'-10" level by operator actions.

A survey of all vital equipment by Shoreham identified a number of components for the various accident mitigation systems which could potentially be submerged in the event of an internal flood. Based on this information, the critical height of 3'-10" was defined. It was assumed that if flood water exceeds the 3'-10" level, all ECCS equipment would be disabled. Flooding scenarios which are arrested before reaching the 3'-10" level, have been found to contribute negligibly in the core damage frequency.

Functional event trees were used in the Shoreham internal flood PRA to model the plant response given an internal flood initiator. The flood initiator frequency was calculated based on two types of internal flood precursors: online maintenance and rupture of piping, valves or pumps. These precursor frequencies are described in Section 2.2. Given the occurrence of these flood precursors, the progression of events was modeled using initiator event trees. Details of the initiator event trees are presented in Section 2.3.

Since all the ECCS systems are assumed lost given a 3'-10" flood, the only available means for cooling the core are the feedwater and the condensate pump injection. The availability of these two systems depends on the state of the MSIVs and on the ultimate source of the flood (condensate storage tank or suppression pool).

Because of these dependences, the end states of the initiator event trees were classified into six categories each of which becomes the entry condition for the functional event trees. Table 2.1.3 summarizes the information in a matrix form. Each row of the matrix depicts one of the 17 types of internal flood precursors, the columns represent the six entry conditions to the functional event trees. The six entry conditions can be grouped into manual shutdown, turbine trip and MSIV closure. Two possible entry conditions are considered for each of these three initiators: flooding due to water from the condensate storage tank (CST) and flooding due to water from other sources.

Based on these six entry conditions, six functional event trees were developed. An example is given in Figure 2.1.1.

2.2 SNPS-PRA Quantification of the Frequency of Flood Initiators

Two types of flood initiators were considered in the SNPS-PRA.

1. Floods initiated by an accidental loss of isolation (valve opening) while a component in the Elevation 8 area is dismantled for maintenance.

2. Floods initiated by a rupture in the pressurized or the non-pressurized part of the piping.

2.2.1 Maintenance-induced Flood Initiators

The frequency of the first type of initiator was calculated by estimating the frequency of maintenance of various components based on operating experience data. The LER data base in Ref.2 identifies the observed failures from turbine-driven and motor-driven pump failures. The data used in the SNPS-PRA are summarized in Table 2.2.1. There are four failure modes for pumps, i.e., leakage/rupture, does not start, loss of function, and does not continue to run. The hourly LER failure rates characterize the leakage/rupture failure mode, while demand failure rates consider other failure modes.

The following LER rates are found for the four failure modes in motor-driven and turbine-driven standby pumps.

Motor Driven Pumps

- Leakage/rupture: 6 events/6,777,627 hrs. = 8.9×10^{-7}/hr.
- Does not start, loss of function, and does not continue to run: (5+4+6) events/(13,644 demands) = 1.1×10^{-3}/demand

SNPS-PRA assumed that these pumps are in standby status until there is a demand. The number of demand used in SNPS-PRA are 12 on the average per year (four scheduled tests plus eight other occurrences). Hence, the maintenance frequency for motor driven standby pumps per year is calculated as

$(8.9\times10^{-7}$ failure/hr)*(24 hr/day)*(365 day/yr) +
$(1.1\times10^{-3}$/demand)*(12 demands/yr) = 2.0×10^{-2} failure/year.

Turbine Driven Pump

Similiarly, the maintenance frequency for turbine driven standby pumps per year is calculated as 0.079 failure/year.

There are two motor driven pumps associated with the Core Spray System, four motor driven pumps with the LPCI System, and four motor driven pumps associated with the Service Water System in which two are linked as a pair to the RHR Heat Exchanger System. There is only one turbine driven pump associated with the HPCI System and one with the RCIC System. Table 2.2.2 summarizes the SNPS-PRA frequencies associated with major maintenance operations based upon the above evaluation and a conservative estimate of heat exchanger online maintenance.

2.2.2 Rupture-Induced Flood Initiators

The frequencies of the initiators caused by loss of system integrity from breaks or ruptures were derived from WASH-1400 failure rates of major components involving external leak and external ruptures, based on assumptions made in NUREG/CR-1363 (Reference 3). This information has been summarized in Table 2.2.3.

The calculation of each initiator is done by identifying the appropriate type and length of piping and number of components susceptible to rupture and summing the estimated yearly rupture rates. As an example, the total number of valves involved in the HPCI discharge system are 3 (2 MOV's and 1 Check Valve); there is no pump involved (Table 2.2.5) and the total length of piping is 76'. Referring to Table 2.2.3, the rupture failure rate for 100' of pipe section is 4.3×10^{-11}/hr, and for external failure of a valve is

1.3×10^{-9}/hr. The total length of pipe in the HPCI Discharge System is estimated to be 76' (Table 2.2.5).

$$(3 \text{ valves})*(1.3\times10^{-9}/\text{hr}) + 76'/100' \, (4.3\times10^{-11}/\text{hr})$$
$$= 3.9\times10^{-9}/\text{hr or } 3.5\times10^{-5}/\text{yr}.$$

Since the flow rates through suction line breaks are time dependent (i.e., a function of the varying water head in the source) and a strong function of the break shape and size, a simplified model based on historical experience and engineering judgement is used in the Shoreham PRA to describe the conditional probability of break size. Table 2.2.4 summarizes the classes of break size examined.

These probabilities, are combined with the frequencies estimated for initiators associated with core spray, HPCI, RCIC, LPCI, and Service Water Rupture/Leak Suction System failure to obtain the initiating event frequencies for non-pressurized piping. Table 2.2.6 summarizes the frequencies of initiators due to the loss of system integrity from breaks or ruptures.

2.3 Initiator Event Trees

The probability of causing a flood due to component under maintenance or the probability of not arresting the flood is calculated with the help of initiator Event Trees. These trees are shown in Figures 2.3.1 through 2.3.17. A discussion of the P, D, E, I, and A events in the event trees follows.

a. Event P - Operator removes power from equipment and valves.
   The removal of power from equipment and its isolation valves is a required procedure during a maintenance in both fossil and nuclear power stations. The equipment and isolation valves are electrically disconnected from their associated power supply by pulling and tagging the appropriate breaker at the MCC. A second qualified person verifies the correct implementation of the tagging order and placement of the clearance tags.

   A human error probability (HEP) of 0.01 is assigned for this operator action. This value is determined using the probability data given in NUREG/CR-1278[4] (p.20-23).

b. Event D - System not demanded.

During the maintenance process there is a possibility that the safety systems will be demanded because of a transient challenge. Isolation valves will automatically open if the operator has failed to remove power from the isolation valves (Event P).

c. Event E - Operator maintains isolation.

During on-line maintenance with the equipment disassembled, the isolation valves need to be maintained in closed position throughout the duration of the maintenance process. However, an operator error could inadvertently open isolation valves.

SNPS concludes that it is unlikely that the operator will manually open these valves locally in the RB and fail to notice the flood. Opening of the isolation valves at the MCC is also concluded by SNPS to be unlikely.

The remaining possibility is that the valve is opened from the control room (given Event P). The panel switch could be activated by three events. These events are: the operator mistakenly operates the switch; a command fault to the valve; or the operator inadvertently operates the switch. The probabilities for these events are 10^{-3}, 10^{-4}, and 10^{-2}, respectively.

d. Event I - Flood annunciation.

The excessive water in reactor building is annunciated by alarms in the control room. The probability of the operator to fail to notice the alarm (the light is in a "back" panel) is assessed at 10^{-3}.

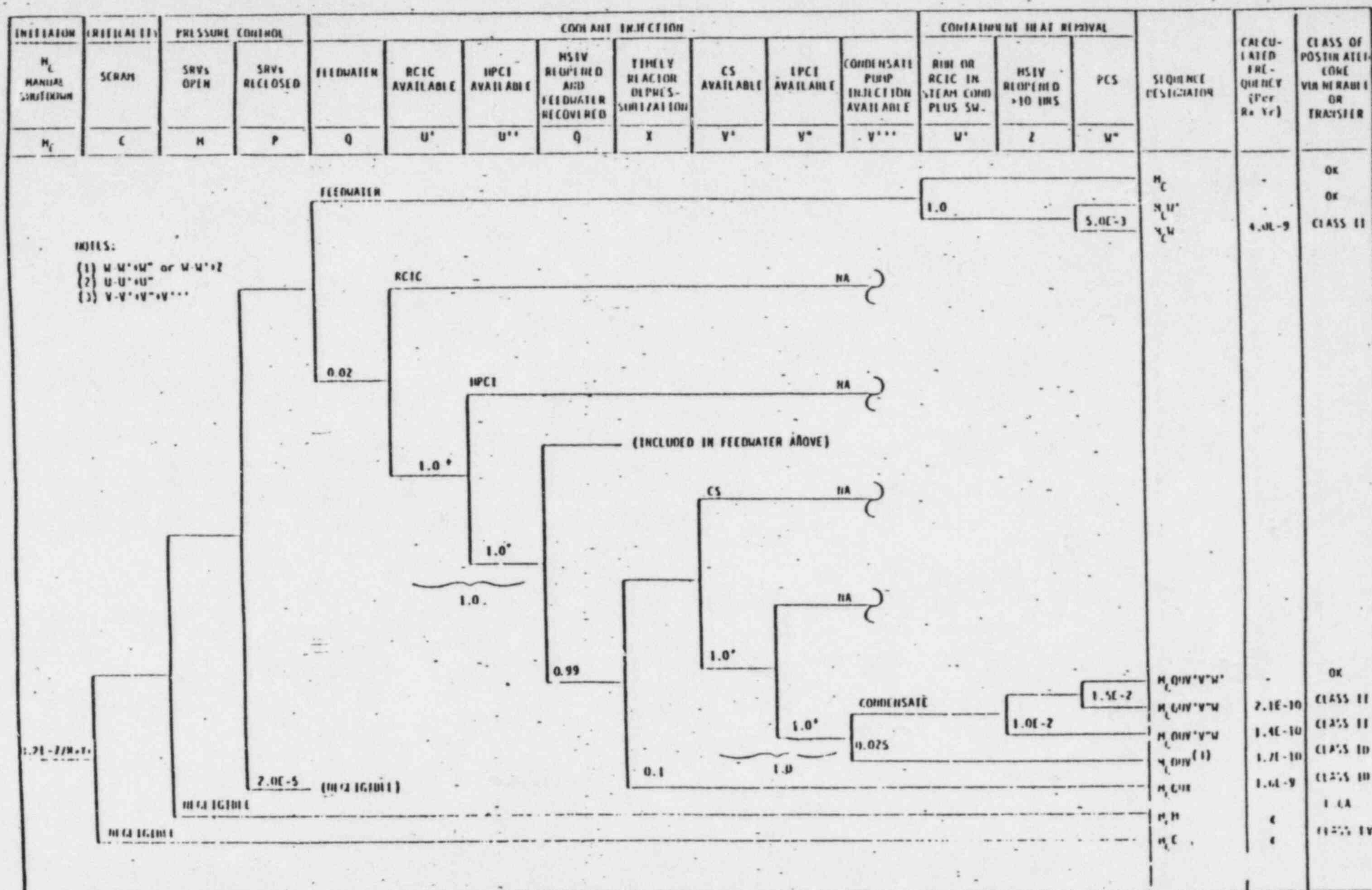
e. Event A - Operator diagnoses and responds to isolate the flood.

The operator must identify the source of and isolate the flood before it reaches the 3"-10" level. This event is considered by SNPS under two conditions as follows.

1. Operator isolates flood after auto occurrence, e.g., turbine trip or MSIV closure (Event A_A). Multiple alarms will occur in the control room at the same time as the flood alarm.

2. Operator isolates flood after manual occurrence, e.g., power operation or manual shutdown (Event A_M). Only the flood related alarms will annunciate in the control room.

The HEP data provided in NUREG/CR-1278[4] (1982 Edition, Chapter 12) are applied by SNPS for their evaluation. Figure 2.3.18 and Table 2.3.1 show the time varying cumulative HEP for both the single and the multiple occurrence conditions.



*In maintenance or flooded.

Figure 2.1.1 System Event Tree for Manual Shutdowns With Greater Than 3'-10" of Water in the Reactor Building (Source = CST).

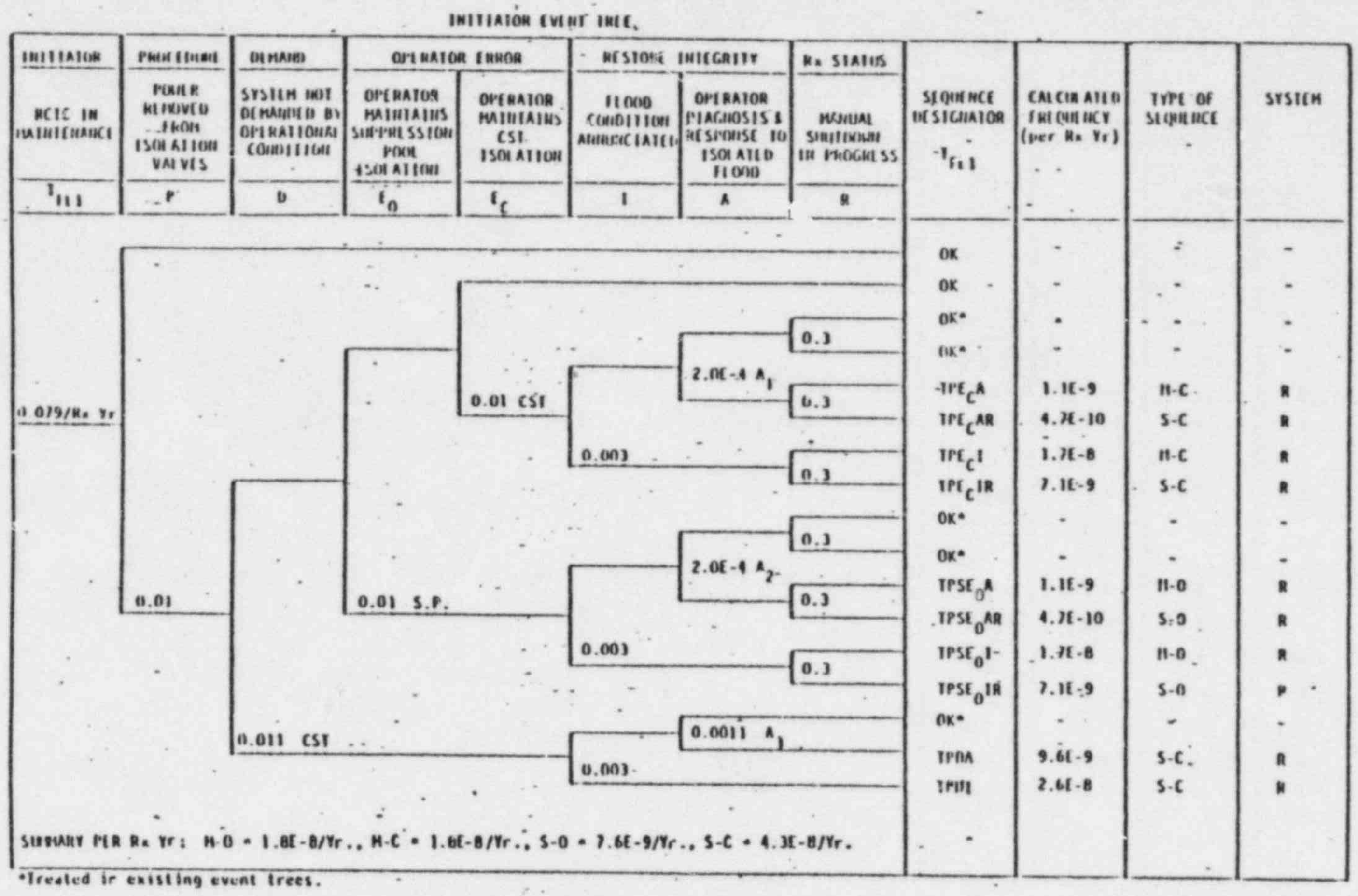


Figure 2.3.1 T_{FL1}: Initiator Event Tree for Postulated Flooding Sequences Initiated During RCIC Maintenance

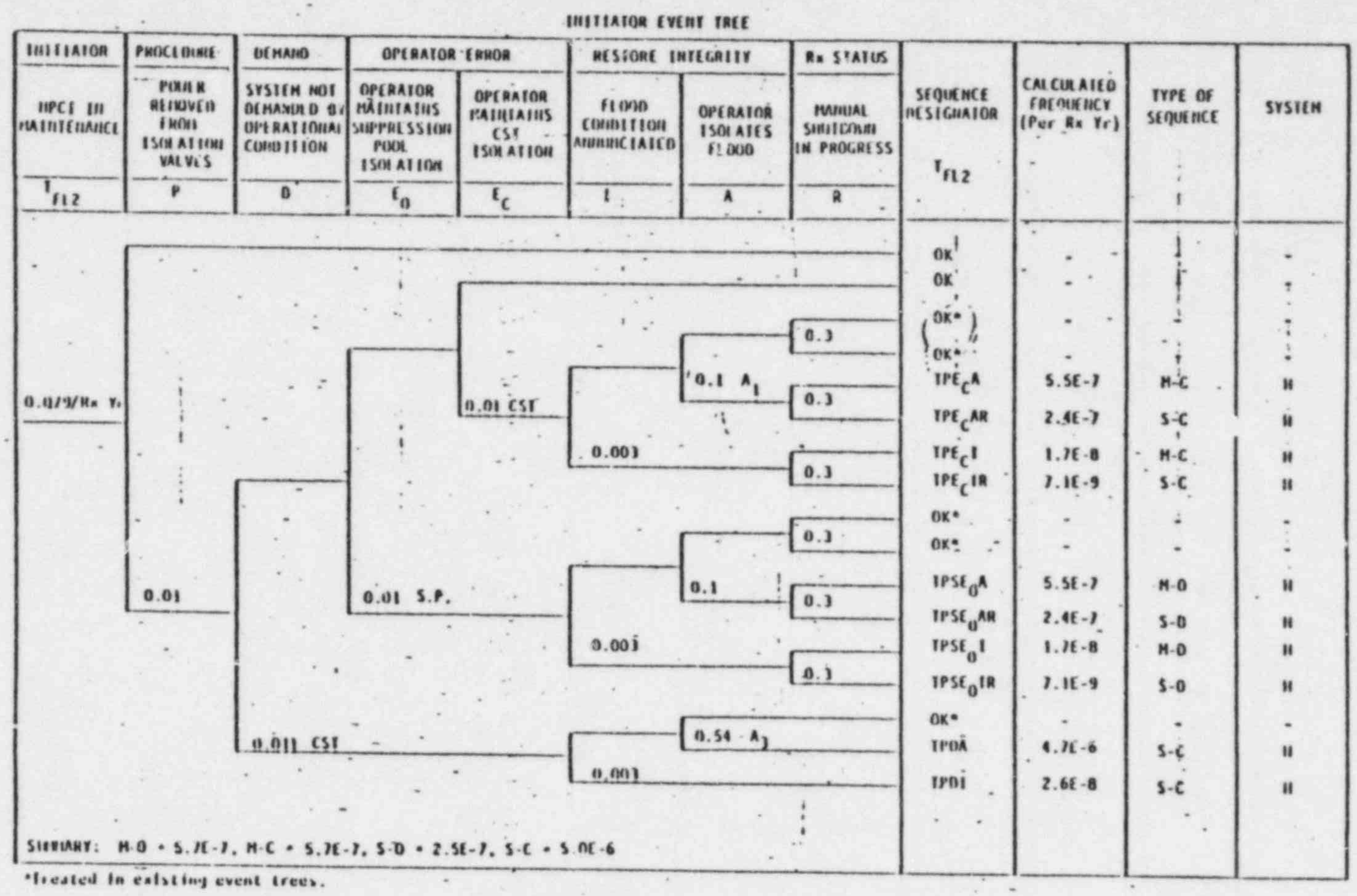


Figure 2.3.2 T_{FL2} - Initiator Event Tree for Postulated Flooding Sequences Initiated by an Error During HPCI Major Maintenance

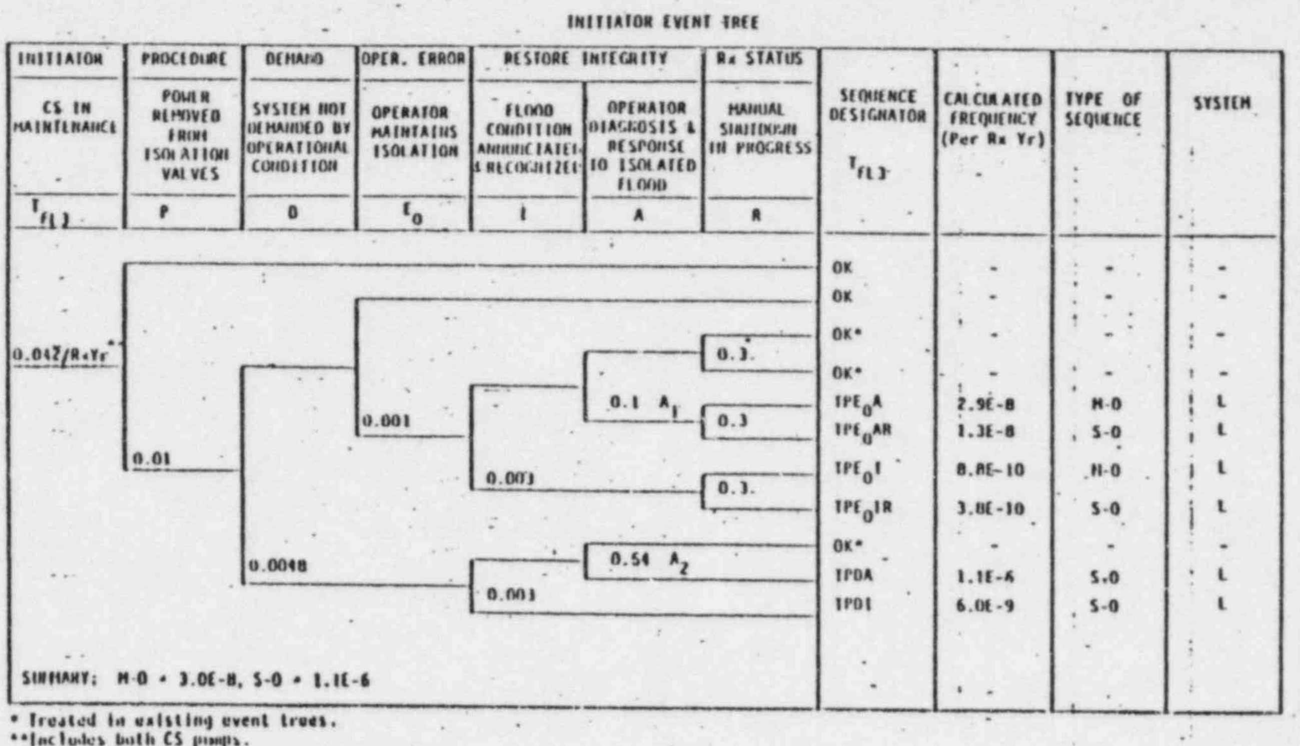
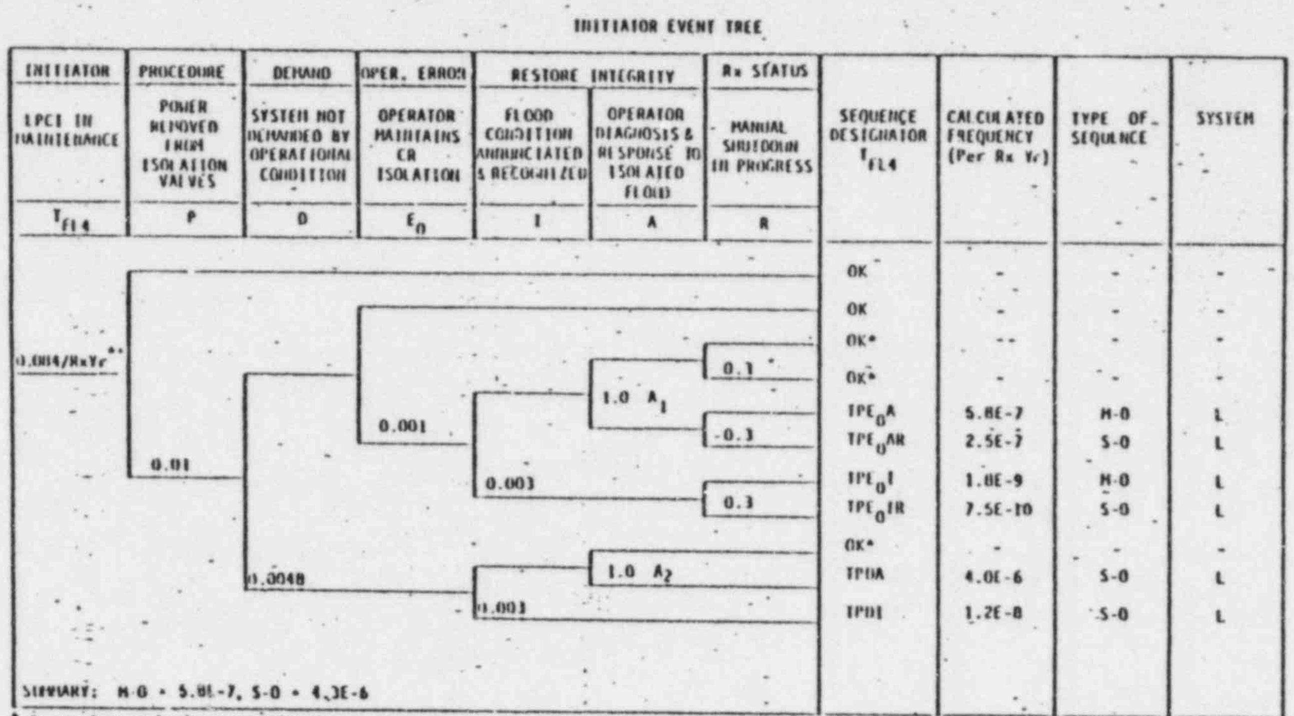


Figure 2.3.3 T_{FL3} - Initiator Event Tree for Postulated Flooding Sequences Initiated by an Error During Core Spray Major Maintenance



* Treated in existing event trees.
**Includes four LPCI pumps.

Figure 2.3.4 T_{FL4} - Initiator Event Tree for Postulated Flooding Sequences Initiated by an Error During LPCI Major Maintenance

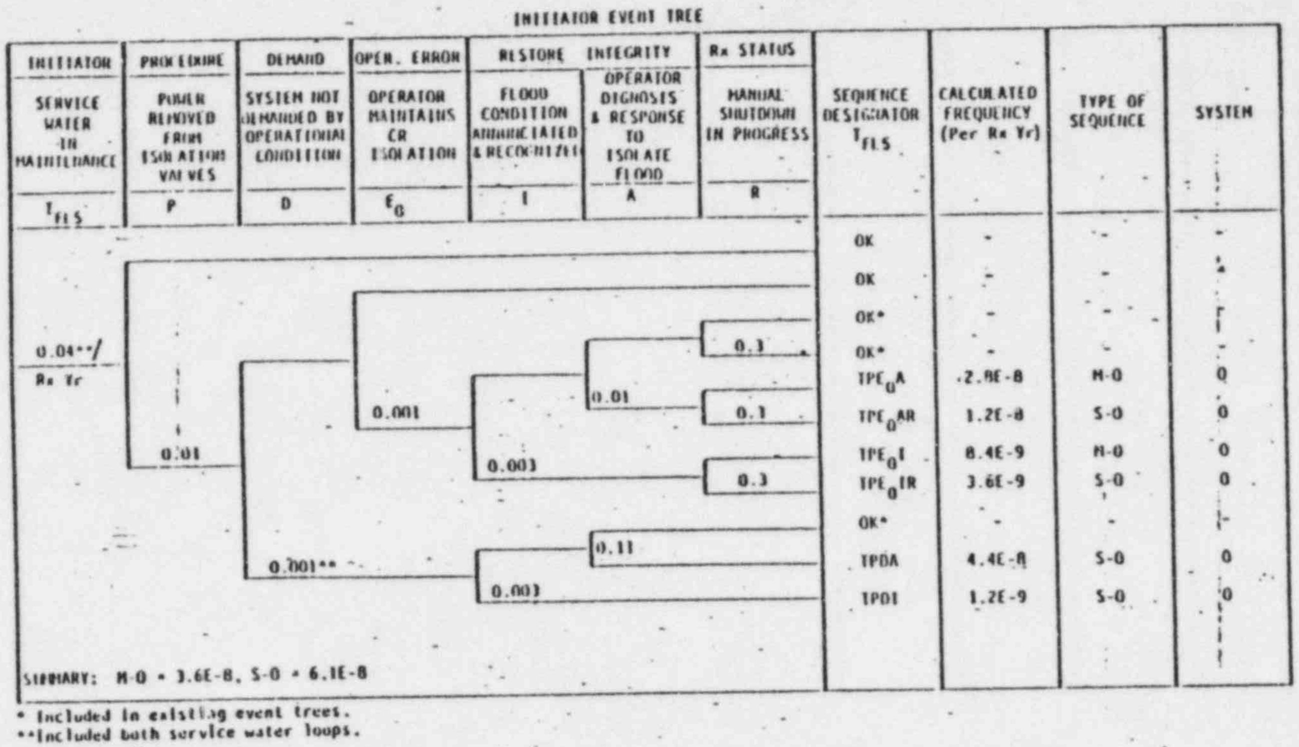
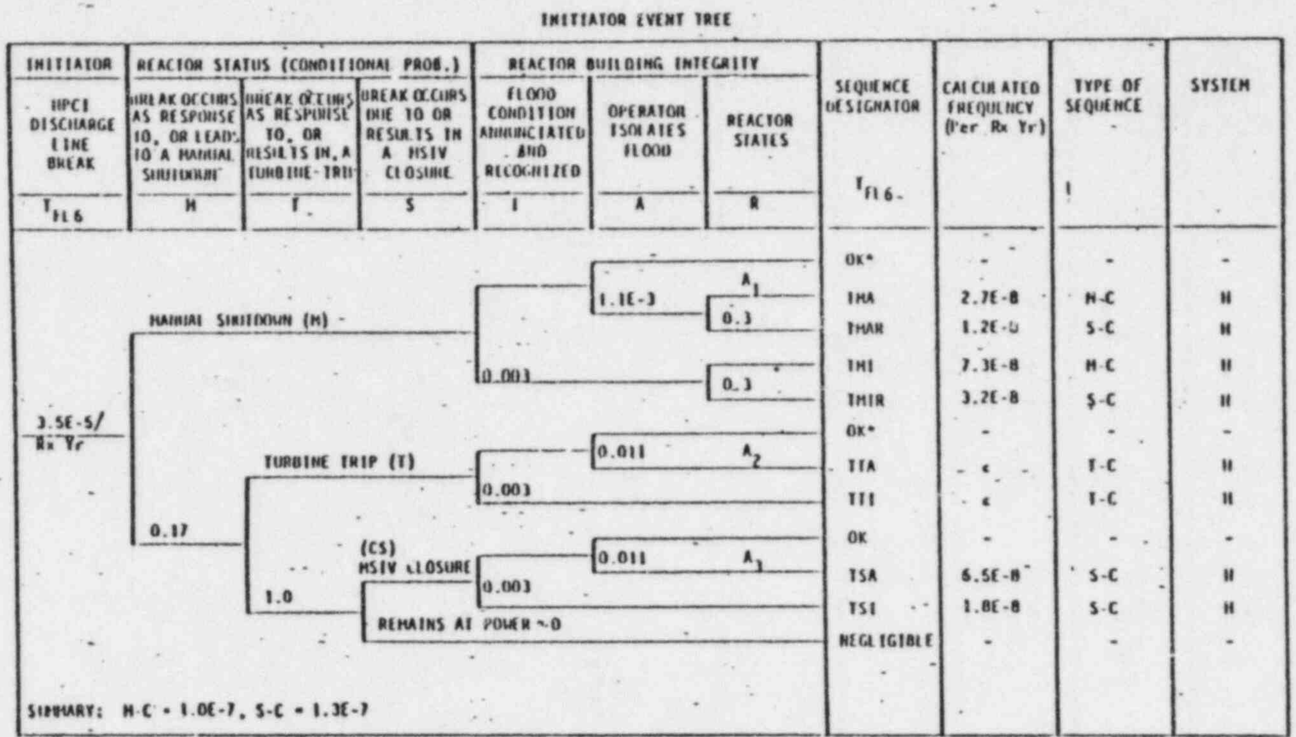
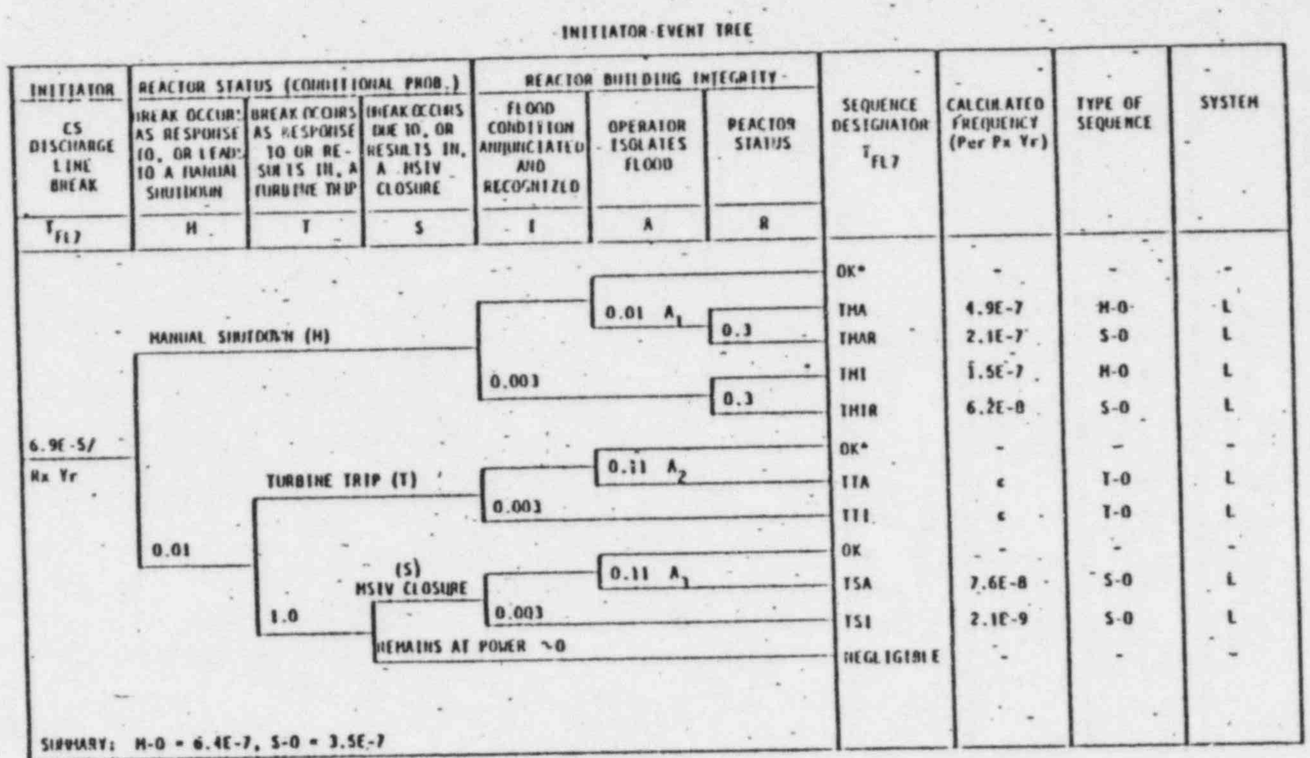


Figure 2.3.5 T_{FL5} - Initiator Event Tree for Postulated Flooding Sequences Initiated by an Error During Service Water Major Maintenance (i.e., Heat Exchangers)



*Included in the previously evaluated event tree.

Figure 2.3.6 Initiator Event Tree for Postulated Flooding Sequences Initiated by a HPCI Discharge Pipe Break



*Included in the previously evaluated event tree.

Figure 2.3.7 Initiator Event Tree for Postulated Flooding Sequences Initiated by a CS Discharge Pipe Break

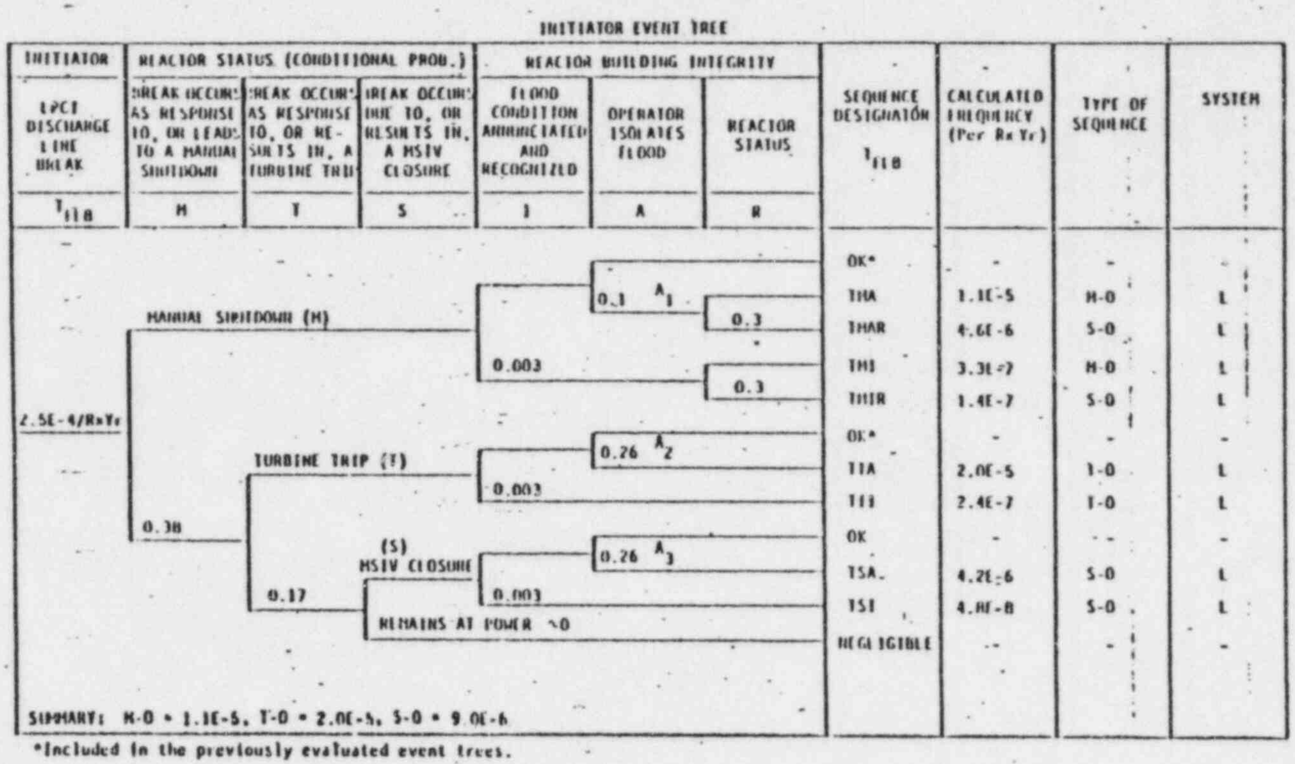
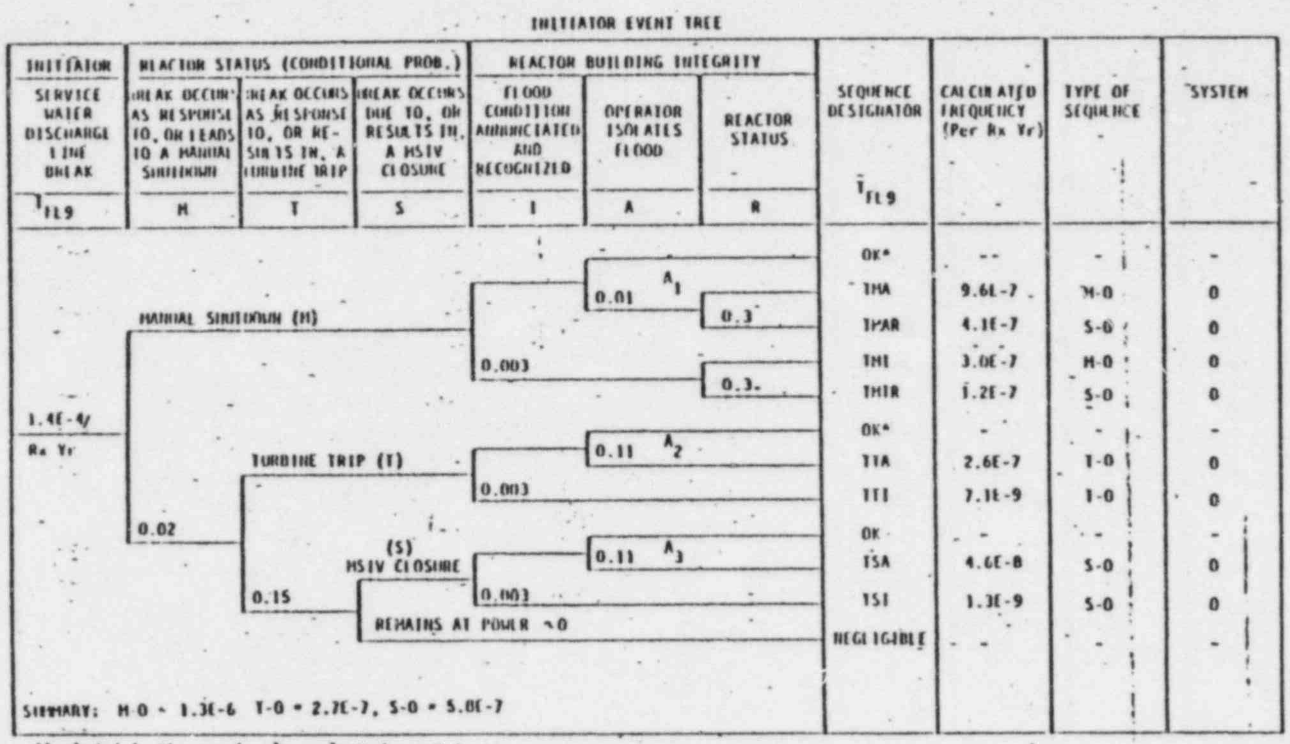


Figure 2.3.8 Initiator Event Trees for Postulated Flooding Sequences Initiated by a LPGI Discharge Pipe Break



*Included in the previously evaluated event trees.

Figure 2.3.9 Initiator Event Tree for Postulated Flooding Sequences Initiated by a Service Water Line Break

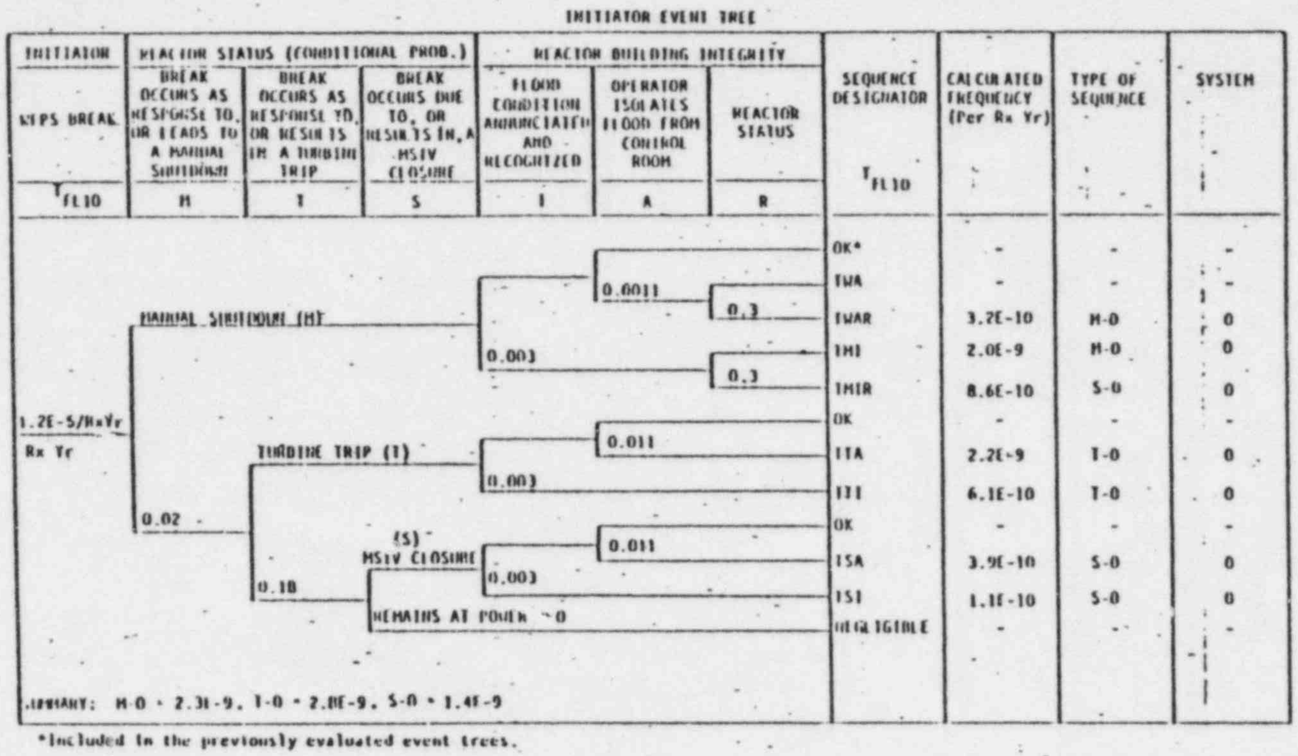
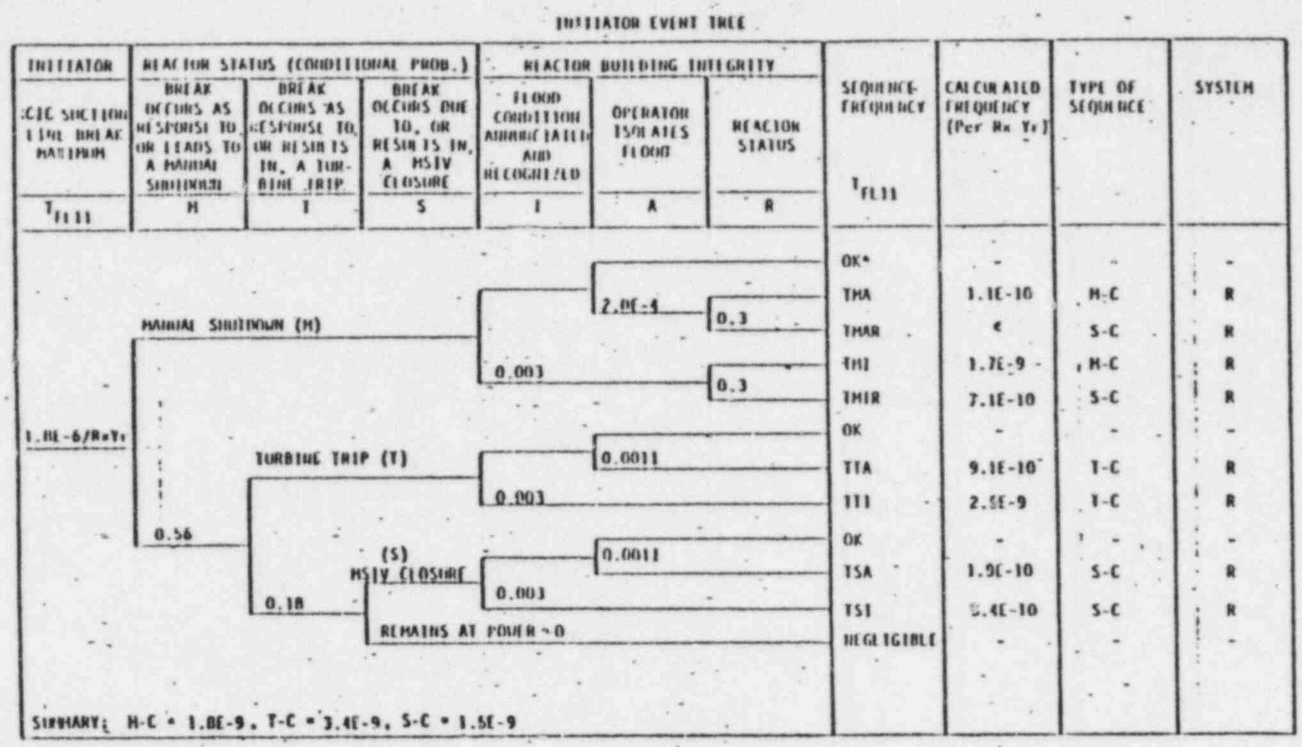
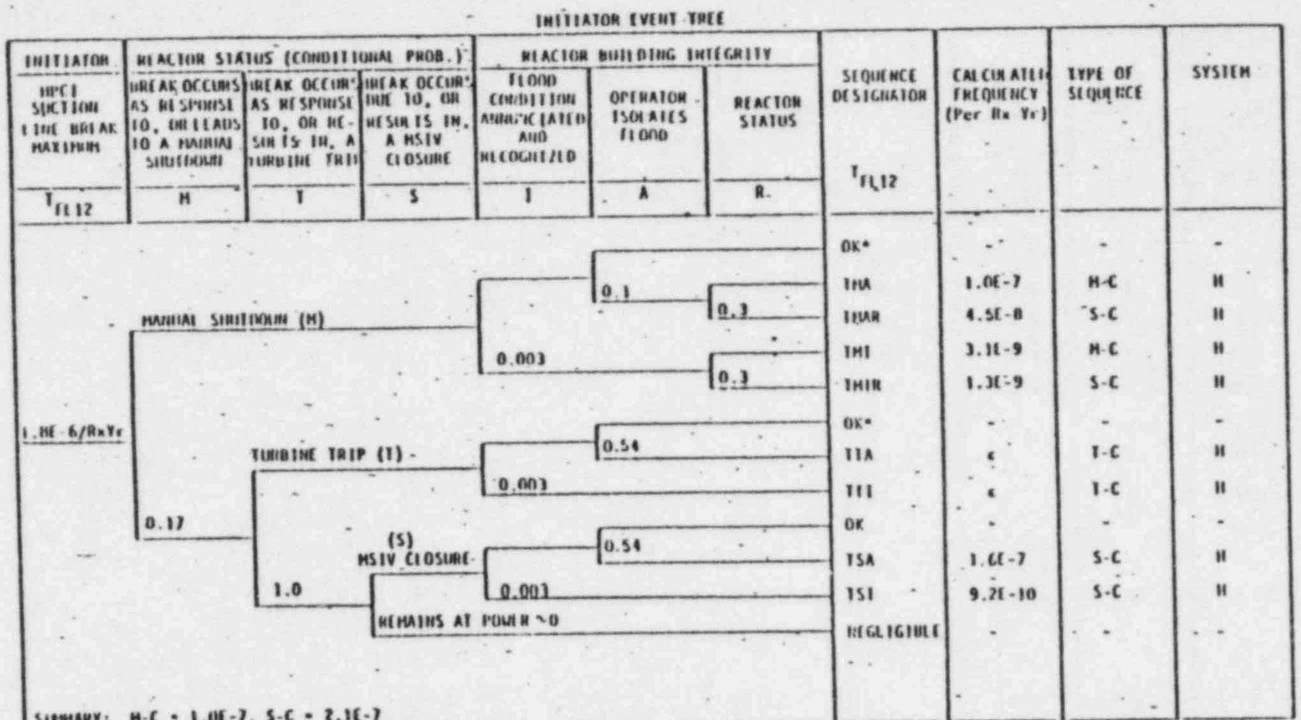


Figure 2.3.10 Initiator Event Tree for Postulated Flooding Sequences Initiated by a WFPS Break.



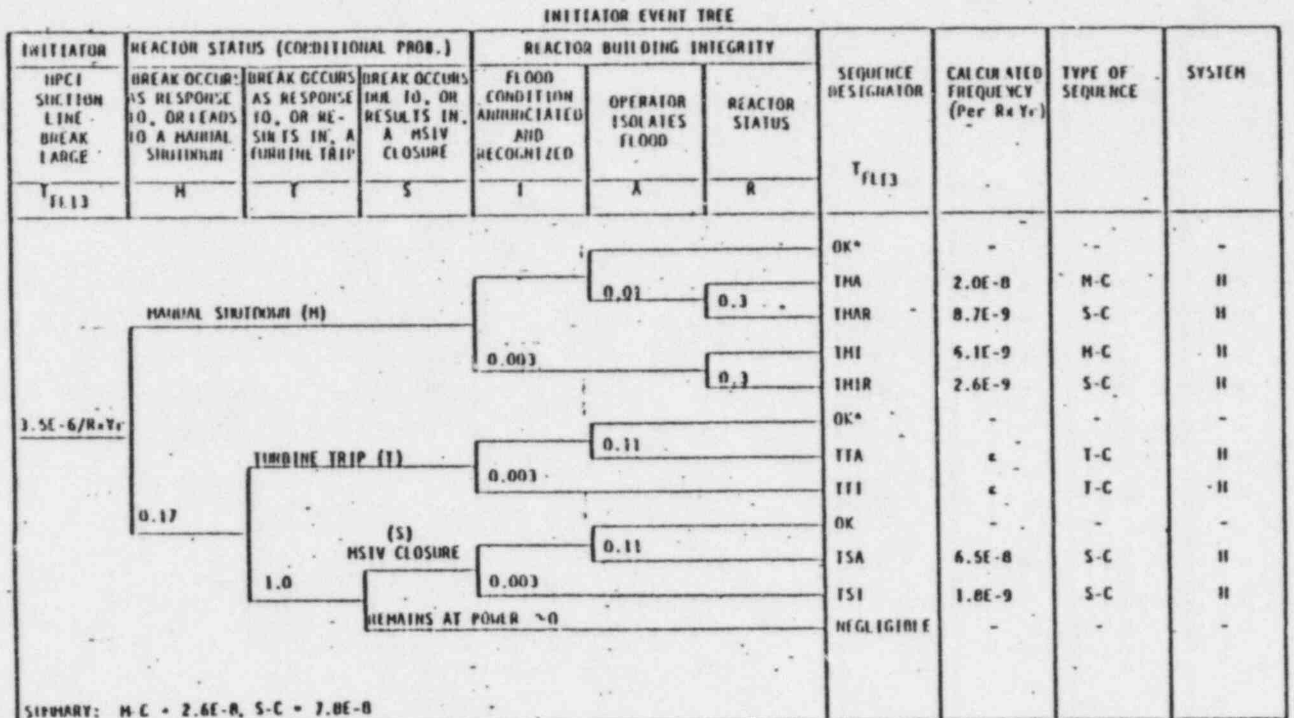
*Included in the previously evaluated event trees.

Figure 2.3.11 Initiator Event Tree for Postulated Flooding Sequences Initiated by a Maximum RCIC Suction Line Break



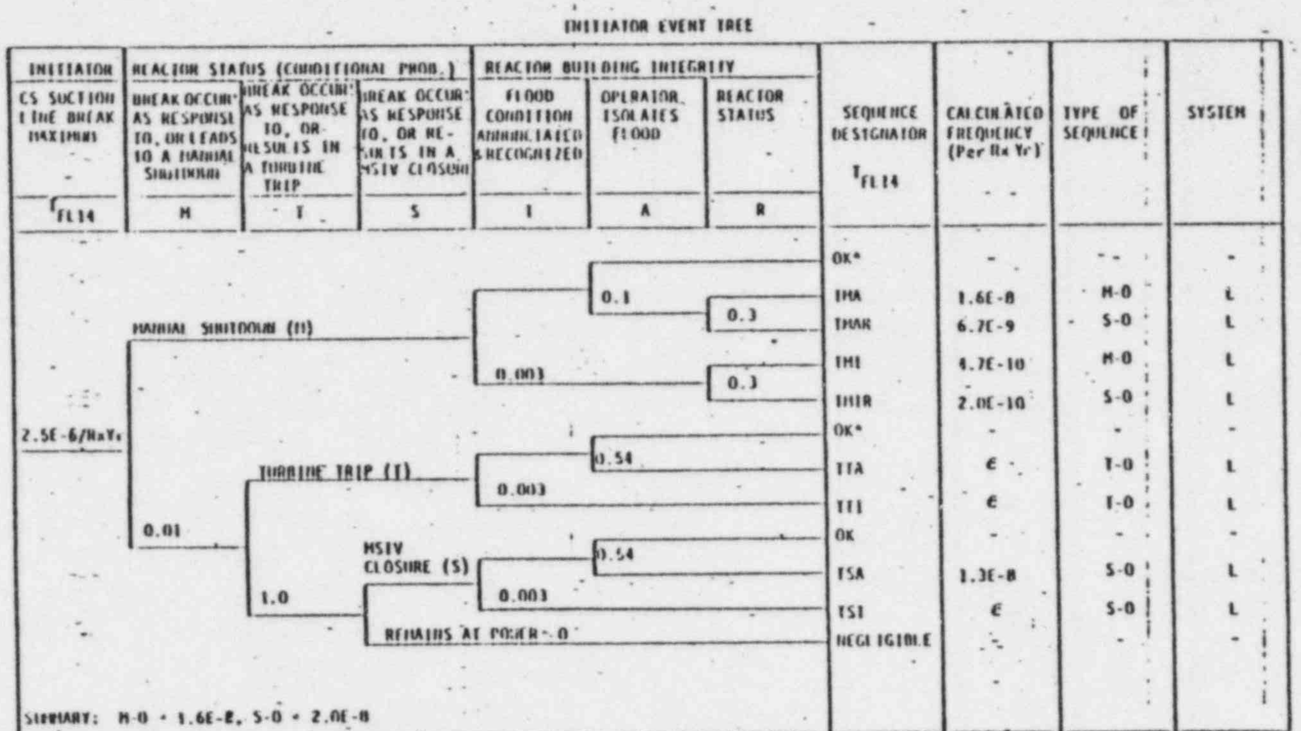
*Included in the previously evaluated event trees.

Figure 2.3.12 Initiator Event Tree for Postulated Flooding Sequences Initiated by a Maximum HPCI Suction Line Break



*Included in the previously evaluated event trees.

Figure 2.3.13 Initiator Event Tree for Postulated Flooding Sequences Initiated by a Large HPCI Suction Line Break



*Included in the previously evaluated event trees.

Figure 2.3.14 Initiator Event Tree for Postulated Flooding Sequences Initiated by a Maximum Core Spray Suction Line Break

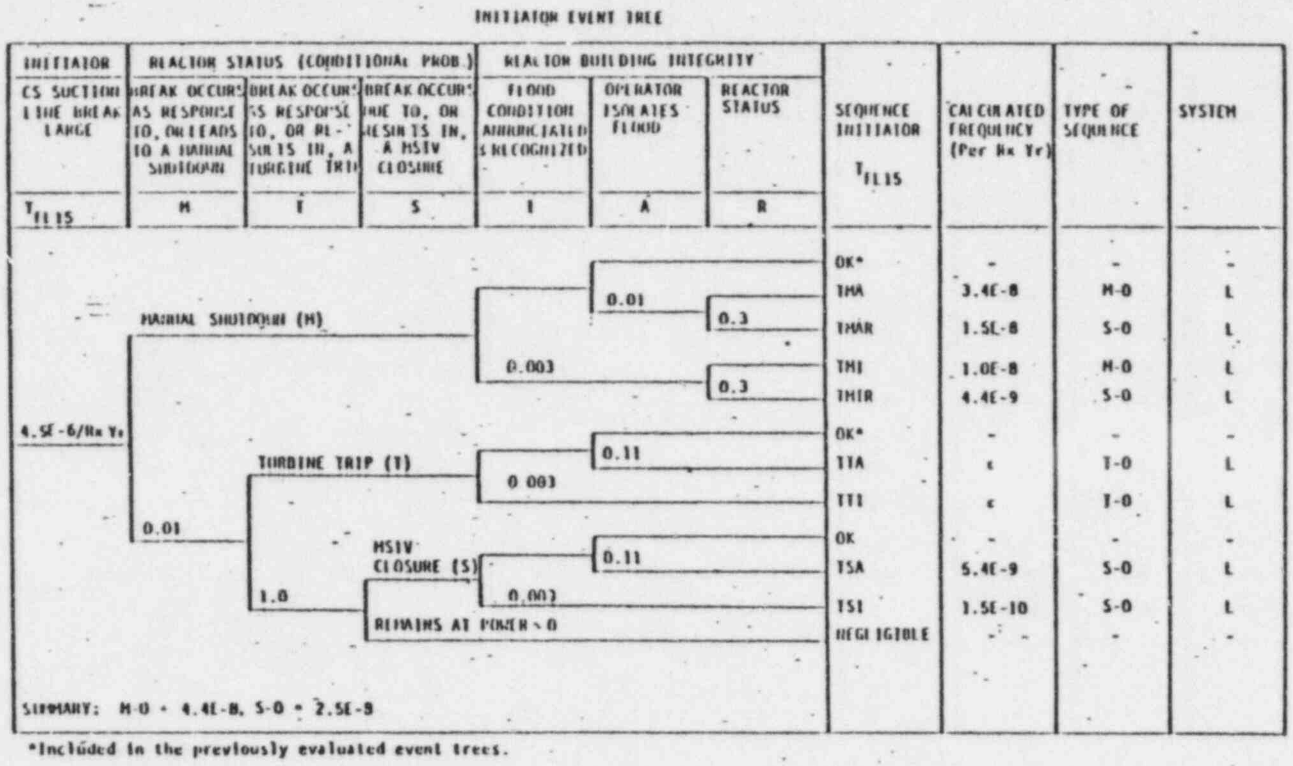
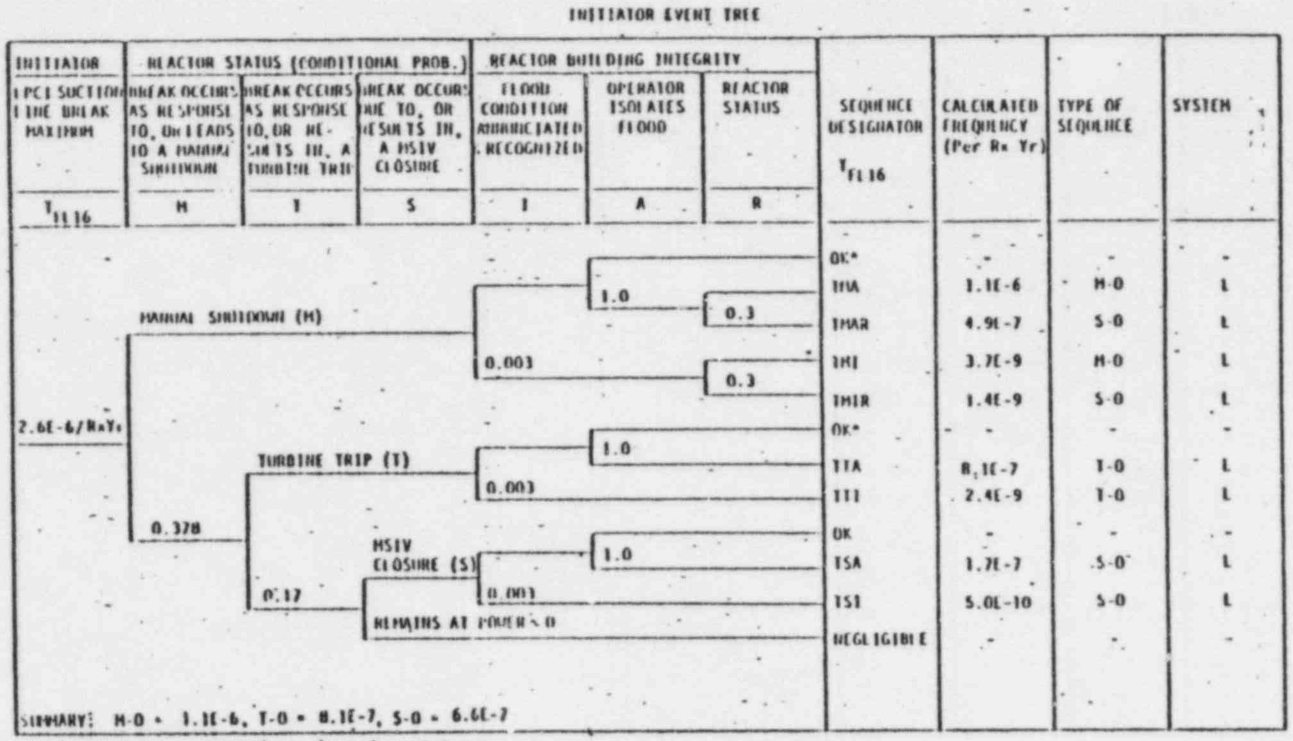


Figure 2.3.15 Initiator Event Tree for Postulated Flooding Sequences Initiated by a Large Core Spray Suction Line Failure



*Included in the previously evaluated event trees.

Figure 2.3.16 Initiator Event Tree for Postulated Flooding Sequences Initiated by a Maximum LPCI Suction Line Break

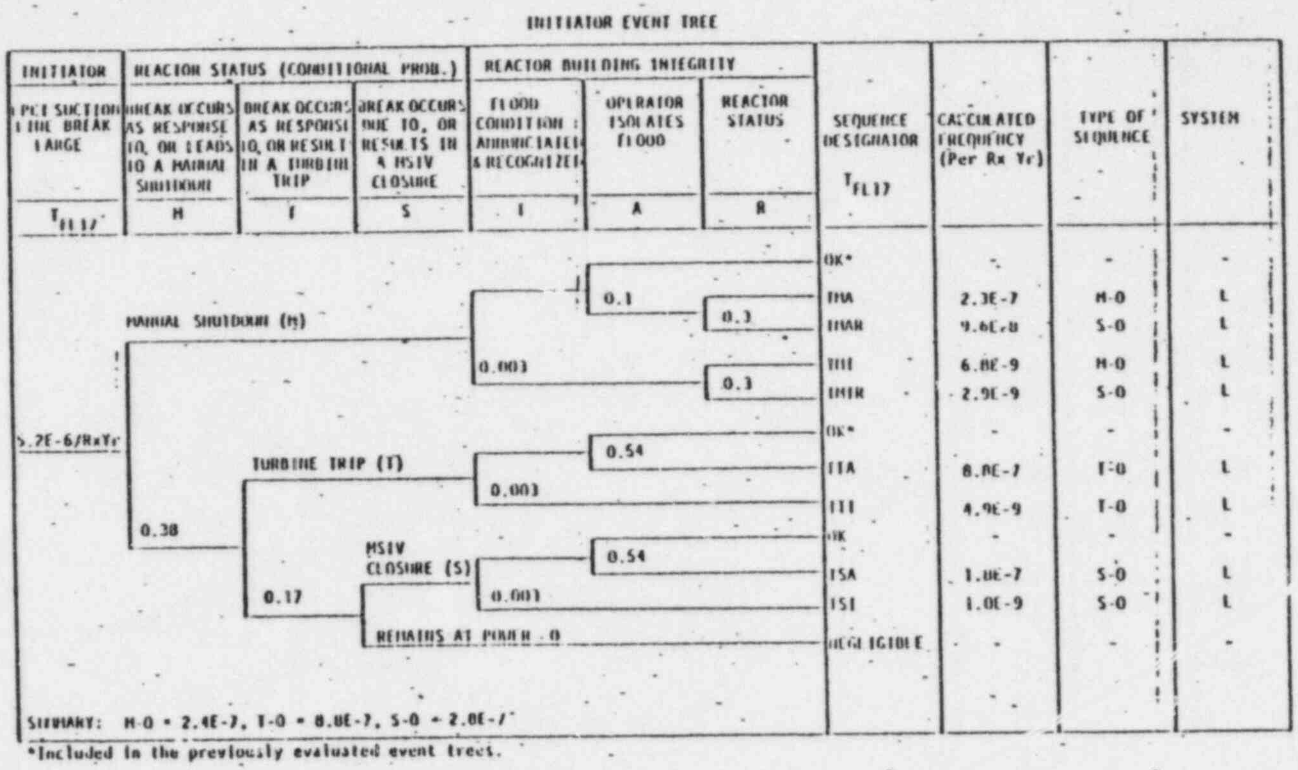


Figure 2.3.17 Initiator Event Tree for Postulated Flooding Sequences Initiated by a Large LPCI Suction Line Break

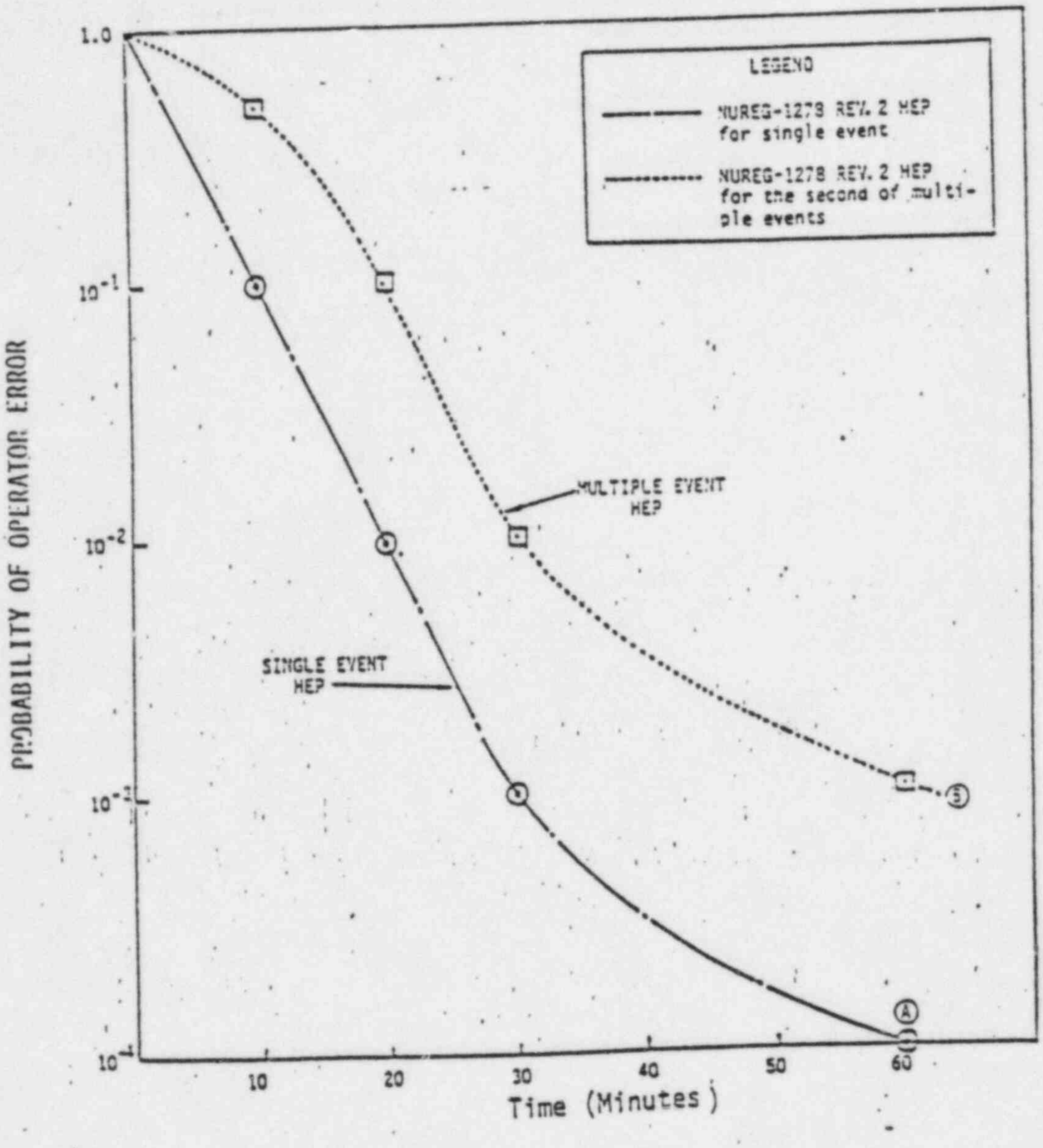


Figure 2.3.18 Comparison of the HEPs Associated with Operator Actions for Singular Events and Coincident Multiple Events

Table 2.1.1 Summary of Potential Water Sources and Types of Initiators Which may Lead to Release of Excessive Water in the Elevation 8 Compartment

Source	Quantity (Gallons)	No. of Lines	Systems Involved
Suppression Pool	160,000*	8	CS,LPCI,RCIC,HPCI
Condensate Storage Tank (CST)	550,000	4	CS,HPCI,RCIC
Reactor Primary System**	a) 42,928 b) 152,928		
Screenwell (Long Island Sound)	Unlimited	4	Service Water
Water Fire Protection System Storage Tank	600,000	Many	Fire Main

*Total water volume in the suppression pool at the high water level mark is 608,500 gallons. However, only a portion of the water can be drained through ECCS pump suction piping.

**Figure (a) includes water from the bottom of the core to normal water level in the RPV. Figure (b) includes (a) plus condenser hotwell water.

Table 2.1.2 Summary of Internal Flooding Initiator Types: Source, Pathway, Flowrates, and Time to Critical Flooding Depth

Source	Location	Flow Rate gpm*	Elevation 8 Flooding Time (Minutes*) 3'-10"
Suppression Pool	HPCI Pump Suction	9600	17.6
	RCIC Pump Suction	1500	10.6
	LPCI Pump Suction (Max/Large)**	17000/8500	9.4/19.0
	CS Pump Suction	13000	12.0
	LPCI Pump Suction	10500 (1 Pump Runout)	15.0
	CS Pump Discharge	6850 (1 Pump Runout)	23.0
Condensate Storage Tank (CST)	HPCI Pump Suction (Max/Large)**	1200/6000	13.0/27.0
	RCIC Pump Suction	2100	76.0
	CS Pump Suction (Max/Large)**	1200/6000	13.0/27.0
	HPCI Pump Discharge	4350 (Design)	37.0
Service Water	RHR Heat Exchanger	8000 (Pump Runout)	20.0
WFPS	Rupture of 8" Pipe	4000	40.0

*These flood times were calculated based on a failure of the sump pumps to successfully operate and a 41,600 gallon per foot depth in the reactor building given in the Shoreham FSAR.

**Large flow rates assumed to be 1/2 maximum flow.

Table 2.1.3 Summary of System Event Tree Entry States by Initiator Type

	SYSTEM EVENT TREE ENTRY CONDITION FREQUENCY (Per Rx Yr)					
INITIATOR	M-O	M-C	T-O	T-C	S-O	S-C
T_{FL1}	1.8×10^{-8}	1.8×10^{-8}			7.6×10^{-9}	4.3×10^{-8}
T_{FL2}	5.7×10^{-7}	5.7×10^{-7}			2.5×10^{-7}	5.0×10^{-6}
T_{FL3}	3.0×10^{-8}				1.1×10^{-6}	
T_{FL4}	5.8×10^{-7}				4.3×10^{-6}	
T_{FL5}	3.6×10^{-8}				6.1×10^{-8}	
T_{FL6}		1.0×10^{-7}				1.3×10^{-7}
T_{FL7}	6.4×10^{-7}				3.5×10^{-7}	
T_{FL8}	1.1×10^{-5}		2.0×10^{-5}		9.0×10^{-6}	
T_{FL9}	1.3×10^{-6}		2.7×10^{-7}		5.8×10^{-7}	
T_{FL10}	2.3×10^{-9}		2.8×10^{-9}		1.4×10^{-9}	
T_{FL11}		1.8×10^{-9}		3.4×10^{-9}		1.5×10^{-9}
T_{FL12}		1.0×10^{-7}				2.1×10^{-7}
T_{FL13}		2.6×10^{-8}				7.8×10^{-8}
T_{FL14}	1.6×10^{-8}				2.0×10^{-8}	
T_{FL15}	4.4×10^{-8}				2.5×10^{-8}	
T_{FL16}	1.1×10^{-6}		8.1×10^{-7}		6.6×10^{-7}	
T_{FL17}	2.4×10^{-7}		8.8×10^{-7}		2.8×10^{-7}	
TOTALS	1.6×10^{-5}	8.2×10^{-7}	2.2×10^{-5}	3.4×10^{-9}	1.7×10^{-5}	5.5×10^{-6}

Table 2.2.1 LER Data for BWR Standby Pumps for the Period of January 1972 Through April 1978

Standby Pumps	Demands	Standby Hours	Leakage Rupture	Does Not Start	Loss of Function	Does Not Continue To Run
Motor Driven	13,644	6,777,627	6	5	4	6
Turbine Driven	1,820	868,033	-	1	6	5

Table 2.2.2 Frequency of Online Major Maintenance System in the Reactor Building

System	Frequency (Per Year) SNSP-PRA	Initiator Event Tree
Core Spray (Motor Driven)	0.042	TFL3
LPCI (Motor Driven)	0.084	TFL4
HPCI (Turbine Driven)	0.079	TFL2
RCIC (Turbine Driven)	0.079	TFL1
Service Water (RHR or RBCLCW HX) (Motor Driven)	0.042	TFL5

Table 2.2.3 Summary of Failure Rates for Major Components Involving External Leak and External Rupture

Parameter Rate	Total Failure Rate/Hr (Mean)	Reference	Rupture* Failure Rate/Hr
Pipe Failure Section (100')	8.5E-10	WASH-1400	4.3E-11
External Failure of a Valve	2.7E-8	WASH-1400	1.3E-9
External Failure of a Pump	3.0E-9	WASH-1400	1.5E-10

*Based upon the operating experience to date, given that a failure occurs, the ratio of external leaks to complete failures appears to be in the range of 20 to 1. This is substantiated by the specific data review cited in the text for values (18 to 1) and data published by Bush(5) on pipes (4 to 1 up to 30 to 1). Because the internal flood evaluation is based upon initiators with substantial flooding rates, i.e., short operator response times. only the catastrophic or large external rupture failures are treated in this evaluation.

Table 2.2.4 Conditional Probability of Pipe Break Size

Break Size	Characterization	Flow Rate	Conditional Probability
Maximum	Guillotine Break	100%	0.05
Large	Substantial Rupture	50%	0.10
Small*	Localized Rupture in Ductile Material	13%	0.85

*Remainder of the conditional probability was allocated to small breaks.

Table 2.2.5 Initiating Event Frequency Estimates Involving Component Leak/Ruptures

INITIATOR	SOURCE	VALVES				PIPING LENGTH (FT)/ SECT/DIA (IN)	ESTIMATED FREQUENCY/ YR
		MOV	MAN	CHK	PUMPS		
HPCI Discharge T_{FL6}	CST/SUPP	2	0	1	0	76/1/14	3.5E-5
CS Discharge T_{FL7}	SUPP	4	0	2	0	128/2/12	6.9E-5
LPCI Discharge T_{FL8}	SUPP	14	4	4	0	240/6/16	2.5E-4
Service Water T_{FL9}	Service Water	4	4	4	0	715/8/10-20	1.4E-4
WFPS T_{FL10}	WFPS	1				157/2/6-8	1.1E-5
RCIC** Suction T_{FL11}	CST	1	1	1	1	70/1/6	3.5E-5
HPCI** Suction T_{FL12}, T_{FL13}	CST**	1	1	1	1	87/1/16	3.5E-5
CS** Suction T_{FL14}, T_{FL15}	CST*	2	2		2	120/2/12	4.9E-5
LPCI** Suction T_{FL16}, T_{FL17}	SUPP	4			4	120/2/20	5.2E-5

*CST is assumed to be the source.
**Suction failures are also classified by flow rate.

Table 2.2.6 Calculated Frequencies for Initiating Events Resulting from System Ruptures (SNPS-PRA)

Initiator	Frequency (Per RX Yr)
Pressurized Piping	
HPCI Discharge Break, TFL6	3.5×10^{-5}
CS Discharge Break, TFL7	6.9×10^{-5}
LPCI Discharge Break, TFL8	2.5×10^{-4}
SW Discharge Break, TFL9	1.4×10^{-4}
WFPS Discharge Break, TFL10	1.1×10^{-5}
Non-Pressurized Piping	
RCIC Suction Failure, TF11 (max)	1.75×10^{-6}*
HPCI Suction Failure, TF12 (max)	1.75×10^{-6}*
HPCI Suction Failure, TF13 (large)	3.5×10^{-6}*
CS Suction Failure, TF14 (max)	2.5×10^{-6}*
CS Suction Failure, TF15 (large)	4.9×10^{-6}*
LPCI Suction Failure, TF16 (max)	2.6×10^{-6}*
LPCI Suction Failure, TF17 (large)	5.2×10^{-6}*

*Modified based upon engineering judgement made on the size of low pressure suction line breaks.

Table 2.3.1

THE PROBABILITY THAT FLOOD REMAINS UNISOLATED FOR X MINUTES AFTER AUTOMATIC PLANT ACTION: E.G., TURBINE TRIP OR MSIV CLOSURE

X	P(for multiple event)	P(for single event)
1	1	1.0
10	1st + 2nd = 0.54	0.1
20	0.11	0.01
30	0.011	1.1E-3
60	0.0011	2.0E-4
1500	1.1E-4	1.1E-4

3.0 BNL ACCIDENT REVIEW AND SEQUENCE QUANTIFICATION

This section discusses the quantification and review of the internal flooding accident sequences in the SNPS-PRA due to system maintenance and pipe ruptures. The section is organized as follows. Subsection 3.1 presents a summary of the approach used by BNL to calculate the initiator frequencies. Subsection 3.2 discusses BNL quantitative review of the initiator event trees, and Subsection 3.3 presents the functional event tree analysis and evaluation.

3.1 Flood Precursor Frequency

This review revised the assessment of the frequency of the flood initiators in two ways. First the experiential data for the estimation of the various failure rates were revised to include recent events. Second, the models for calculating the frequency of floods (or probability per year of reactor operation) have been improved by removing unnecessary conservatisms. As it was already discussed in Section 2.2, two types of initiators were considered: a) maintenance-induced initiators; and b) rupture-induced initiators. The revised frequencies for these types of initiators are presented in the following two subsections.

3.1.1 Maintenance-Induced Flood Initiators

A flood can be initiated during the maintenance of a component of the ECCS or of another system in the Elevation 8 area if the maintenance process requires dismantling of the component and one of the isolation valves opens inadvertently while the component is being maintained.

The components that contribute to these initiators are the pumps and the heat exchangers in the Elevation 8 area. These are standby components that can fail in a time-dependent fashion while on standby. Periodic tests are performed to check their operability and if found failed they are put under repair.

A Markov model that describes the stochastic behavior of these components has been developed and quantified. The important characteristics of this model are as follows:

i) The component can be in six states (see Figure 3.1.1).

ii) In state 1 the component (pump, heat exchanger) is available, that is ready to start operating if asked to do so.

iii) The component while on standby can fail with exponentially distributed times to failure. A failure brings the component into state 2 (see Figure 3.1.1).

iv) The failure remains undetectable until a test is performed or a real challenge is posed to the component. A test that will find the component in state 2 will initiate a repair action. The same will happen following a real demand for the component.

v) There are three repair states. States 3 and 3' in which the component is under repair while the reactor is online, and State 4 where the component is under repair with the reactor shutdown.

vi) Following a test that finds the component failed and before the dismantling of the component, all the appropriate motor operated valves must be closed and their breakers racked out from the corresponding MCCs. There is, however, a chance that the operator will not remove the breakers from the MCCs leaving then the MOVs able to open following a signal to do so. If the probability of such an error is P, then a test brings the component from State 2, to State 3 with probability 1-P (breaker removed) and to State 3' with probability P.

vii) The component remains in States 3 or 3' until the repair is completed and then it returns to State 1, or until the allowable outage time is exhausted and then the component transit to State 4 where the repair continues with the reactor shutdown. When the repair is completed, the reactor is brought back online and the component returns to State 1. (Transition 4 to 1).

Quantification

The solution of the model requires the quantification of the following parameters.

i) The catastrophic failure rate λ. This failure mode implies such failures that require major maintenance (dismantling) of the component. The SNPS-PRA used the data presented in Table 2.2.1 from Ref. 2. BNL has updated this table using additional data included in an updated version of Ref. 2 (Ref.6). The new data are summarized in Table 3.1.1.

Maximum likelihood estimators for the failure rates

$$\lambda=\left(\frac{\text{number of failures}}{\text{total operating time}}\right) \text{ yield}$$

$$\lambda=5.7\times10^{-5}/\text{hr for Turbine Driven Pumps}$$

and

$$\lambda=3.3\times10^{-6}/\text{hr for Motor Driven Pumps}$$

ii) The mean times to repair were assumed 100 hrs and 50 hrs for the turbine driven and the reactor driven pumps, respectively. Thus

$$\mu=10^{-2}/\text{hr for Turbine Driven Pumps}$$

and

$$\mu=2\times10^{-2}/\text{hr for Motor Driven Pumps.}$$

iii) In the BNL revision of the SNPS-PRA, the frequency of transients involving MSIV closure has been assessed at 4.42/yr. Thus, the frequency of transients on an hourly basis is

$$\lambda_D=5.0\times10^{-4}/\text{hr}$$

iv) Tests are performed every 3 months (4 times a year) for both motor-driven and turbine-driven pumps. The allowable outage times are 14 and 7 days for turbine-driven and motor-driven pumps, respectively.

v) The probability of not racking out the breakers of the isolation valves (P) is assessed in the SNPS-PRA as 10^{-2}. The same value is used in these requantifications.

vi) The mean time for inadvertently activating a particular switch in the control room has been assumed equal to 10,000 hrs. This implies a rate of

$$\lambda_0 = 10^{-4}/\text{hr}.$$

Quantification of the Markovian model with the numerical values of the parameters mentioned above yields the probabilities per year for the various maintenance induced floods. The results are tabulated in Table 3.1.2. Additional assumptions are: the Core Spray System consists of two motor driven pumps, the LPCI consists of four motor driven pumps and that RBCLCW heat exchangers are equivalent to motor driven pumps.

3.1.2 Rupture-Induced Flood Initiators

A flood can be initiated if a rupture occurs at any point in the pressure boundary of the various systems in the Elevation 8 area. Such a rupture will involve one of the following three types of components: 1) piping; 2) valve; and 3) pump. The model assumes that catastrophic ruptures occur in the following way. A component fails in such a way that if it is demanded to operate then a catastrophic rupture (large enough to allow the flow rates necessary for the flood sizes of interest to this analysis) will occur. That is, the component transits first in a rupture-vulnerable state and then, when a demand occurs, it ruptures.

A Markov model that decribes this stochastic behavior has been developed and quantified. The model is graphically depicted in Figure 3.1.2. The basic characteristics of the model are as follows:

(i) The system in question (HPCI, RCIC, LPCI, CS, RHR, RBCLCWHX) is in state where it is available tc perform its function.

(ii) The system transits to State 2, which is a rupture vulnerable state with failure rate λ_R.

(iii) If a demand occurs while in State 2 a flood is initiated. A demand occurs whenever a transient, a manual shutdown or a test occurs. We distinguish three flood states: State 3, which is a rupture triggered by a transient involving an MSIV closure; State 4, which is a rupture triggered by a turbine-trip transient; and State 5 which is rupture triggered by a manual shutdown or an equipment test.

The solution of this model yields the probabilities that the system will occupy States 3, 4 and 5 denoted by P_S, P_T, P_M, respectively. These probabilities at the end of one-year period provide the frequency of rupture-initiated flood precursors. The expression for these probabilities is

$$P_i(t) = F\,\frac{\lambda_i\lambda_R}{\lambda-\lambda_R}\,[(1-e^{-\lambda_R t})/\lambda_R-(1-e^{-\lambda t})/\lambda] \tag{1}$$

where i = S, T

F is the number of tests per year.

λ_i is the rate of arrival of a transient of type i (i=S,T)

λ_R is the rate of catastrophic rutpure failure in the system

and

λ is the rate of arrival of any transient ($\lambda=\lambda_S+\lambda_T+\lambda_M$)

For the manual shutdown the corresponding expression is

$$P_M(t) = F[\frac{\lambda_M\lambda_R}{\lambda-\lambda_R}(1-e^{-\lambda_R t})/\lambda_R-(1-e^{-\lambda t})/\lambda + \frac{\lambda_R}{\lambda-\lambda_R}(e^{-\lambda_R T}-e^{\lambda T})] \tag{2}$$

Quantification

For a given system having piping of length L, n_v valves and n_p pumps the failure rate λ_R is equal to

$$\lambda_R = L\lambda'+n_v\lambda_v+n_p\lambda_p \tag{3}$$

where λ_V, λ_p are the catastrophic rupture failure rates for valves and pump and λ' the same failure rate per unit of piping length.

A search of the LER, has indicated that at least one pipe rupture (welding failure) has occurred in the ECCS piping in the 215 accumulated BWR years (see Ref.8).

This provides a maximum likelihood estimator for the rupture failure rate of ($1/215yr=5.31x10^{-7}/hr$). Assuming, as in the SNPS-PRA, that only one out of every twenty ruptures will create a break large enough to generate floods of the sizes of concern to this analysis, the catastrophic piping rupture rate becomes $\lambda=2.7x10^{-8}$. This of course is applicable for the total length of safety related piping (denoted by L).

For a particular system with a total of piping length ℓ, then the catastrophic rupture rate for piping becomes

$$\lambda'' = (\frac{\ell}{L})x2.7x10^{-8}/hr \qquad (4)$$

where ℓ/L denotes the fraction of the total length of the piping that belongs to the particular system.

For the rupture rates of the valves and the pumps, the WASH-1400 values were used (see Table G.4-4 in SNPS-PRA). Using the length of piping, number of valves and pumps provided in Table G.4-5 of the SNPS-PRA, and by virtue of Eqs.1-3. The total failure rate λ_R for the various systems along with the probabilities P_S, P_T and P_M were calculated. The results are tabulated in Table 3.1.3.

A total of 13.51 transients per year were assumed (4.42 MSIV closures, 4.89 turbine trips and 4.2 manual shutdowns).

The splitting between maximum and large floods for initiators TFL12-TFL13, TFL14-TFL15, TFL16-TFL17 was done as in the SNPS-PRA, that is, 1 to 2. The additional factor of 20 used in the SNPS-PRA to account for non-pressurized piping is not assumed in the BNL quantification.

3.2 BNL Quantitative Review of the Initiator Event Tree

The quantitative review of the initiator event trees is discussed in the following subsections.

3.2.1 Review of Flooding Alarm Related Procedures

The RB water level is detected by two RB water level monitors installed on the RB floor. The flood alarms are activated by the monitors when the water level is more than 0.5 in. above the floor. The sump alarms will be activated when water level reaches the sump alarm setpoints installed at a level right below the level that activates the RB flood alarms. Sump alarm sensors are installed at various locations in the RB.

The immediate operator action specified in the Alarm Response Procedure (ARP5671) is to initiate the Suppression Pool Leakage Return System. The required subsequent actions are:

1. Monitor RB water level to determine approximate leak rate. Use sump alarms to supplement the information obtained from the above instruments to ascertain the approximate location of the leak.
2. Monitor parameters (such as line pressure and flow rate) of the safety systems as a leak would affect the system parameters. Isolate the source of leakage per procedure listed below in Step 3.
3. If required and plant condition permit, dispatch an operator to the RB floor to visually locate the source of leakage. Isolate using the appropriate system procedure listed below.

System

HPCI, Procedure No.SP23.202.01

Leakage indication:
- Abnormal suction or discharge piping pressure.
- Excessive HPCI Loop Level Pump Flow or low discharge pressure.

- Reactor building sump high water levels in vicinity of leak.
- Reactor building flooding alarm.

Leakage isolation:
- If in standby, isolate the HPCI system by securing the HPCI Loop Level Pump and then closing CST Suction Valve (MOV-031).
- If the system is operating, secure per shutdown procedure and then isolate as described above.

RCIC, Procedure No.SP23.119.01

Leakage indication:
- Abnormal suction or discharge piping pressure.
- Excessive HPCI Loop Level Pump.
- Reactor building sump high water levels.
- Reactor building flooding alarm.

Leakage isolation:
- If in standby, isolate the RCIC system by securing the RCIC Loop Level Pump and then closing CST Suction Valve (MOV-031).
- If the system is operating, secure per shutdown procedure and then isolate as described above.

RHR, Procedure No.SP23.121.01

Leakage indication:
- Heat exchanger service water side temperature inconsistencies.
- Abnormal RHR system flow for mode of operation.
- Abnormal RHR system pressures for mode of operation.
- Reactor water level inconsistencies for mode of operation.
- Sump high level alarms.
- Reactor building flooding alarm.

Leakage isolation:
- Isolate the leakage by shutting down the affected loop in accordance with the appropriate procedure

for the mode in which it was operating and then systematically shutting valves to isolate areas of the system found above to be possible sources of leakage.

- The above isolation procedure may require intermittent operation of the leakage return system to observe the effects on water buildup.
- When the leakage has been isolated return the unaffected portions (as required) to service.

BNL has found that SNPS alarm response procedures specify general guidelines for monitoring system parameters for determining the leakage location and for initiating the leakage isolation. However, the procedures fail to include specific requirements for operators to systematically check the operation parameters of relevant systems. Since there are many system parameter indicators in the control room, the operators may possibly fail to observe the indication of an abnormal system parameter.

When the abnormal condition is severe enough to actuate the alarm of a particular system parameter, the corresponding Alarm Response Procedure will then be followed by operators. However, BNL has reviewed the relevant Alarm Response Procedures for abnormal system parameters, and found that these procedures do not contain steps that should be followed under RB flood conditions. These procedures provide guidelines for conditions other than RB flood, such as water source abnormal or isolation valves abnormal, etc. The operator responses to the flood could be delayed or confused when these Alarm Response Procedures are followed.

3.2.2 Requantification

The revised initiator frequencies are applied for evaluating the sequence frequencies of the initiator event tree. In addition to the critical flood depth of 3'-10" used by SNPS, BNL also evaluated the sequence frequencies corresponding to flood depth of 1'-10" and 1'-3". This is because, as indicated in Table 3.2.1, flood heights of 1'-10" and 1'-3" will disable several vital systems such as HPCI and RCIC. The times for the flood to reach 3'-10", 1'-10", and 1'-3" depth were calculated based on the leakage flow rates determined in SNPS PRA. The calculated times are shown in Table 3.2.2.

The HEP values used by SNPS are identical to the nominal HEP values provided in the Probabilistic Risk Analysis Procedure Guide(7) (see Figure 3.2.1 and Table 3.2.3). BNL feels that the HEP could be higher than the nominal HEP values because the flooding alarm related procedures fail to provide specific guidelines to identify and to isolate the flood source (see Section 3.2.1).

The HEPs under the multiple alarm and the single alarm conditions are listed in Tables 3.2.4 and 3.2.5.

3.3 BNL Review of Functional Event Tree

This section is divided into three subsections. Section 3.3.1 provides a qualitative review of the Shoreham Internal Flood event tree analysis and Section 3.3.2 presents the BNL revised time phased event trees. Section 3.3.3 describes the results obtained from the quantification of the BNL event trees.

3.3.1 Qualitative Review

In general, BNL is of the opinion that the methodology used in the Shoreham Internal Flood Analysis is consistent with that of the state-of-the-art and the approach is reasonable. The analysis for the internal flood postulated much severe scenarios than those of the Shoreham FSAR.

The Shoreham Internal Flood functional event tree analysis is based predominantly on the event trees developed for the internal event initiators, namely, turbine trip, MSIV closure and manual shutdown. These internal flood functional event trees only model flood scenarios where the flood water height at Elevation 8 exceeds 3'-10". While it appears that the Shoreham functional event trees do provide a representative modeling of the plant response, it is not well substantiated that floods that are arrested before reaching 3'-10" will result in negligible core vulnerable frequency.

Table 3.3.1 enumerates the vital equipment that has been identified in the Shoreham analysis. The components that are located at the lowest elevation are presented first. It can be seen that at the 1' level, both the RCIC and HPCI vacuum pumps and condensate pumps are expected to be disabled. However, it is judged that their failures do not lead to the failure of the respective high pressure systems. Similar arguments apply to the loop level pumps of the low pressure core spray, HPCI and the RCIC systems as well. At approximately 2', instrumentation for both high pressure injection systems are submerged and hence resulting in failure of both systems. At 3'-10" instrumentation for both LPCS and RHR is submerged leading to the failure of those low pressure systems. In the Shoreham analysis the critical height of 3'-10" is selected. However, since both HPCI and RCIC have failed at about the 2' level, these scenarios with termination of the flood prior to 3'-10" may not contribute an insignificant amount to the core vulnerable frequency. In the BNL revised event trees, a time-phased approach is used to include the contribution from flooding below the 3'-10" level.

Another area of concern stems from the treatment of propagation of failures in the Shoreham analysis. As noted in Table 3.3.1, at the 1' level, 4-480V pumps are expected to experience electrical shorts. The Shoreham analysis did not investigate any cascading failure which may result from the electrical shorts. BNL reviewed the electrical drawings and elementary drawings for some of the systems. It appears that for each pump there is only one electrical breaker which separates it from the rest of the loads in the same motor control center (MCC). Random failure of this breaker to open could result in the propagation of the short circuit fault upstream to the MCC, other MCCs and the load center. BNL's review of the electrical diagrams indicates that failure of the breaker to open will result in tripping the breaker at the load center. Discussions with Shoreham engineers suggested that there may possibly be an additional breaker per pump that is in series with the first breaker. However, this was not confirmed by BNL. In the BNL revised event trees, only one breaker is assumed and its failure is modeled explicitly.

BNL did not review the spraying effects due to water cascades from higher elevations.

3.3.2 BNL Time Phase Event Tree

The determination of the time periods which are critical to the consideration of the progression of the flood is based on the vital equipment location list (Table 3.3.1). Three heights were selected for the BNL analysis: at the 1'-3" level, at the 1'-10" level, and at the 3'-10" level. If the flood is terminated prior to reaching the 1'-3" level, no impact is assumed for any equipment and the plant will be shutdown, this is Phase 1. However, if the flood water exceeds the 1'-3" level but is terminated before the 1'-10" level, this is Phase II. Phase III entails the failures of both HPCI and RCIC system as well as the loss of power to the MG set recirculation pump fluid coupler before arresting the flood below the 3'-10" level. Any flood level which exceeds the 3'-10" level, it is treated in Phase IV.

The event trees of these four phases are presented in Figures 3.3.1 through 3.3.4. Given that the flood is terminated in Phase I, BNL assumed that the reactor has a high probability (0.9) that it will be manually shutdown. Ten percent of the time, it may result in a MSIV closure event. These two branches of the Phase I event trees are transferred to the respective internal event tree, Figure 3.3.1.

Figure 3.3.2 depicts the Phase II functional event tree which considers the various accident mitigation systems. Moreover, owing to the fact that a number of the 480V pumps will be flooded, the possibility of a breaker failure to isolate the fault is also evaluated. It is assumed that the breaker failure to open probability is 1×10^{-3} and there are a total of five pumps in Division 1 and two pumps in Division II that will be short circuited. A probability of 0.5 is also assumed that failure of a load center in a division would lead to failure of other equipment connected to that division. In the event of a MSIV closure, the feedwater system is considered to be unavailable. The probability that the reactor will be manually shutdown is also assumed to be 0.9 for the maintenance induced flood events.

Figure 3.3.3 illustrates the functional event tree used to describe the Phase III events. The major difference between this event tree and the Phase

II tree is the high pressure systems. In the Phase III events, both the RCIC and the HPCI systems are unavailable due to the failure of respective instrumentation. The probability that the reactor will be manually shutdown is assumed to be 0.5 for the maintenance induced flood events.

The Phase IV event tree is presented in Figure 3.3.4. This tree is drastically different from the other ones in that it only considers the feedwater system, the depressurization function and the PCS. All the other systems are disabled due to flooding. The likelihood that the reactor will be manually shutdown is the same as in Phase III for maintenance-induced floods.

3.3.3 Quantitative Analysis

Based on the development of the revised flood initiator frequency, the BNL time-phased event tree and the modified human response to arrest flood, quantitative results are obtained. In the BNL analysis, there are 17 different flood precursors. Similar to the Shoreham classification, the first five precursors are online maintenance related; the remaining twelve of them are rupture related. A detailed discussion on the BNL flood precursors is given in Section 3.1.

Owing to the ways that these flood precursors are calculated, the initiator event trees have been modified to include only three functions: the flood alarm annunication, I; operator action to isolate flood, A; and reactor status. The entry condition to the different time phase event trees is determined by the A function (see Section 3.2 for details).

Each of the 17 flood precursors were evaluated with the initiator event tree and the four time phase event trees. The unavailability values for the various event trees are the same as those used in the Shoreham analysis except as noted in the last section.

When the time phase event trees were quantified for the 17 flood precursors, the results are the conditional frequency of core vulnerable given the particular flood precursor. These frequencies are summarized in Table

3.3.2. The seventeen precursors are listed as rows while the four phases are shown as columns. Within each precursor, contributions from manual shutdown, MSIV closure or turbine trip are also shown. For instance, the conditional frequency of core vulnerable with operator arresting the flood prior to 3'-10" but after 1'-10" - Phase III, for TFL1 is 2.0(-5) given the reactor is manually shutdown. However, if instead of a manual shutdown, the plant experiences a MSIV closure, then the conditional frequency is 8.5(-4).

As expected, the conditional frequency consistently increases as the flood progresses to higher elevations. In other words, the conditional frequency of Phase IV is always larger than any of the other phases. Another noteworthy observation is the unusually large conditional frequency of core vulnerable for the LPCI system induced flood, i.e., TFL4 and TFL8. The TFL9 and TFL5 values are also large since they disabled the LPCI systems as well.

The core vulnerable frequency given the BNL revised flood precursors, initiator event trees and time phase event trees is shown in Table 3.3.3. In this table, the 17 precursors are depicted on the left with the 4 phases depicted as columns. Each precursor also identifies the contributions from the various plant states. Core vulnerable frequency contributions from Phase I and II are very small, in the order of 10^{-9}. Contributions from Phase III are not insignificant but not substantial, approximately 10^{-6}. Seventy percent of the total core vulnerable frequency (70% of 2.0(-5)) is attributable to LPCI system maintenance or rupture induced flood. The maintenance contribution to flood is about 37% while the balance is due to rupture.

It appears also that failure to properly model the fault propagation of the short circuits through the breakers does not have a significant effect on core vulnerable frequency.

3.4 Uncertainty Estimates

This section presents a limited uncertainty assessment on the BNL quantitative analysis for the core vulnerable frequency due to reactor building flooding.

A rigorous propagation of the uncertainties is outside the scope of the present review. The BNL approach for the uncertainty evaluation consisted of the following general steps.

1. The uncertainties in the human errors as well as the split ratio between the manual shutdown and the MSIV closure event were quantified by fitting lognormal distributions to evaluate uncertainty measures (mean and variance). An error factor of 10 was applied to human errors and the split ratio.

2. Human errors of the following operator actions were included for the uncertainty evaluation:

   - Operator maintains isolation valves in closed position during the online maintenance (Event E, see Section 2.3).

   - Operator diagnoses and responds to isolate the flood (Event A, see Section 2.3).

   - Operator depressurizes the Reactor Pressure Vessel (Event X, Figures 3.3.2-3.3.4).

3. The uncertainties in the core vulnerable frequency were evaluated using the major accident sequences and the distributions assessed in Step 1.

The SAMPLE code was used for the estimaton of uncertainties. The mean, median, 5% and 95% probability intervals for the core vulnerable frequency are shown as follows.

Mean	=	1.91E-5
Median	=	1.90E-6
5% Confidence	=	2.19E-7
95% Confidence	=	7.51E-5

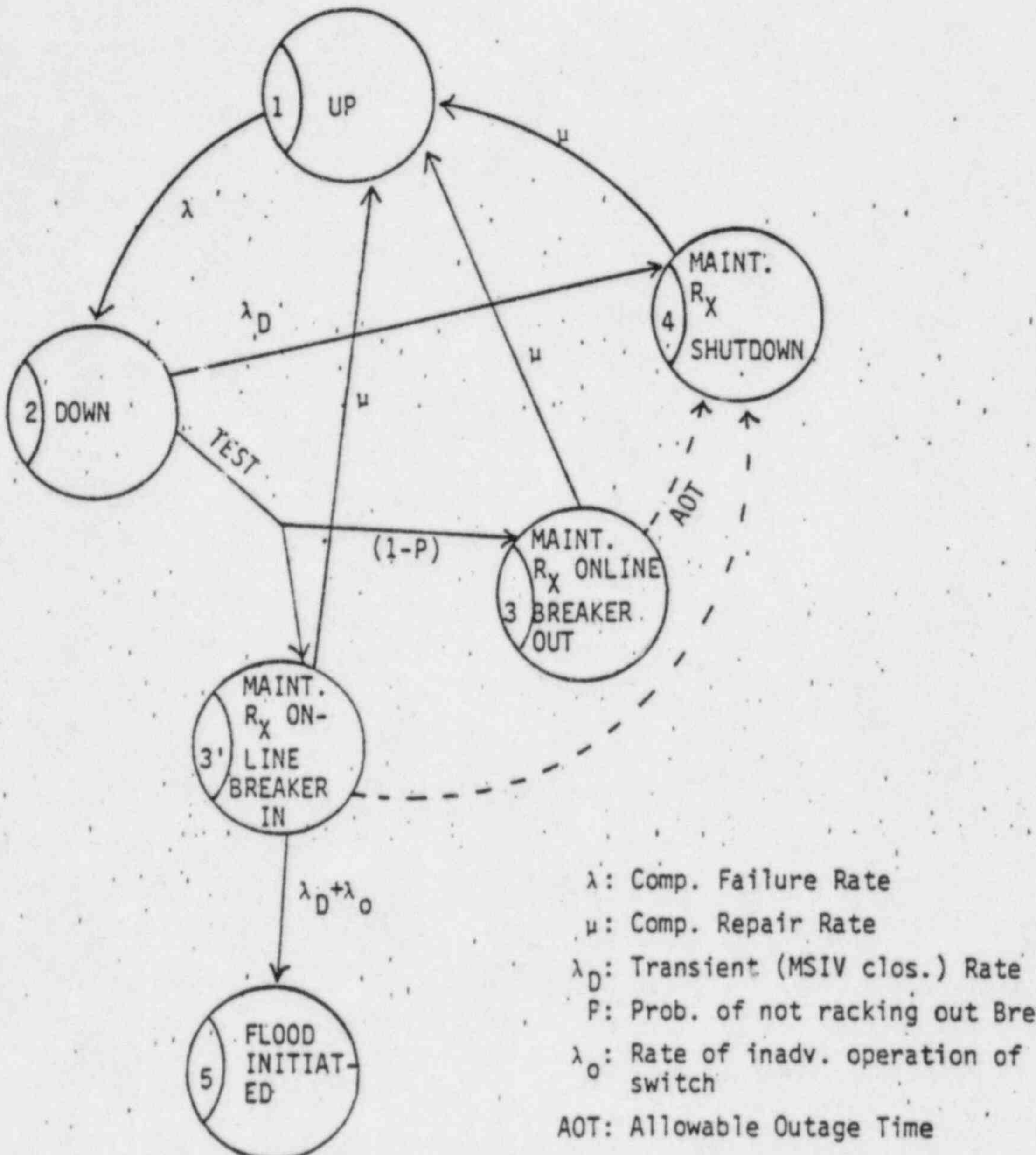


Figure 3.1.1 State Transition Diagram for Component-Maintenance Induced Floods.

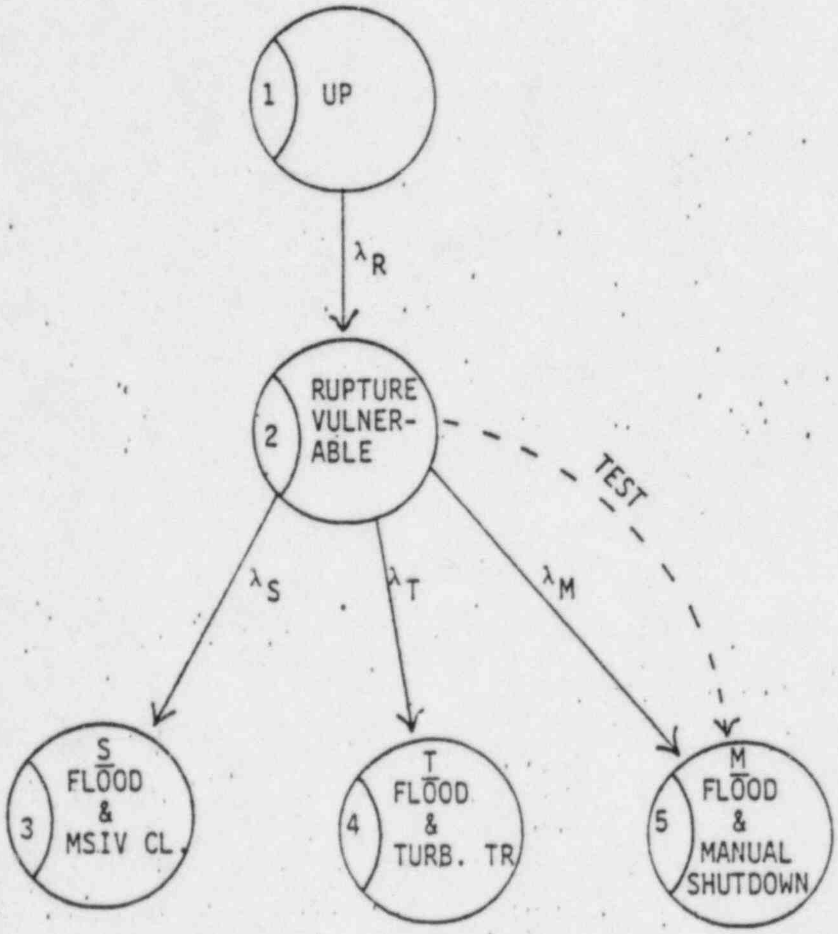


Figure 3.1.2 State Transition Diagram for Rupture-Induced Floods.

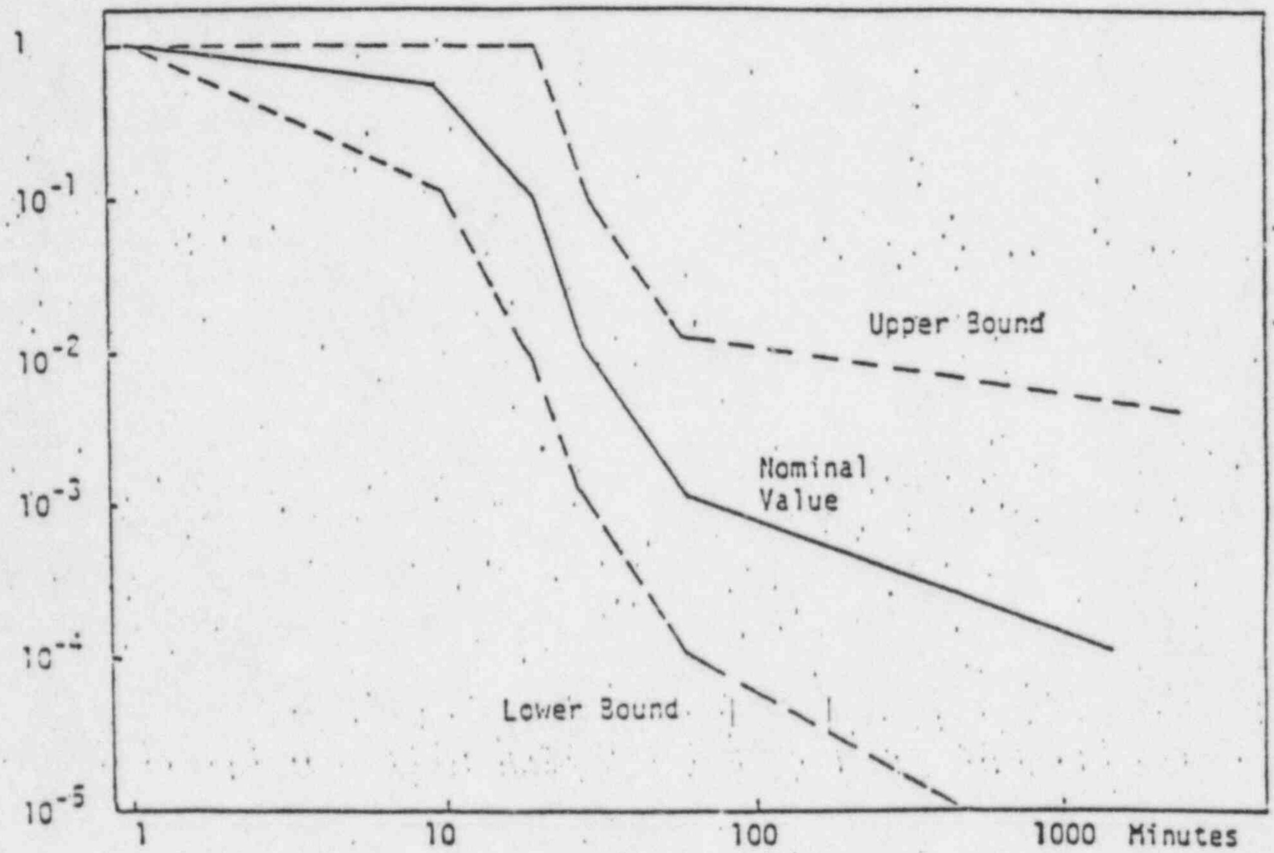


Figure 3.2.1 Problem-solving human error probability vs time - screening values.

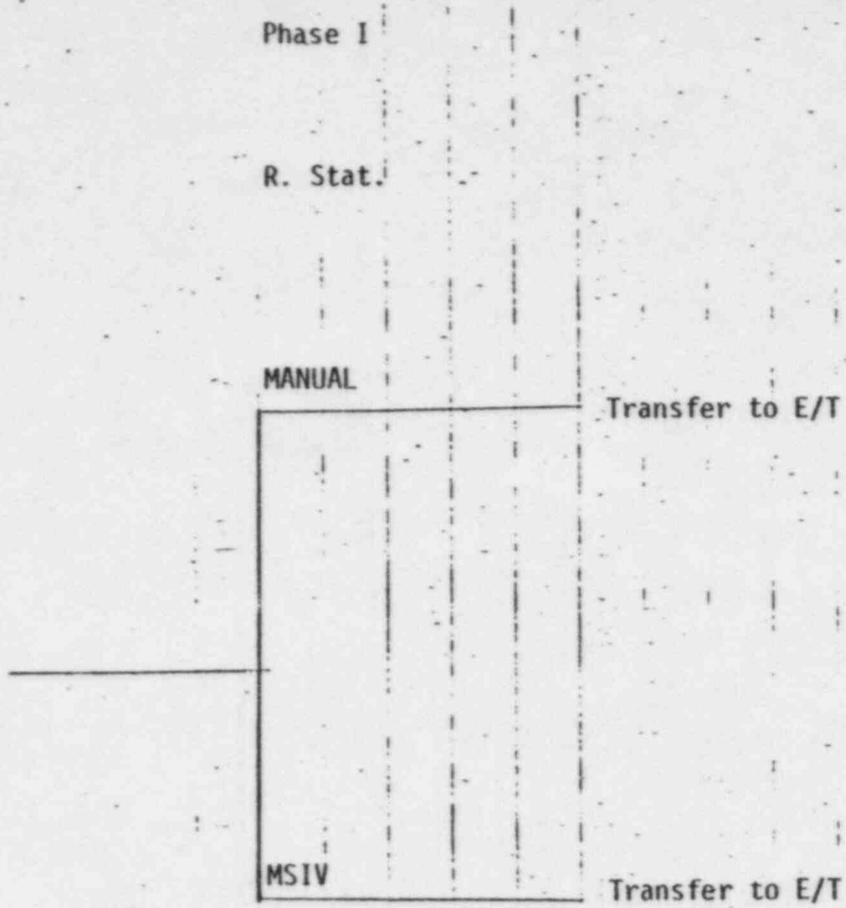


Figure 3.3.1 Phase I of Internal Flood Functional Event Tree

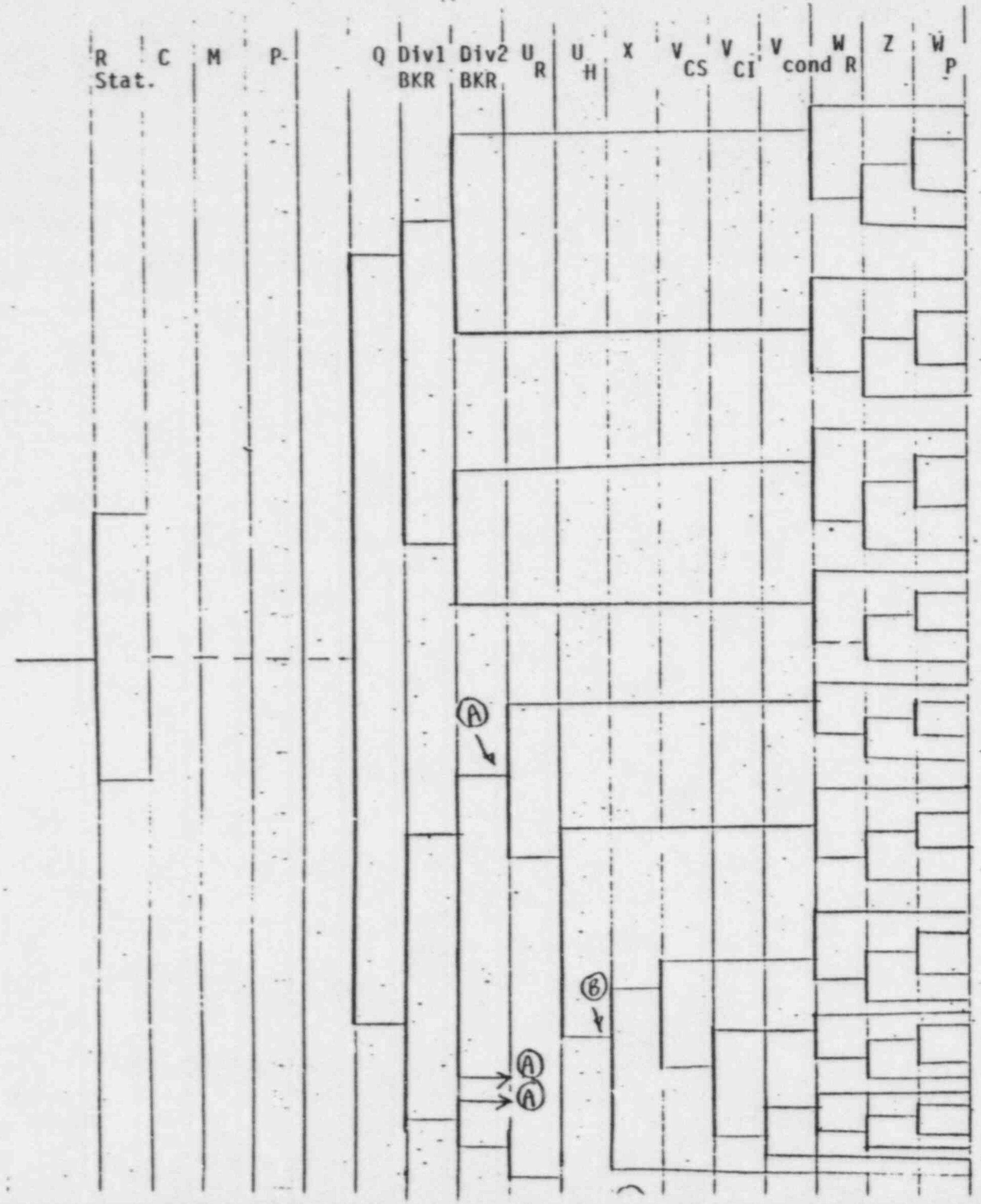


Figure 3.3.2 Phase II of Internal Flood Functional Event Tree

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

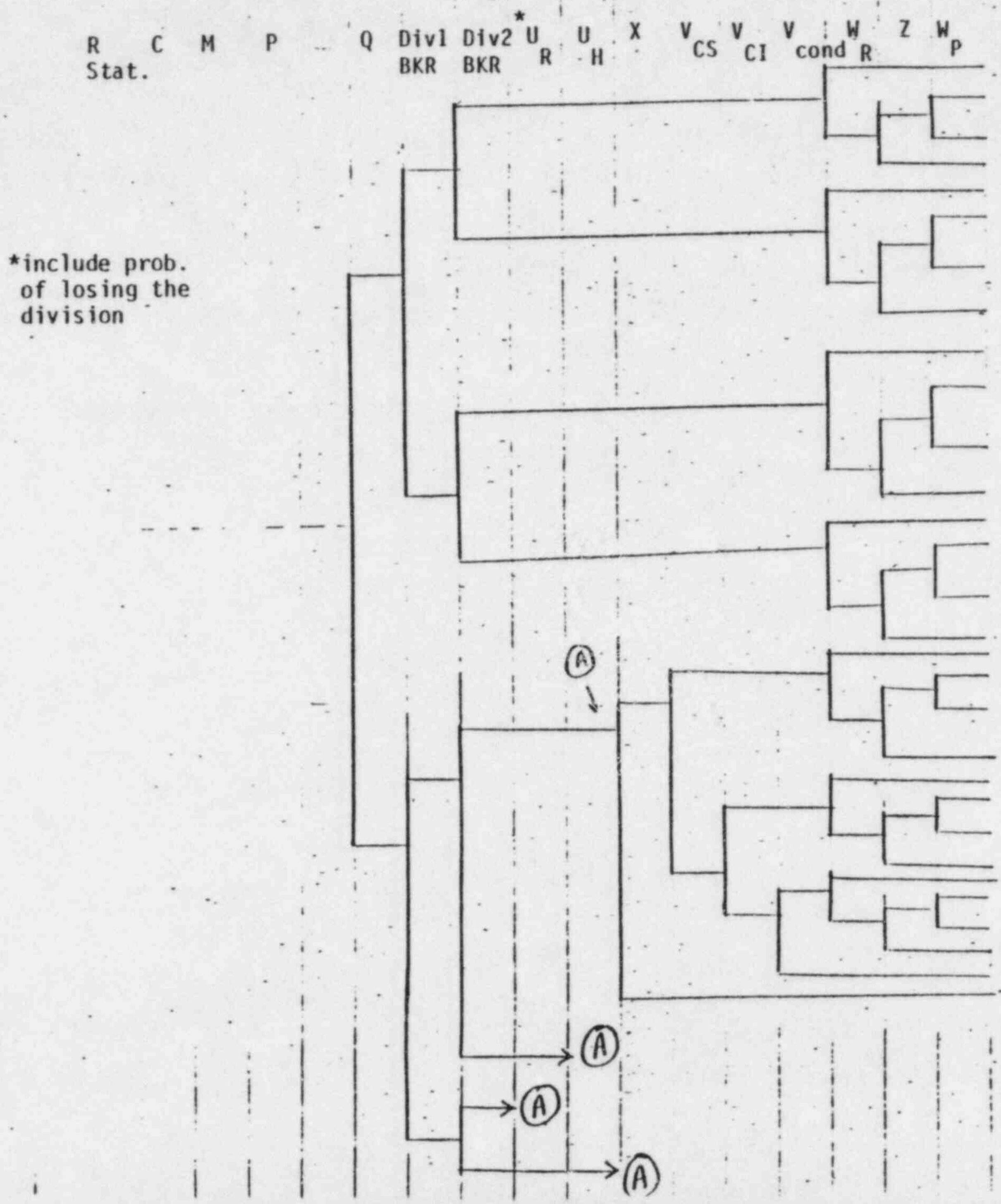


Figure 3.3.3 Phase III of Internal Flood Functional Event Tree

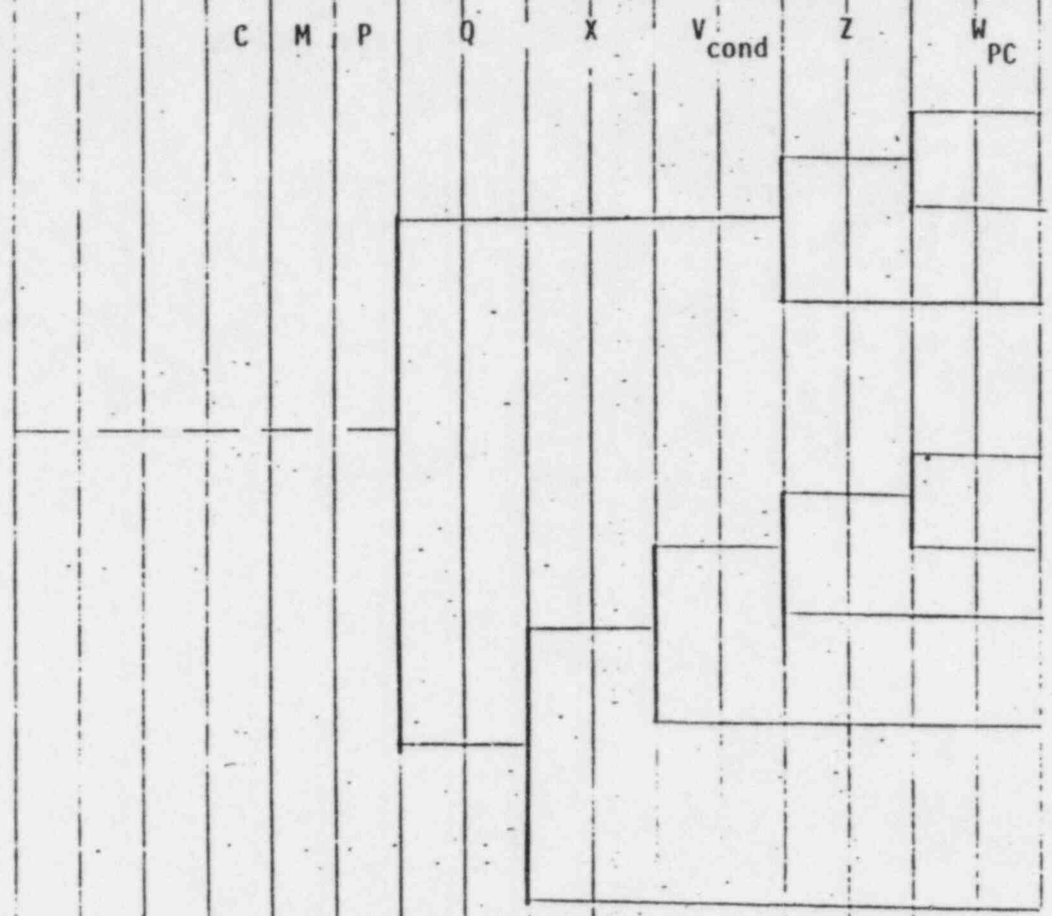


Figure 3.3.4 Phase IV of Internal Flood Functional Event Tree

Table 3.1.1 LER Data for BWR Standby Pumps for the Period of January 1972 Through September 1980

Standby Pumps	Demands	Standby Hours	Leakage Rupture	Does Not Start	Loss of Function	Does Not Continue To Run
Motor Driven	20,321	10,453,806	9	8	8	9
Turbine Driven	2,860	1,439,491	-	34	25	23

Table 3.1.2 Frequency of Maintenance - Induced Flood Precursors

System	Initiator Event Trees	Probability per Year
1. RCIC	TFL1 P.D	1.05×10^{-4}
	TFL1 P.E_O	2.10×10^{-5}
	TFL1 P.E_L	2.10×10^{-5}
2. HPCI	TFL2 P.D	1.05×10^{-4}
	TFL2 P.E_O	2.10×10^{-5}
	TFL2 P.E_L	2.10×10^{-5}
3. Core Spray (2 motor driven pumps)	TFL3 P.D	1.89×10^{-5}
	TFL3 P.E_O	1.87×10^{-6}
4. LPCI (4 motor driven)	TFL4 P.D	3.78×10^{-5}
	TFL4 P.E_O	3.74×10^{-6}
5. Service Water (RHR or RB(LW HX) 2 motor driven pumps	TFL5 P.D	1.89×10^{-5}
	TFL4 P.E_O	1.88×10^{-6}

Table 3.1.3 Flood Precursor Frequency

	Pipe	Valves	Pump	Total λ_R	P_S	P_T	P_M
TFL6	1.2(-9)	6.5(-9)	0	7.7(-9)	1.6(-5)	1.7(-5)	1.5(-5)
TFL7	2.0(-9)	1.3(-8)	0	1.5(-8)	3.1(-5)	3.4(-5)	2.9(-5)
TFL8	3.7(-9)	2.9(-8)	0	3.2(-8)	6.5(-5)	7.3(-5)	6.2(-5)
TFL9	1.1(-8)	2.3(-8)	6.0(-10)	1.3(-8)	2.6(-5)	2.9(-5)	2.5(-5)
TFL10	2.4(-9)	1.3(-9)	0	3.7(-9)	7.5(-6)	8.4(-6)	7.2(-6)
TFL11	1.1(-9)	9.1(-9)	1.5(-10)	1.0(-8)	2.1(-5)	2.4(-5)	2.0(-5)
TFL12	1.4(-9)	3.9(-9)	1.5(-10)	5.5(-9)	3.7(-6)	4.0(-6)	3.6(-6)
TFL13	-	-	-	-	7.3(-6)	8.0(-6)	7.1(-6)
TFL14	1.9(-9)	5.2(-9)	3.0(-10)	7.4(-9)	5.0(-6)	5.6(-6)	4.8(-6)
TFL15	-	-	-	-	1.0(-5)	1.1(-5)	9.6(-6)
TFL16	1.9(-9)	5.2(-9)	6.0(-10)	7.7(-9)	5.2(-6)	5.8(-6)	5.0(-6)
TFL17	-	-	-	-	1.0(-5)	1.2(-5)	1.0(-5)

Table 3.2.1
MAJOR ELEVATION 8 EQUIPMENT LIST

EQUIP. TYPE	EQUIPMENT DESCRIPTION	PART NO.	POSTULATED DISABLED HEIGHT*
PUMPS			
	Floor Drain Sump Pumps	1G11*P-035A-D 1G11*P-036A-F	1'-0"
	Dry Floor Drain Tank Pumps	1G11*P-161A,B	1'-0"
	Radwaste Equip. Drain Sump & Pump to Porous Sump	1G11*P-224A,B	1'-1"
	** HPCI Pump	1E41*P-016	-----
	HPCI Vacuum Pump	1E41*P-075	1'-0"
	HPCI Con. Pump	1E51*P-076	1'-0"
	** RCIC Pump	1E51*P-015	-----
	RCIC Vacuum Pump	1E51*P-076	1'-0"
	RCIC Con. Pump	1E51*P-077	1'-0"
	** RHR Pump Motors	1E11*P-014A-D	5'-4"
	** Leakage Return Pump	G11-*P-270	3'-9"
	** Core Spray Loop Level Pumps	1E21*P-049A,B	1'-3"
	Drywell Equip. Drain Tank Pumps	1G11*P-0332A,B	1'-2"
	RCIC Loop Level Pump	1E51*P-051	1'-4"
	** HPCI Loop Level Pump	1E41*P-050	2'-3"
TURBINES			
	** HPCI Turbine	1E41*-TU-002	6'-0"
	** RCIC Turbine	1E41*-TU-005	4'-0"

Table 3.2.1 (Continued)
MAJOR ELEVATION 8 EQUIPMENT LIST

EQUIP. TYPE	EQUIPMENT DESCRIPTION	PART NO.	POSTULATED DISABLED HEIGHT*
MOTOR CONTROL CENTERS			
	Sump Pumps and Cooling Water Pumps to Recirc. Pump MG-Set Fluid Coupler	1R24-11D1 1R24-12D1	1'-6" 1'-6"
TANKS			
	Floor Drain Sump Tank	1G11*TK-050A,B 1G11*TK-056A-C	----- -----
	Drywell Floor Drain Receiver	1G11*TK-057	-----
	Salt Water Drain Tank	1G11*TK-190	-----
	Drywell Equip. Drain Receiver	1G11*TK-049	-----
HEAT EXCHANGER			
	HPCI Barometric Con. Vacuum Tank	1E41*E-036	-----
	RCIC Barometric Con. Tank	1E51*E-038	-----
	RHR Heat Exchanger	1E114*E-034A,B	-----
	RBCLCW Heat Exchangers	1P42*-011A,B	-----
	Drywell Equip. Drain Cooler	1G11*E-094	-----

Table 3.2.1 (Continued)

MAJOR ELEVATION 8 EQUIPMENT LIST

EQUIP. TYPE	EQUIPMENT DESCRIPTION	PART NO.	POSTULATED DISABLED HEIGHT+
ELEC. PANELS			
	** RCIC Instr. Rack	1H21*PNL-017	2'-0"
	** RCIC Instr. Rack	1H21*PNL-037	2'-0"
	** Core Spray Rack	1H21*PNL-01	3'-10"
	** Core Spray Rack	1H21*PNL-019	3'-10"
ELEC. PANELS			
	** RHR Inst. Rack A	1H21*PNL-018	3'-10"
	** RHR Inst. Rack B	1H21*PNL-021	3'-10"
	** HPCI Inst. Rack A	1H21*PNL-036	1'-10"
	** HPCI Inst. Rack B	1H21*PNL-14	1'-10"

** Equipment required for operation of the identified system.

\+ Heights are taken from a physical survey measurement from the bottom of the component to floor level.

---- Non-electrical component

Table 3.2.2 Times to Flood Depth of 3'-10", 1'-10", and 1'-3" in Reactor Building

System	Water Source	Leakage Location	Time (min.) to Flood Depth of 3'-10"	1'-10"	1'-3"
HPCI	S.P.	pump suction (max.)	17	7.9	5.4
	S.P.	pump suction (large	34	15.8	10.8
	CST	pump suction (max.)	13	6.4	4.4
	CST	pump suction (large)	27	12.8	8.7
	---	pump discharge	37	17.5	11.9
RCIC	S.P.	pump suction (max.)	110.0	50.8	34.6
	S.P.	pump suction (large)	220.0	101.6	69.3
	CST	pump suction (max.)	76.0	36.3	24.8
	CST	pump suction (large)	152.0	72.6	49.5
LPCI	S.P.	pump suction (max.)	9.4	4.5	3.1
	S.P.	pump suction (large)	19.0	9.0	6.1
	---	pump discharge	15	7.3	5.0
CS	S.P.	pump suction (max.)	12	5.9	4.0
	S.P.	pump suction (large)	24	11.8	8.1
	CST	pump suction (max.)	13	6.4	4.4
	CST	pump suction (large)	27	12.8	8.7
	---	pump discharge	23	11.1	7.6
SW	SW	RHR heat exchanger	20	9.5	6.5
WFPS	WFPS	rupture of 8" pipe	40	19.1	13.0

Note: 1. Large flow rates is 1/2 of maximum flow rates.
2. Flood times were calulated based on a 41,600 gallons per foot depth in the reactor building.
3. S.P. = Suppression Pool
CST = Condensate Storage Tank
SW = Service Water System
WFPS = Water Fire Protection System Tanks

Table 3.2.3 Human Error Probability: Screening Values

Problem-solving

Time	Nominal Value	Error Factor
<1 min.	1	---
10 min.	5E-1	5
20 min.	1E-1	10
30 min.	1E-2	10
60 min.	1E-3	10
1500 min.	1E-4	30

Procedural Errors

Nominal Value	Error Factor
1E-3 (With Recovery)	3
1E-2 (Without Recovery)	3

Table 3.2.4 HEP (Event A) Single Alarm Condition
Manual Shutdown (NUREG/CR-1278)

	1'-3"	1'-10"	3'-10"
TFL1	10^{-3}	10^{-3}	2.0×10^{-4}
TFL2	1	1	0.1
TFL3	1	1	0.1
TFL4	1	1	1
TFL5	1	1	10^{-2}
TFL6	0.1	0.1	10^{-3}
TFL7	1	0.1	10^{-2}
TFL8	1	1	0.1
TFL9	1	1	10^{-2}
TFL10	0.1	0.1	10^{-3}
TFL11	10^{-3}	10^{-3}	2×10^{-4}
TFL12	1	1	0.1
TFL13	0.1	0.1	10^{-3}
TFL14	1	1	0.1
TFL15	1	0.1	10^{-2}
TFL16	1	1	1
TFL17	1	1	0.1

Table 3.2.5 HEP (Event A), Multiple Alarm Condition (Nominal Value, PRA Procedures Guide)

	1'-3"	1'-10"	3'-10"
TFL1	10^{-2}	10^{-2}	10^{-3}
TFL2	1	1	0.5
TFL3	1	1	0.5
TFL4	1	1	1
TFL5	1	1	0.1
TFL6	0.5	0.5	10^{-2}
TFL7	1	0.5	0.1
TFL8	1	1	0.5
TFL9	1	1	0.1
TFL10	0.5	0.5	10^{-2}
TFL11	10^{-2}	10^{-2}	10^{-3}
TFL12	1	1	0.5
TFL13	0.5	0.5	10^{-2}
TFL14	1	1	0.5
TFL15	1	0.5	0.1
TFL16	1	1	1
TFL17	1	1	0.5

Table 3.3.1 Vital Equipment Locations at Elevation 8

1'	HPCI vac. pump cond. pump RCIC vac. pump cond. pump	1'-3"
1'-3"	CS loop level pump	
1'-4"	RCIC loop pump	
1'-6"	recir. pump M-G set	1'-10"
1'-10"	HPCI instrumentation	
2'	RCIC instrumentation	
2'-3"	HPCI loop level pump	
3'-10"	RHR instrumentation CS instrumentation	3'-10"

Table 3.3.2 Conditional Frequency of Core Vulnerable
(1 of 2)

		Phase I	Phase II	Phase III	Phase IV
TFL1	Manual	5.8(-7)	2.7(-6)	2.0(-5)	7.3(-3)
	MSIV	3.2(-6)	8.7(-5)	8.5(-4)	1.2(-1)
	TT	7.7(-7)	2.2(-5)	2.1(-4)	3.3(-2)
TFL2	Manual	5.8(-7)	2.2(-6)	2.0(-5)	7.3(-3)
	MSIV	3.2(-6)	6.8(-5)	8.5(-4)	1.2(-1)
TFL3	Manual	5.8(-7)	1.1(-6)	2.2(-5)	7.3(-3)
	MSIV	3.2(-6)	1.1(-5)	9.5(-4)	1.2(-1)
TFL4	Manual	5.8(-7)	3.9(-4)	5.2(-4)	7.3(-3)
	MSIV	3.2(-6)	2.0(-2)	2.6(-2)	1.2(-1)
TFL5	Manual	5.8(-7)	3.9(-4)	5.2(-4)	7.3(-3)
	MSIV	3.2(-6)	2.0(-2)	2.6(-2)	1.2(-1)
TFL6	Manual	5.8(-7)	2.2(-6)	2.0(-5)	7.3(-3)
	MSIV	3.2(-6)	6.8(-5)	8.5(-4)	1.2(-1)
	TT	7.7(-7)	1.6(-5)	2.1(-4)	3.3(-2)
TFL7	Manual	5.8(-7)	1.1(-6)	2.2(-5)	7.3(-3)
	MSIV	3.2(-6)	1.1(-5)	9.5(-4)	1.2(-1)
	TT	7.7(-7)	3.2(-6)	2.3(-4)	3.3(-2)
TFL8	Manual	5.8(-7)	3.9(-4)	5.2(-4)	7.3(-3)
	MSIV	3.2(-6)	2.0(-2)	2.6(-2)	1.2(-1)
	TT	7.7(-7)	4.7(-3)	6.2(-3)	3.3(-2)
TFL9	Manual	5.8(-7)	3.9(-4)	5.2(-4)	7.3(-3)
	MSIV	3.2(-6)	2.0(-2)	2.6(-2)	1.2(-1)
	TT	7.7(-7)	4.7(-3)	6.2(-3)	3.3(-2)
TFL10	Manual	5.8(-7)	1.1(-6)	2.2(-5)	7.3(-3)
	MSIV	3.2(-6)	1.0(-5)	9.5(-4)	1.2(-1)
	TT	7.7(-7)	3.2(-6)	2.3(-4)	3.3(-2)
TFL11	Same as TFL1				
TFL12	Same as TFL6				
TFL13	Same as TFL6				

Table 3.3.2 Conditional Frequency of Core Vulnerable
(2 of 2)

		Phase I	Phase II	Phase III	Phase IV
TFL14	Manual	5.8(-7)	1.1(-6)	2.2(-5)	7.3(-3)
	MSIV	3.2(-6)	1.1(-5)	9.5(-4)	1.2(-1)
	TT	7.7(-7)	3.2(-6)	2.3(-4)	3.3(-2)
TFL15	Same as TFL14				
TFL16	Same as TFL8				
TFL17	Same as TFL8				

Table 3.3.3 Core Vulnerable Frequency
(1 of 2)

		P-1	P-2	P-3	P-4	TOTAL
TFL1	Man.	7.3(-11)	0	1.7(-11)	1.2(-9)	
	MSIV	4.5(-11)	0	8.2(-11)	1.3(-8)	
		1.2(-10)	0	9.9(-11)	1.5(-8)	1.4(-8)
TFL2	Man.	0	0	9.1(-10)	2.1(-7)	
	MSIV	0	0	3.9(-8)	3.4(-6)	
		0	0	4.0(-8)	3.6(-6)	3.7(-6)
TFL3	Man.	0	0	1.2(-10)	3.5(-8)	
	MSIV	0	0	5.2(-9)	5.8(-7)	
		0	0	5.3(-9)	6.1(-7)	6.2(-7)
TFL4	Man.	0	0	0	1.5(-7)	
	MSIV	0	0	0	2.5(-6)	
		0	0	0	2.7(-6)	2.7(-6)
TFL5	Man.	0	0	4.9(-9)	6.9(-9)	
	MSIV	0	0	2.5(-7)	1.1(-7)	
		0	0	2.5(-7)	1.1(-7)	3.7(-7)
TFL6	Man.	*	0	*	1.6(-10)	
	MSIV	*	0	1.4(-8)	5.6(-8)	
		*	0	1.4(-8)	5.6(-8)	7.0(-8)
TFL7	Man.	0	*	*	1.4(-9)	
	MSIV	0	3.9(-10)	2.6(-8)	8.3(-7)	
		0	3.9(-10)	2.6(-8)	8.3(-7)	8.6(-7)
TFL8	Man.	0	0	1.5(-8)	2.3(-8)	
	MSIV	0	0	1.6(-6)	4.3(-6)	
	TT	0	0	2.3(-7)	1.2(-6)	
		0	0	1.8(-6)	5.5(-6)	7.3(-6)
TFL9	Man.	0	0	6.5(-9)	1.2(-9)	
	MSIV	0	0	9.2(-7)	3.4(-7)	
	TT	0	0	1.8(-8)	9.9(-8)	
		0	0	9.4(-7)	4.4(-7)	1.4(-6)

*Less than 1.0(-10).

Table 3.3.3 Core Vulnerable Frequency
(2 of 2)

		P-1	P-2	P-3	P-4	TOTAL
TFL10	Man.	*	0	*	*	
	MSIV	*	0	3.9(-9)	1.4(-8)	
	TT	*	0	9.4(-10)	3.6(-9)	
		*	0	4.8(-9)	1.8(-8)	2.2(-8)
TFL11	Man.	*	0	*	*	
	MSIV	*	0	1.6(-10)	1.4(-8)	
	TT	*	0	*	3.2(-9)	
		*	0	1.6(-10)	1.7(-8)	1.7(-8)
TFL12	Man.	0	0	*	1.1(-9)	
	MSIV	0	0	4.5(-9)	4.9(-7)	
		0	0	4.5(-9)	4.9(-7)	5.0(-7)
TFL13	Man.	*	0	*	*	
	MSIV	*	0	6.6(-9)	2.5(-8)	3.1(-8)
TFL14	Man.	0	0	*	1.8(-9)	
	MSIV	0	0	7.3(-9)	6.9(-7)	
		0	0	7.3(-9)	6.9(-7)	7.0(-7)
TFL15	Man.	0	*	*	4.6(-10)	
	MSIV	0	*	8.4(-9)	2.6(-7)	
		0	*	8.4(-9)	2.6(-7)	2.8(-7)
TFL16	Man.	0	0	0	1.8(-8)	
	MSIV	0	0	0	9.2(-7)	
	TT	0	0	0	1.9(-7)	
		0	0	0	1.1(-6)	1.1(-6)
TFL17	Man.	0	0	2.3(-9)	3.8(-9)	
	MSIV	0	0	2.5(-7)	6.6(-7)	
	TT	0	0	3.7(-8)	2.0(-7)	
		0	0	2.9(-7)	8.6(-7)	1.2(-6)

*Less than 1.0(-10).

4.0 SUMMARY

BNL reviewed the internal flood analysis which is a part of the Shoreham PRA and found that assumptions, methodology, and results are reasonable. BNL re-evaluated the flood precursor frequency using recent LER data and a more accurate methodology. This methodology avoids some of the conservatisms in the SNPS-PRA approach. A slight increase in the initiator frequency is calculated because of the revised data.

Similarly, based on the PSA Procedure Guide, the HEP was reviewed and only minimal changes were made to the Shoreham HEP values used in the analysis. As for the functional event trees, a time phase approach was adopted to better model the progression of the flood events.

Results are summarized in Table 4.1. This table can be divided into three parts. Part A provides a comparison between the Shoreham results and those obtained in the BNL review. The BNL value is about 5 times that of the Shoreham frequency, 2.0(-5) vs. 3.9(-6). The contributions from the different plant states are also presented. The major increase in the total core vulnerable frequency in the BNL analysis is attributable to the increase in flood precursor frequencies. Part B compares only the contributions from the BNL Phase IV results with the Shoreham values. It can be inferred that by neglecting the initial three phases, the core vulnerable frequency will be underestimated by 3×10^{-6} or about 18%. Part C shows the contributions of core vulnerable frequency for different plant states due to maintenance and rupture induced floods. In the Shoreham analysis 41% of the core vulnerable frequency is calculated to be caused by maintenance related floods while the BNL analysis shows 37%.

An uncertainty estimation has been carried out assuming lognormal distributions. An error factor of 10 was applied to the operator errors and the split ratio for the manual shutdown and the MSIV closure event following the Reactor Building flooding. The results of the uncertainty assessment for the core vulnerable frequency are as follows.

Mean	=	1.9E-5
Median	=	1.9E-6
5% Confidence	=	2.2E-7
95% Confidence	=	7.5E-5

Table 4.1 Summary of Core Vulnerable Frequency

		Shoreham	BNL
Part A			
Manual		8.5(-8)	4.8(-7)
MSIV		3.0(-6)	1.8(-5)
TT		7.7(-7)	2.0(-6)
Total		3.9(-6)	2.0(-5)

		Shoreham	BNL (only Phase IV
Part B			
Manual		8.5(-8)	4.5(-7)
MSIV		3.0(-6)	1.5(-5)
TT		7.7(-7)	1.7(-6)
Total		3.9(-6)	1.7(-5)

		Shoreham	BNL
Part C			
Manual	Maintenance	3.9(-8)	4.1(-7)
	Rupture	1.6(-7)	7.0(-8)
MSIV	Maintenance	1.5(-6)	6.9(-6)
	Rupture	1.4(-6)	1.1(-5)
TT	Maintenance	0	0
	Rupture	6.7(-7)	2.0(-6)
Total	Maintenance	1.6(-6)	7.3(-6)
	Rupture	2.3(-6)	1.3(-5)

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