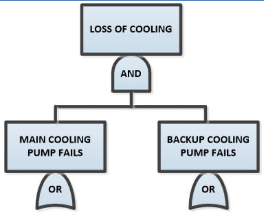
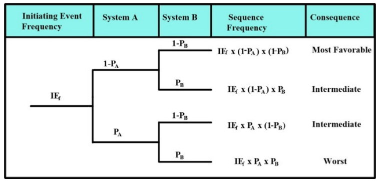
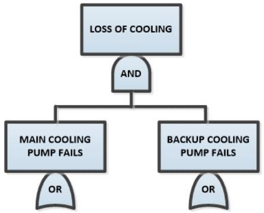


Peer Review of the Shoreham Nuclear Power Plant PRA

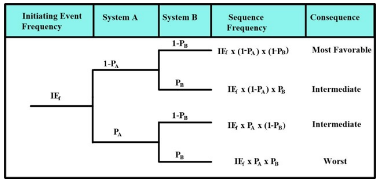


Describes Peer Review of the Shoreham Nuclear Power Plant PRA with emphasis on the internal flooding scenario

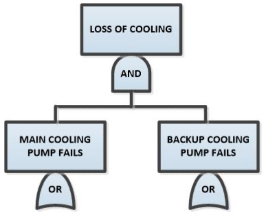
Howard Lambert
FTA Associates
2022



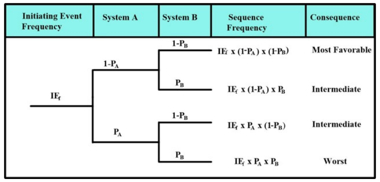
Background of Peer Review



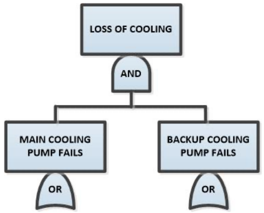
- Peer Review of the draft Probabilistic Risk Assessment (PRA) conducted by SAIC
- Review conducted 1982 by Future Resources Associates
- Shoreham Nuclear Power Plant was built but never licensed
- Shoreham is a BWR-4 Reactor with a Mark II containment located in Suffolk County near Brookhaven National Lab
- Purpose of the Review
  - to provide Suffolk County staff with information about the magnitudes, probabilities, and characteristics of potential large accident releases from the Shoreham reactor.



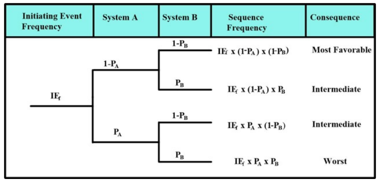
Tasks Shoreham PRA



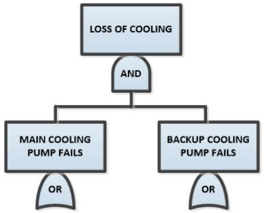
1. Estimation of the frequency of radioactive releases (SAIC)
2. Estimation of the magnitude of the radioactive releases for each accident sequence including the radioactive species and release time (SAIC)
3. Estimation of the consequences to the public of radioactive release to the environment (PL&G)
4. PRA included internal events including internal flooding
5. PRA did not address seismic, fire, high winds and external flooding



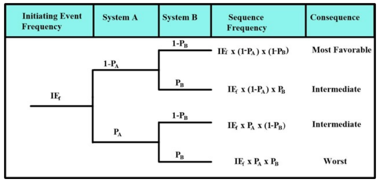
FRA Review included reviewing the following Activities



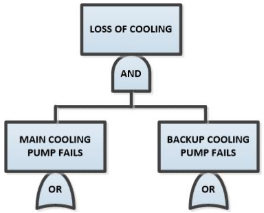
- Level I Activities
  - Systems Analysis
  - Core melt sequences and their frequencies
- Level II Activities --
  - Core melt phenomena
  - Core Melt Damage States called plant damage states
  - Containment Failure Modes
  - Release Categories



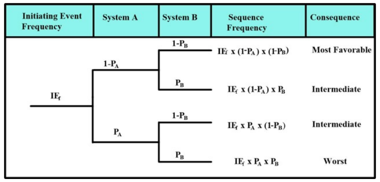
Background and Expertise of the Reviewers



- Four person months for review
- Reviewers were independent – did not work on the study
- Expertise of the reviewers include
  - Systems Analysis
  - thermal hydraulics and core-melt phenomena
  - evaluation of containment performance, including fission product deposition and transport



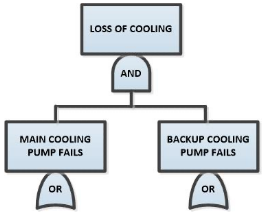
Basic Conclusions of the Study



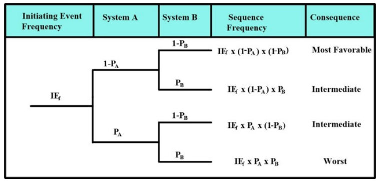
- Study conducted in a competent manner – used Reactor Safety Study methodology with deficiencies corrected

- Issues raised by Peer Review
  - **internal flooding sequences**, concern for which arose out of the review of accident sequences, event trees, fault trees, and system descriptions
  - **core-concrete interactions and vessel melt-through phenomena**, concern for which arose out of the broader review of phenomena that take place during and after core-melting
  - **likelihood of failure to scram on demand**, concern for which arose out of review of the **ATWS** analysis, which the PSA had identified as a key possible contributor to overall risk

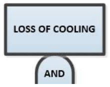
- Presentation focuses on review of **internal flooding sequences and SAIC's reanalysis** as result of the peer review

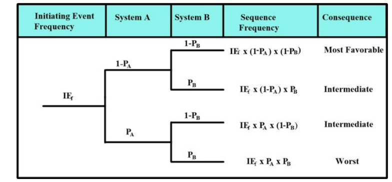


Vulnerability to internal flooding causing damage to ECCS components

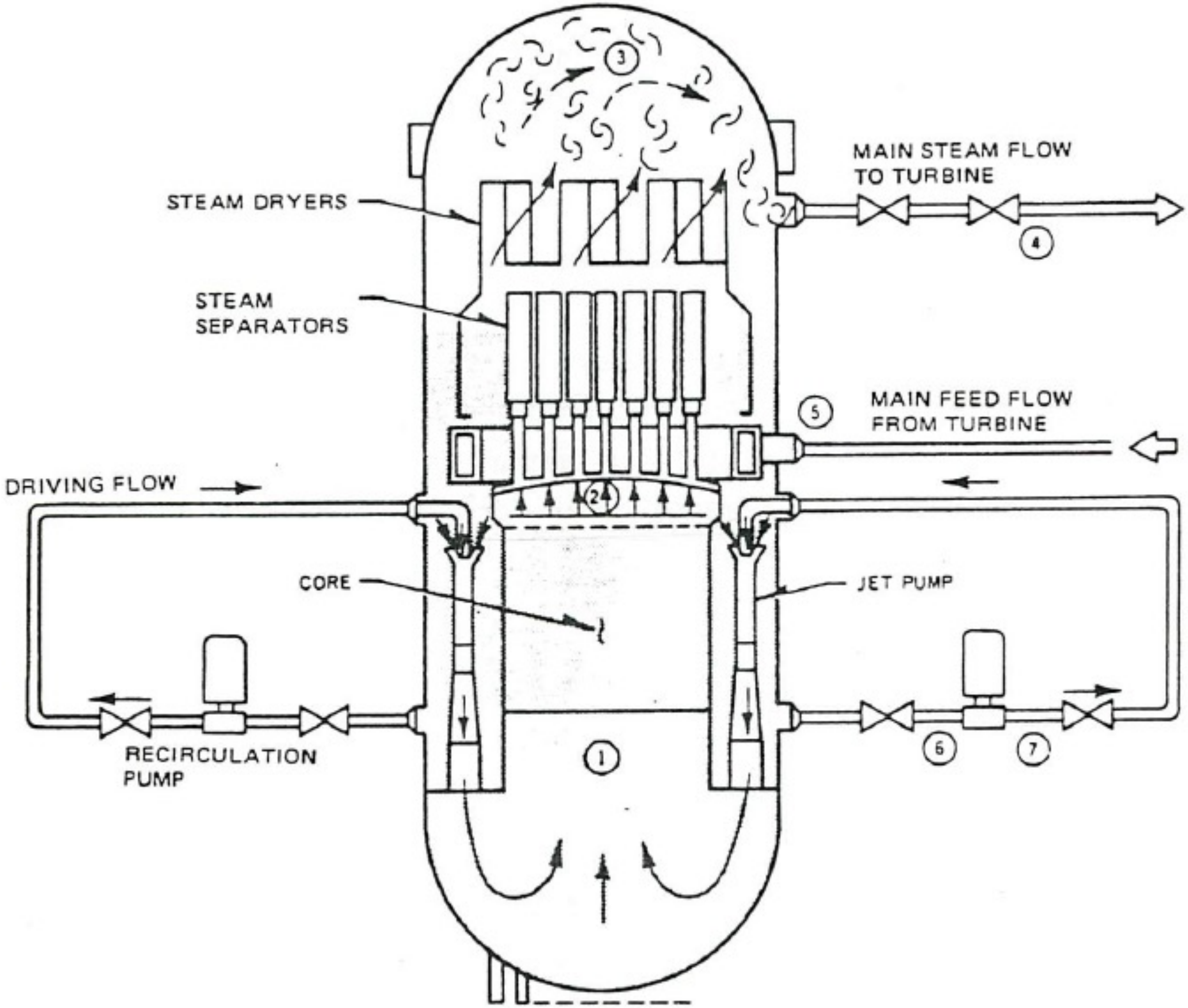


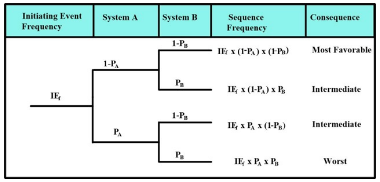
- portions of emergency core cooling systems are disassembled during maintenance (e.g. a pump impeller replacement, valve stem replacement, valve seat adjustments)
- If during this disassembly, human error or set of human errors occur which deisolate the component undergoing maintenance, such as opening a MOV, then release of water through the opened valve can occur flooding level 8 in the reactor building
- Flooding can disable all ECCS components within ~40 minutes



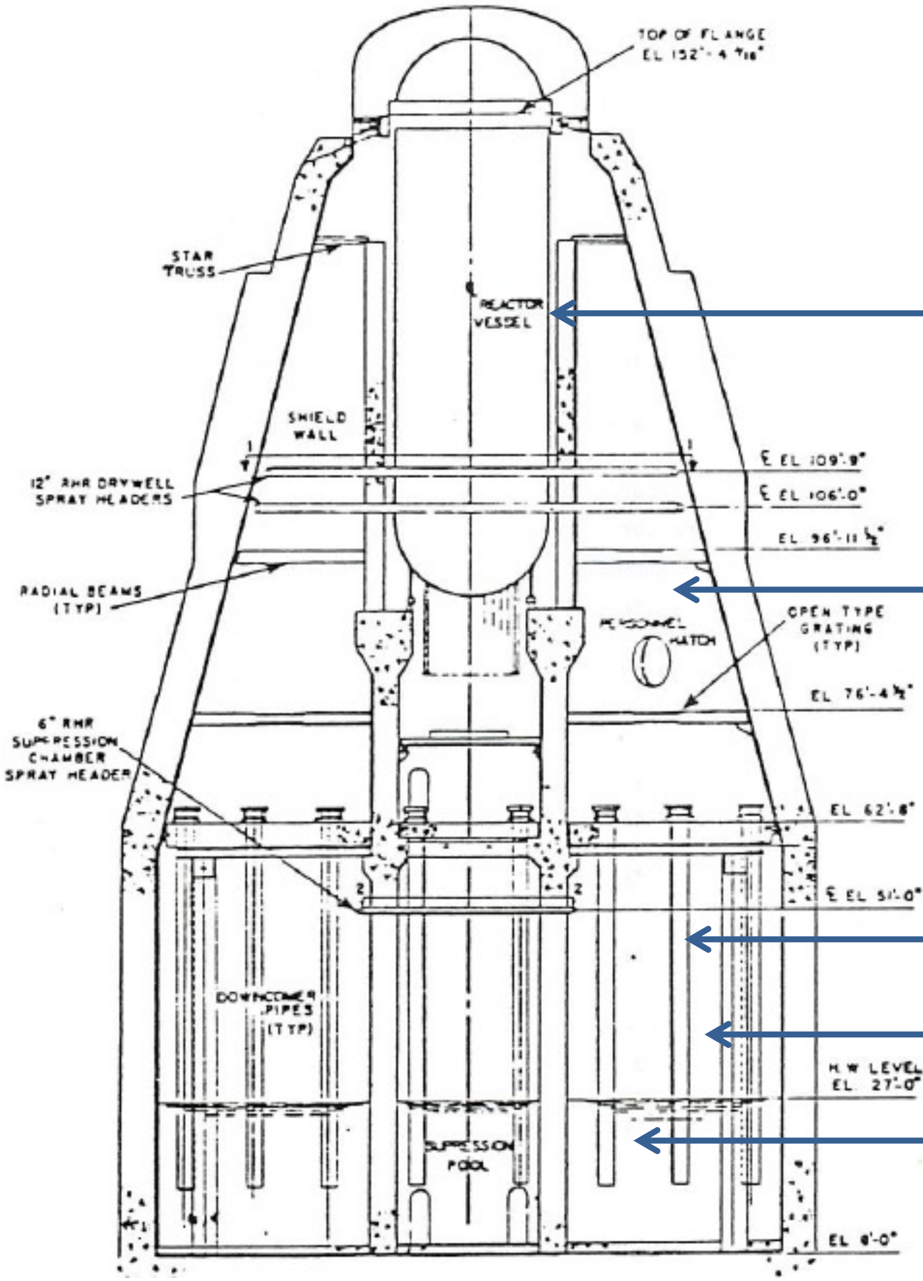


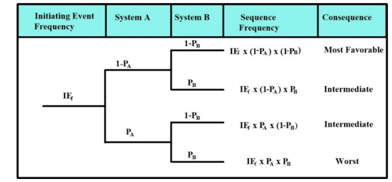
Reactor Core



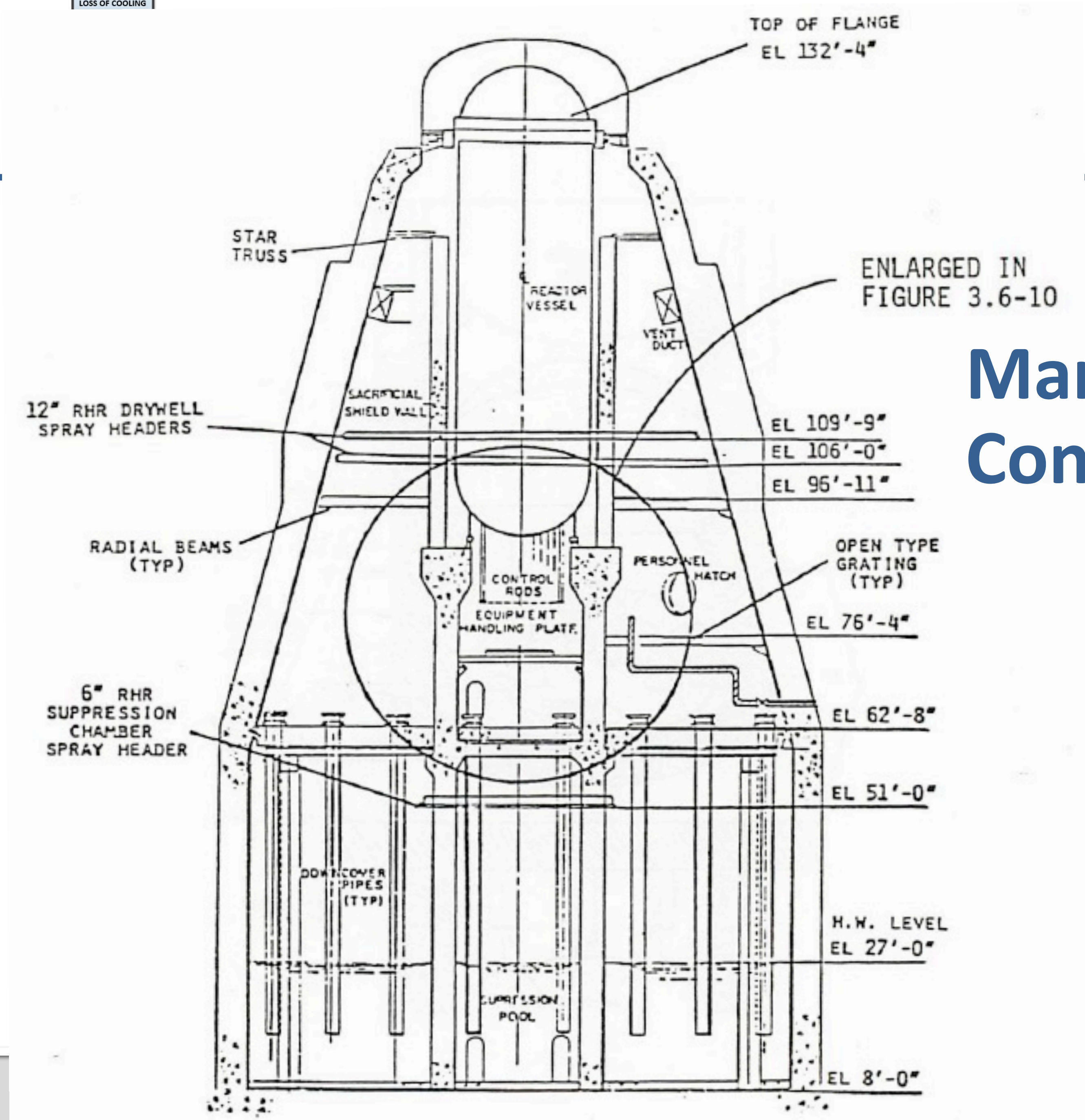


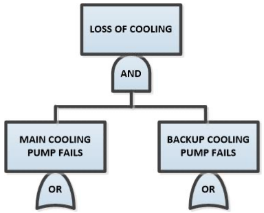
Mark II Containment



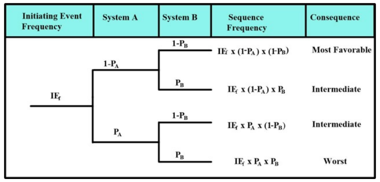


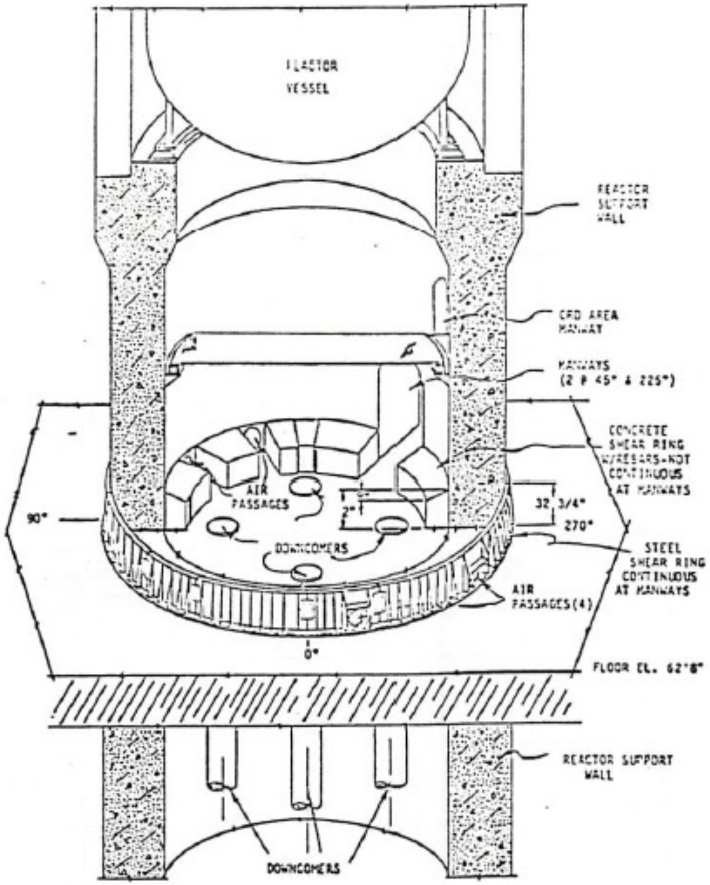
Mark II Primary Containment

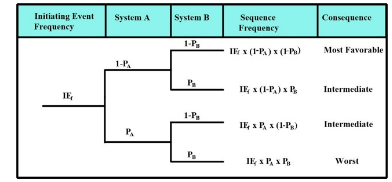




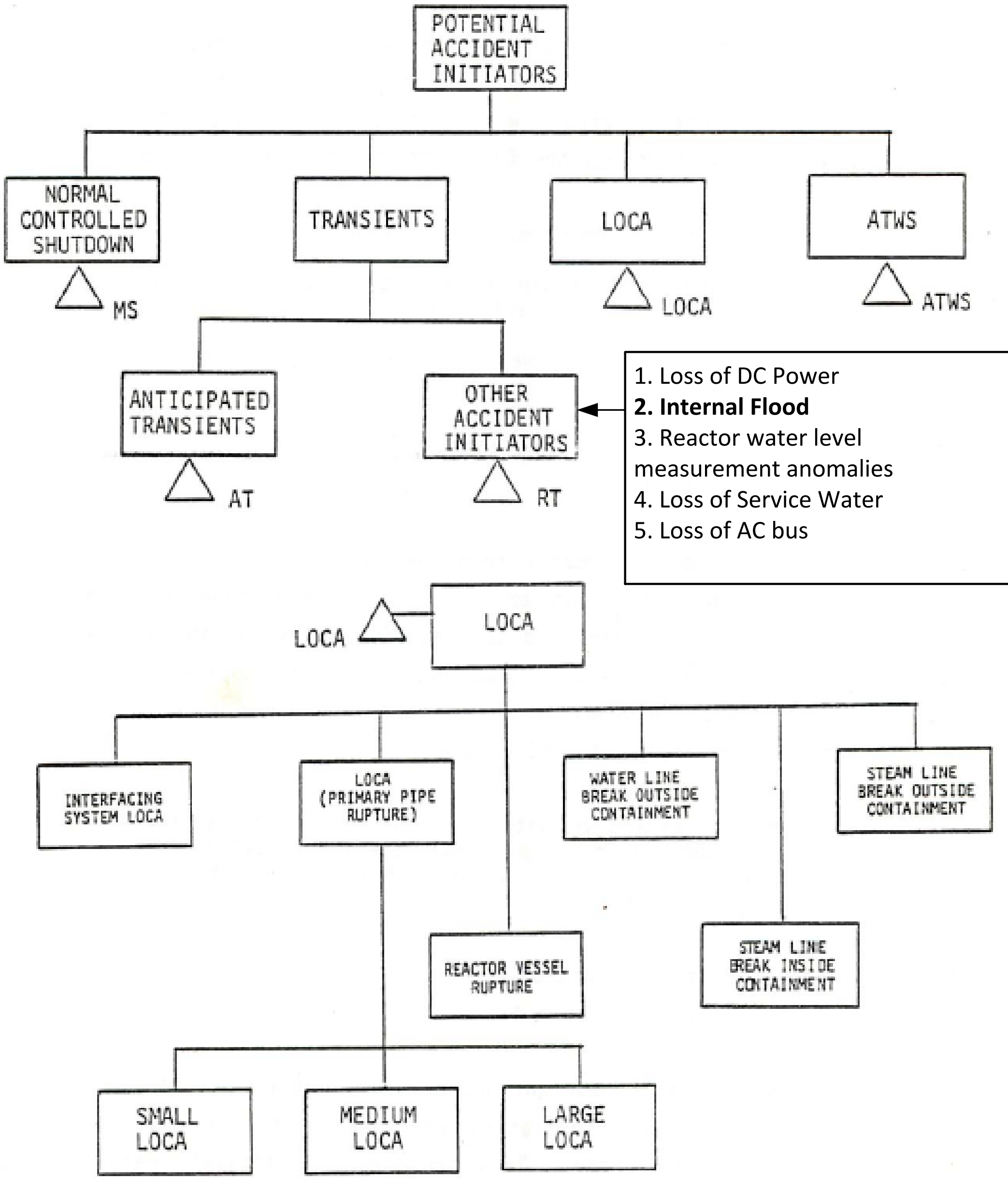
Shoreham Reactor Pedestal

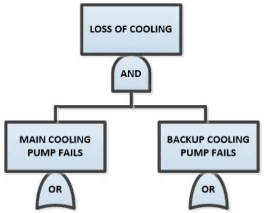




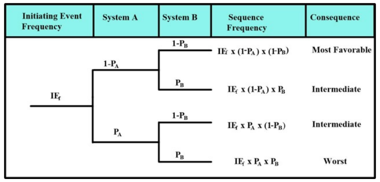


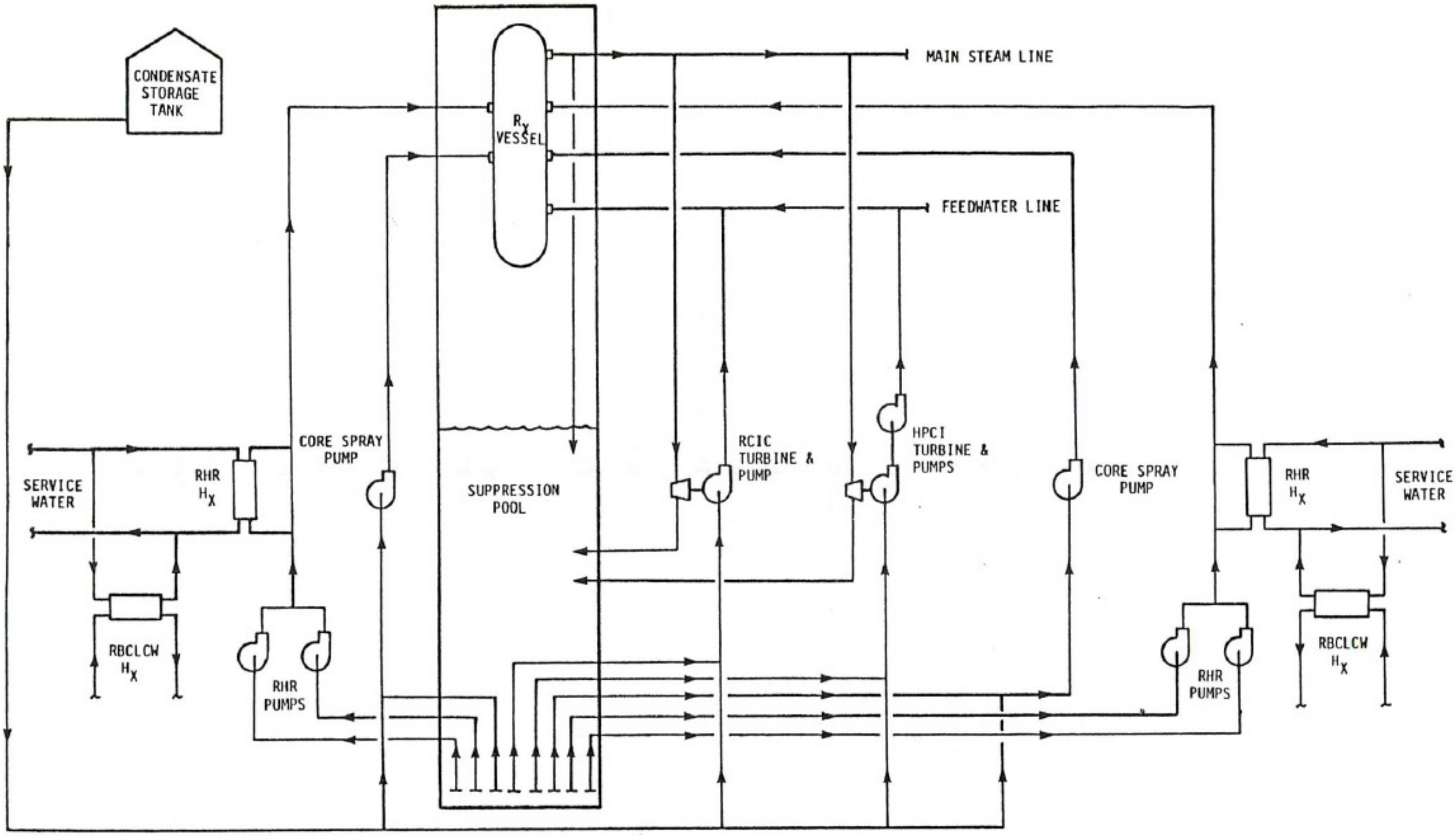
Shoreham Initiating Events

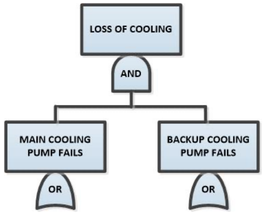




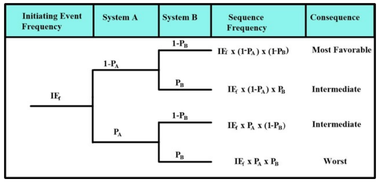
Coolant Injection Systems

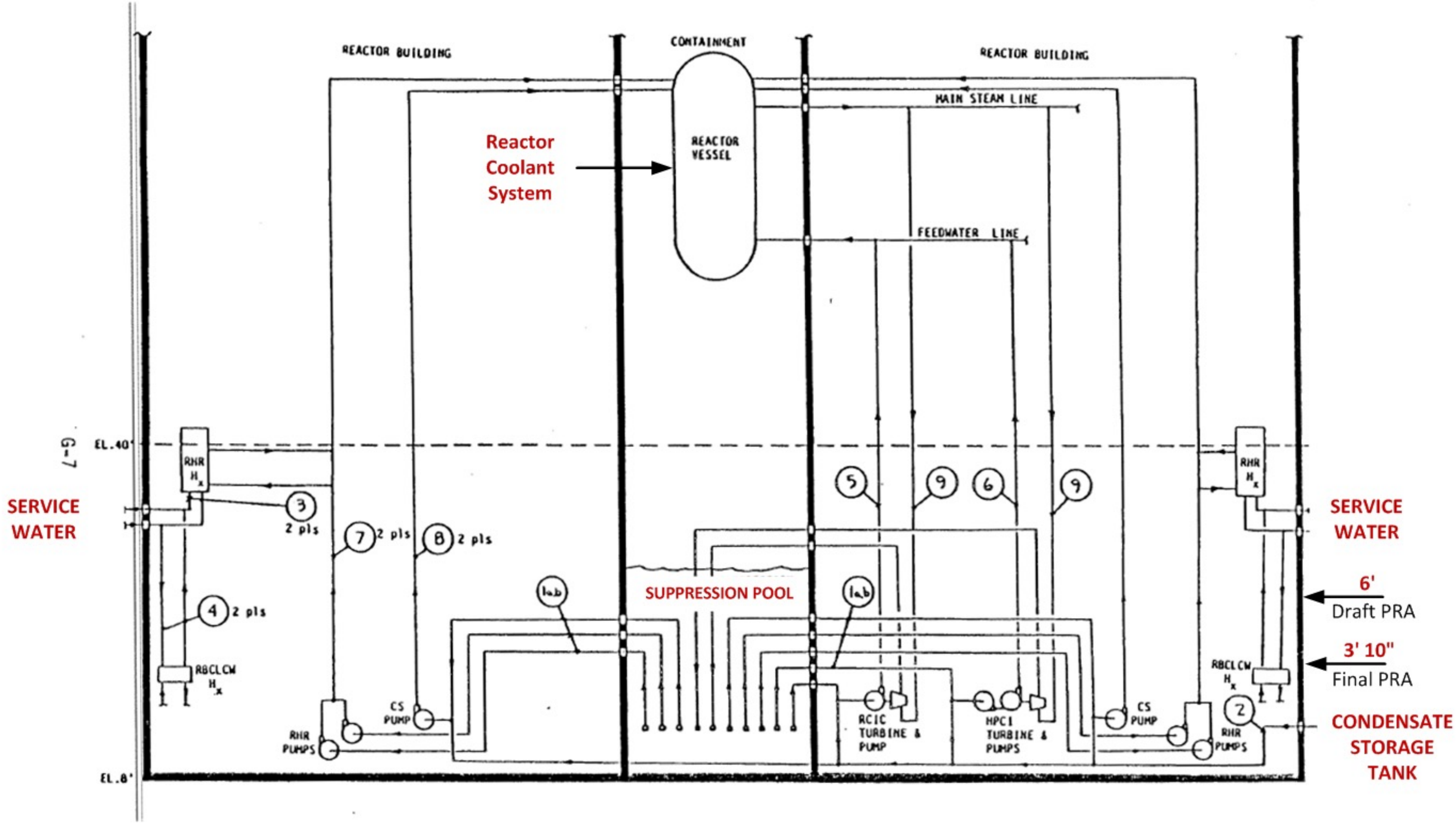


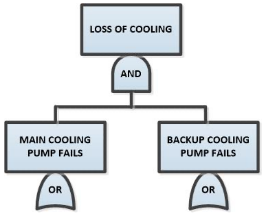




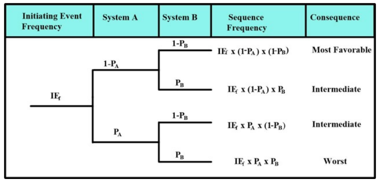
Water Release Sources and Release Points into Reactor Building



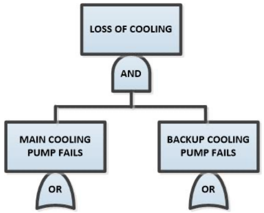




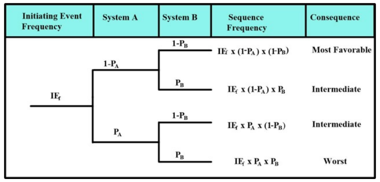
Success Criteria with all ECCS unavailable to prevent core damage

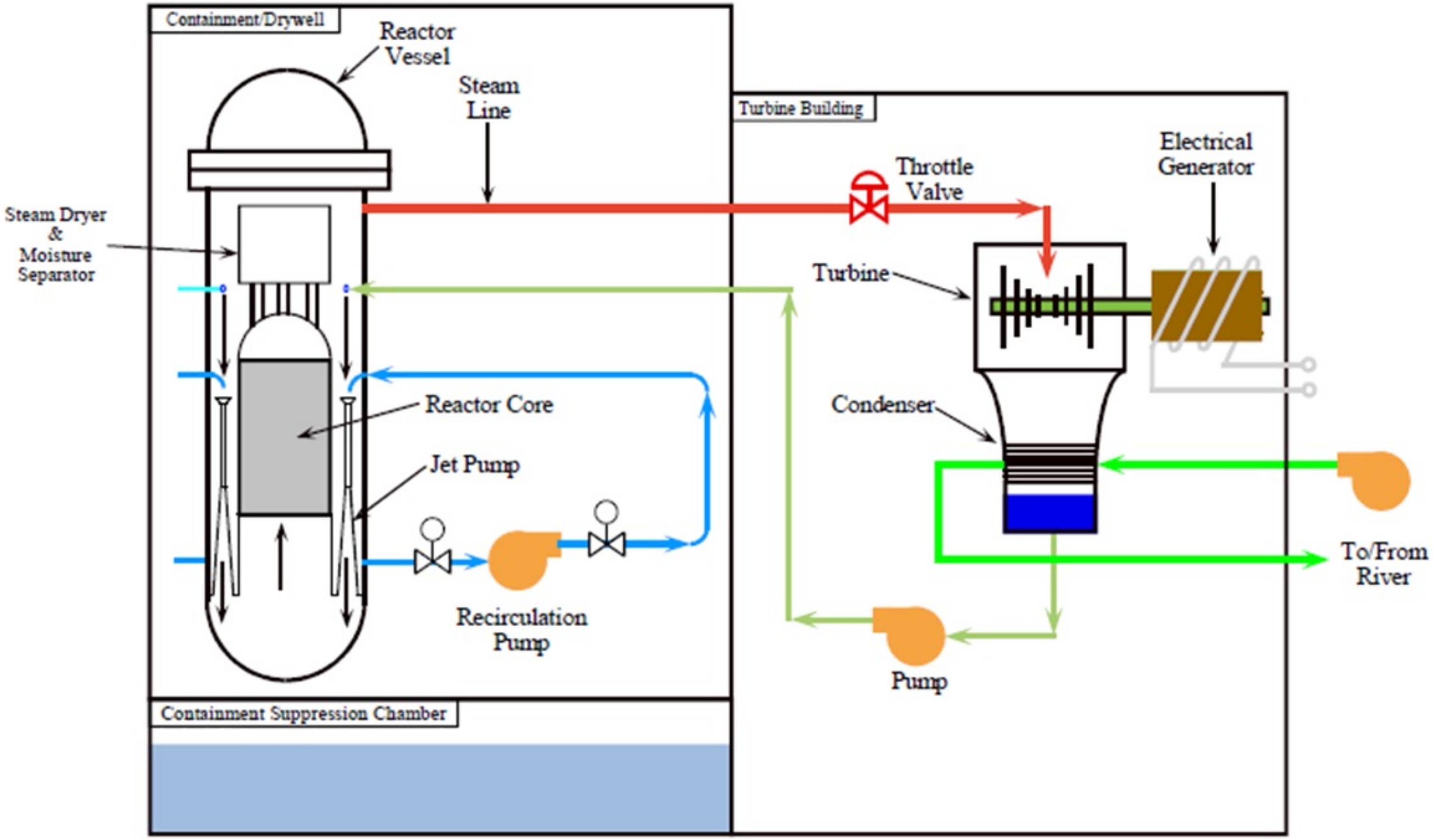


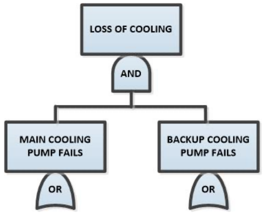
- Coolant makeup AND long term heat removal
- Coolant Makeup
  - Main feedwater system (high pressure) OR
  - Condensate system (low pressure) requires ADS to work
- Long term heat removal (required after 16 hours)
  - Power Conversion System (MSIV must be open)



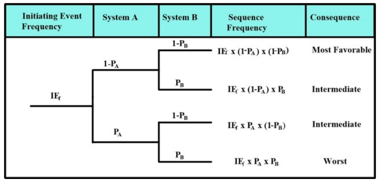
Simplified Schematic Boiling Water Reactor Plant

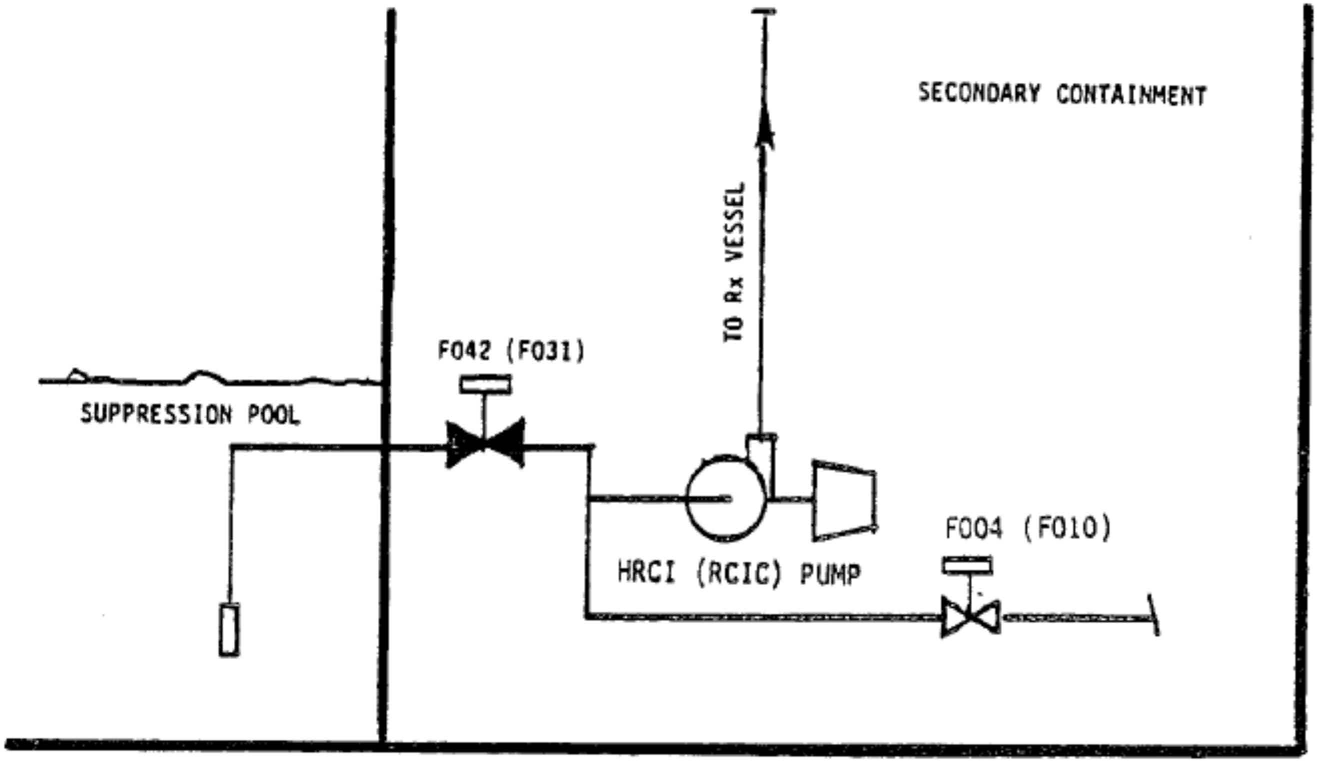


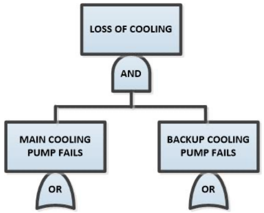




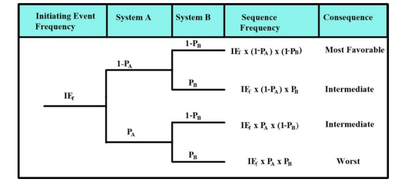
Schematic View of the RCIC and HPCI Piping with the suppression Pool as the Water Source

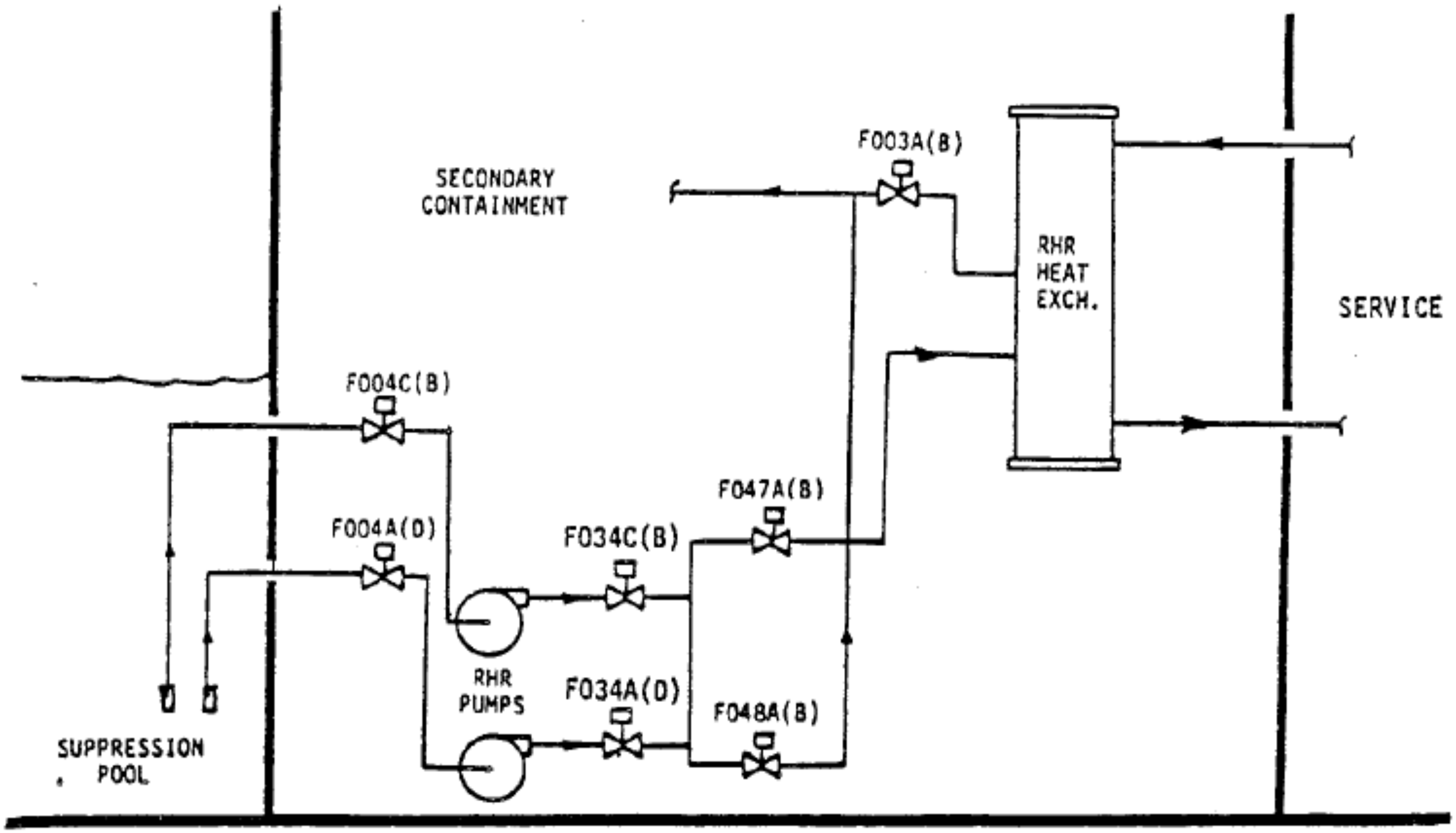


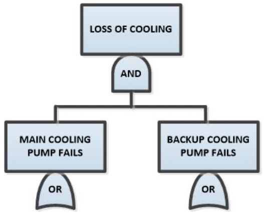




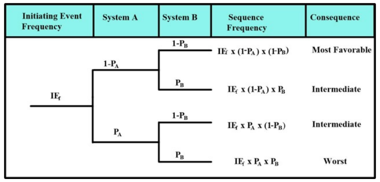
Elevation View of the RHR Piping with the suppression Pool as the Water Source

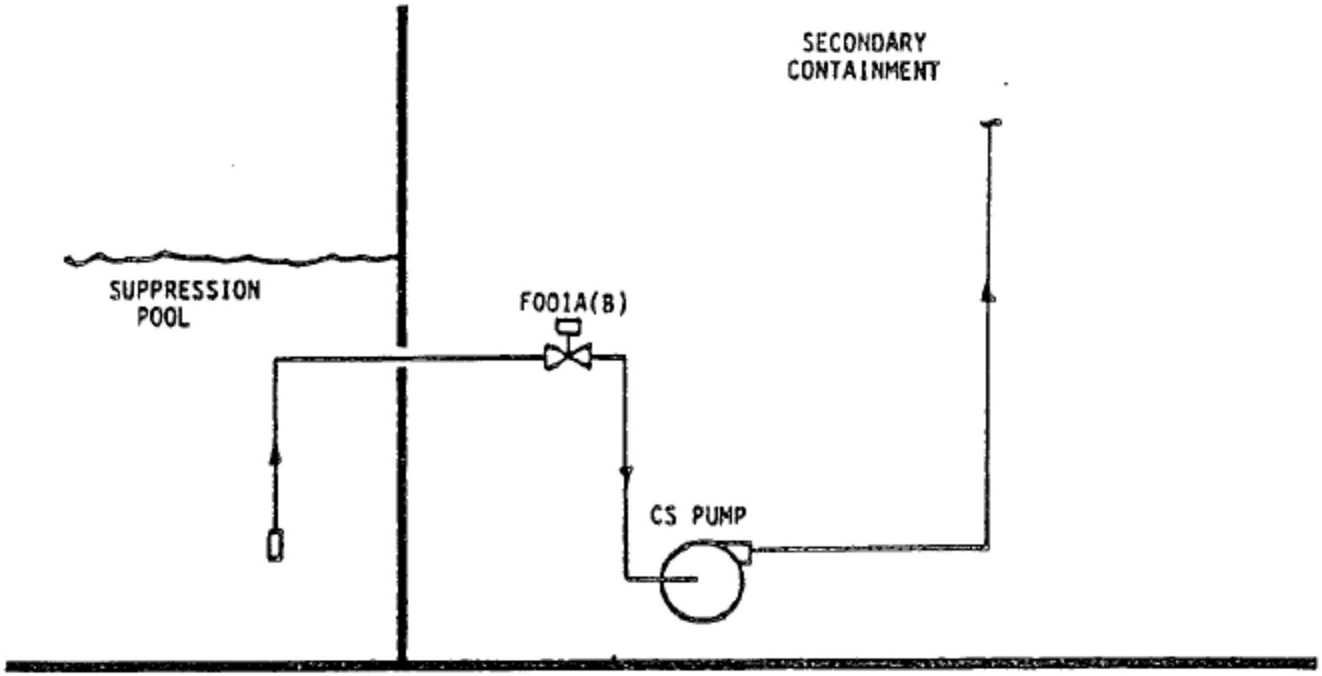




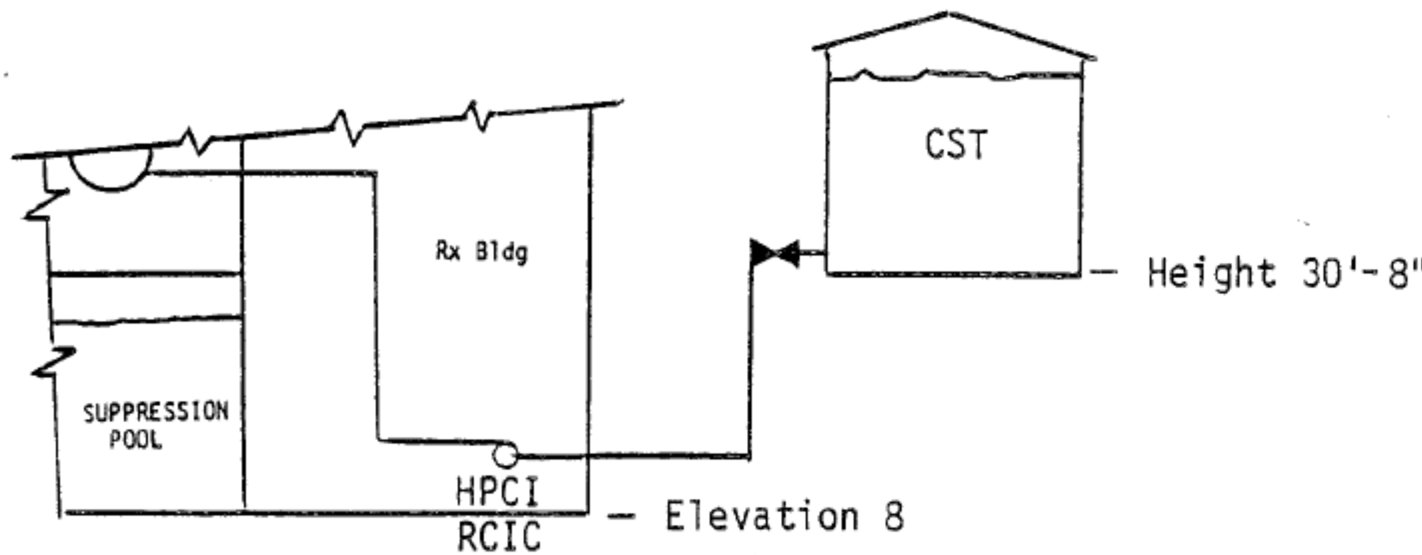


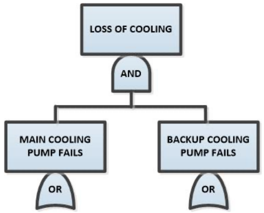
Elevation View of the RHR Piping with the suppression Pool as the Water Source



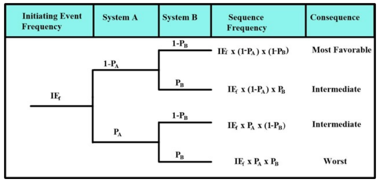


Elevation View of a typical CST pipe connection into Elevation 8 of the reactor building





Flooding Event Tree from draft Shoreham PRA



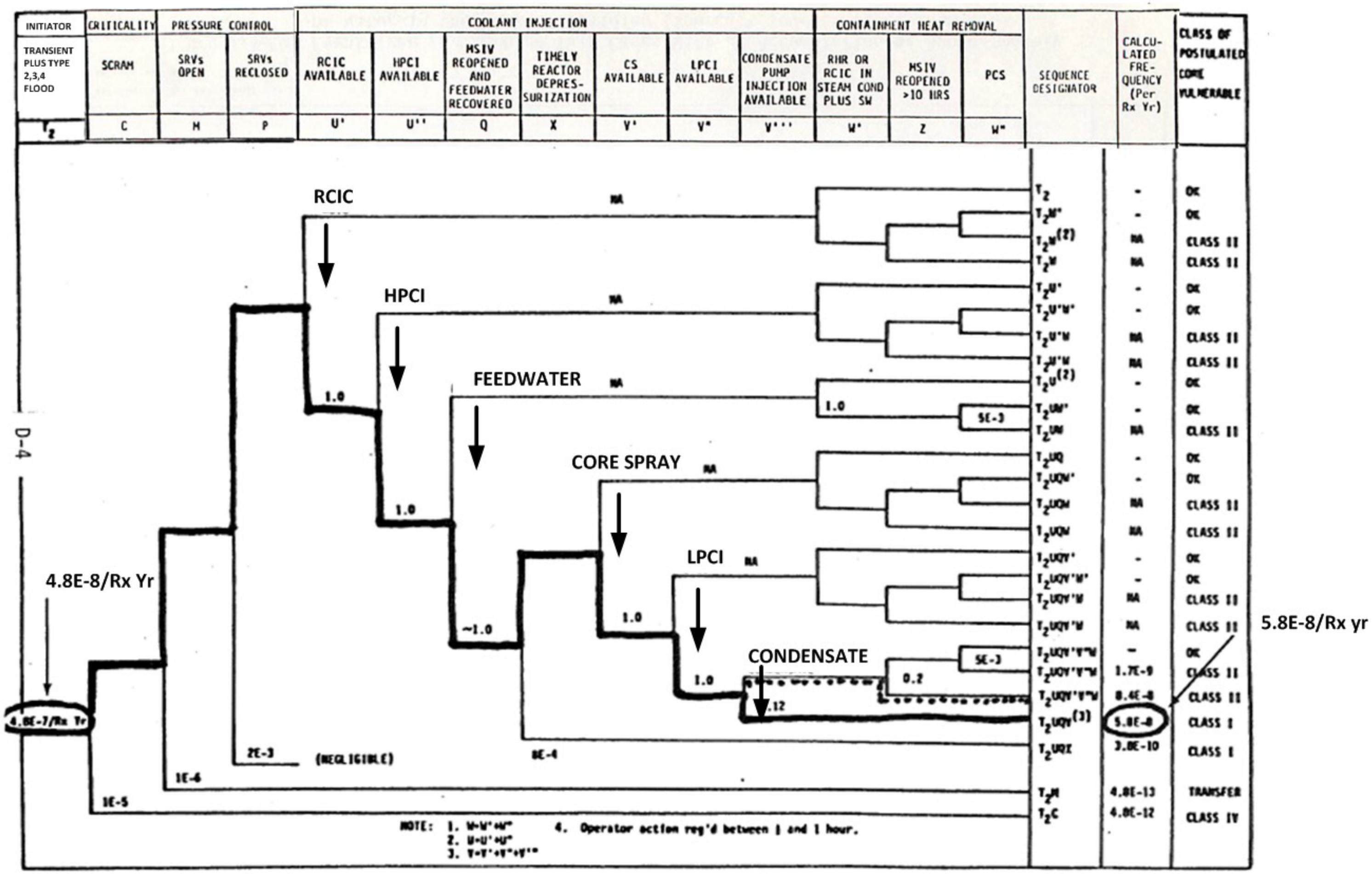
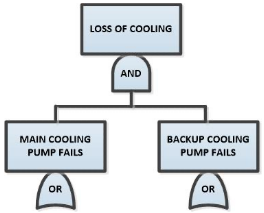
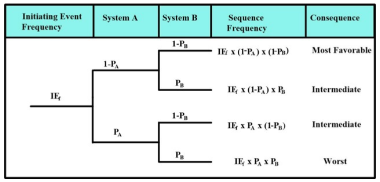


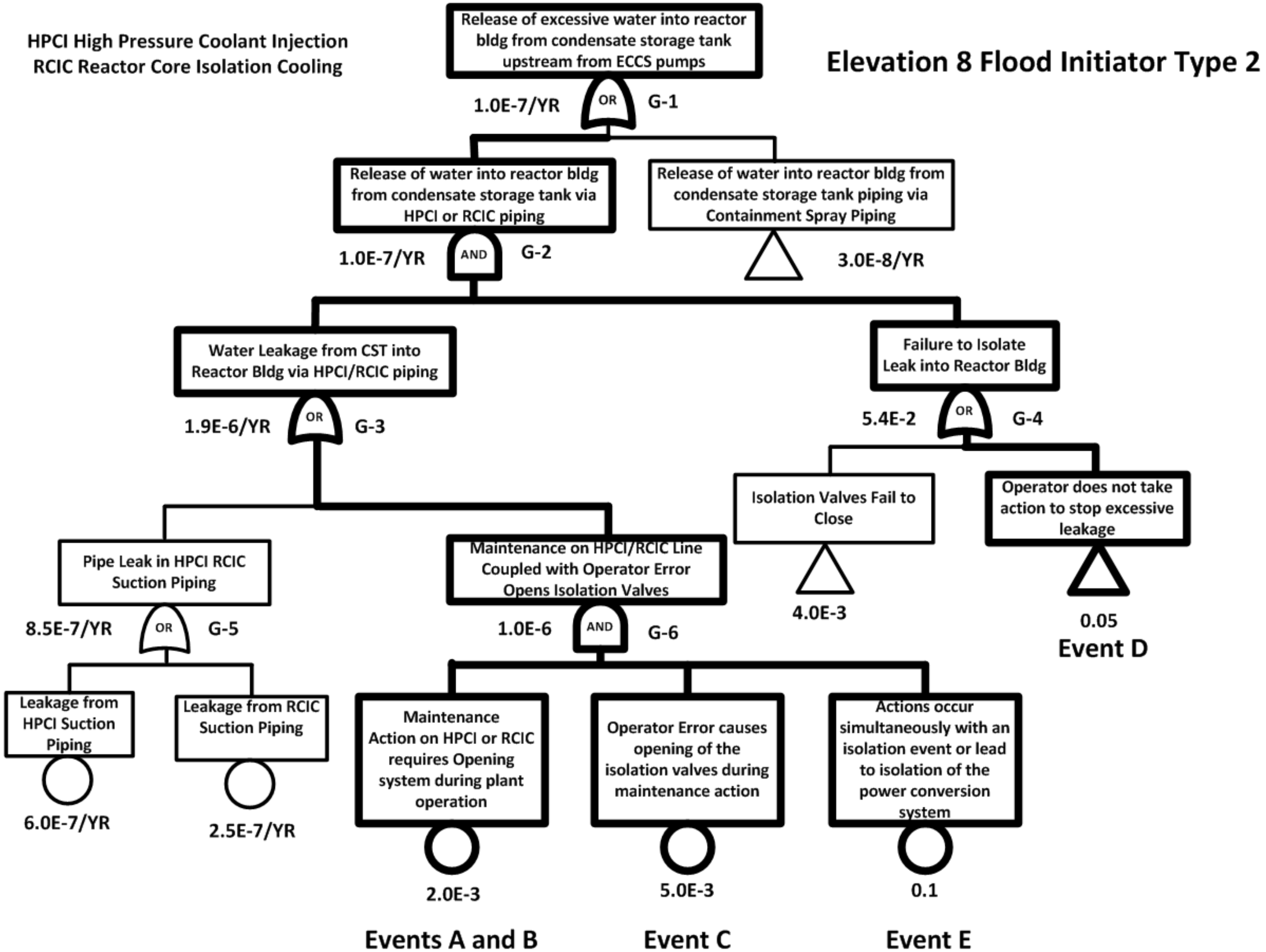
Figure 3.4.23 Event Tree Diagram for Sequences Following a Release of Water into Elevation 8 of the Type 2,3,4 Initiator

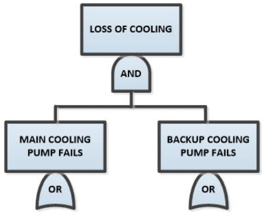
DRAFT - PRELIMINARY



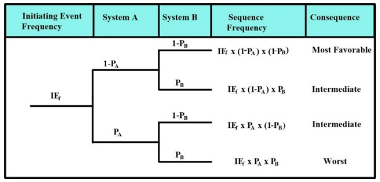
Initiating Event Fault Tree – Water Source CST for RCIC/HPCI disassembly – Draft PRA



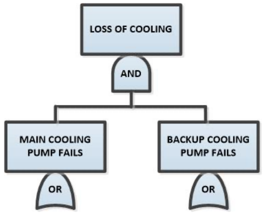




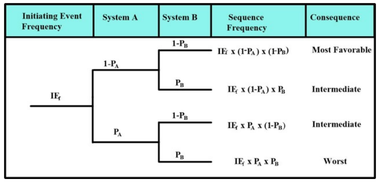
Sequence of Events leading to flooding with isolation of the power conversion system



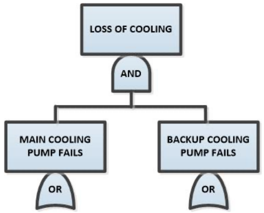
- Event A: On-line maintenance occurs on either RCIC or HPCI
- Event B: System is disassembled for maintenance
- Event C: Operator inadvertently opens an isolation valve during maintenance causing flooding to start (5E-3)
- Event D: Operator fails to reclose the isolation valve within 40 minutes which results in flooding to the six foot level (0.05)
- Event E: Operator erroneously isolates power conversion system during flooding



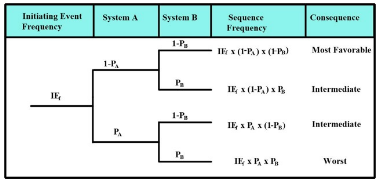
Peer Review Comments on Draft PRA

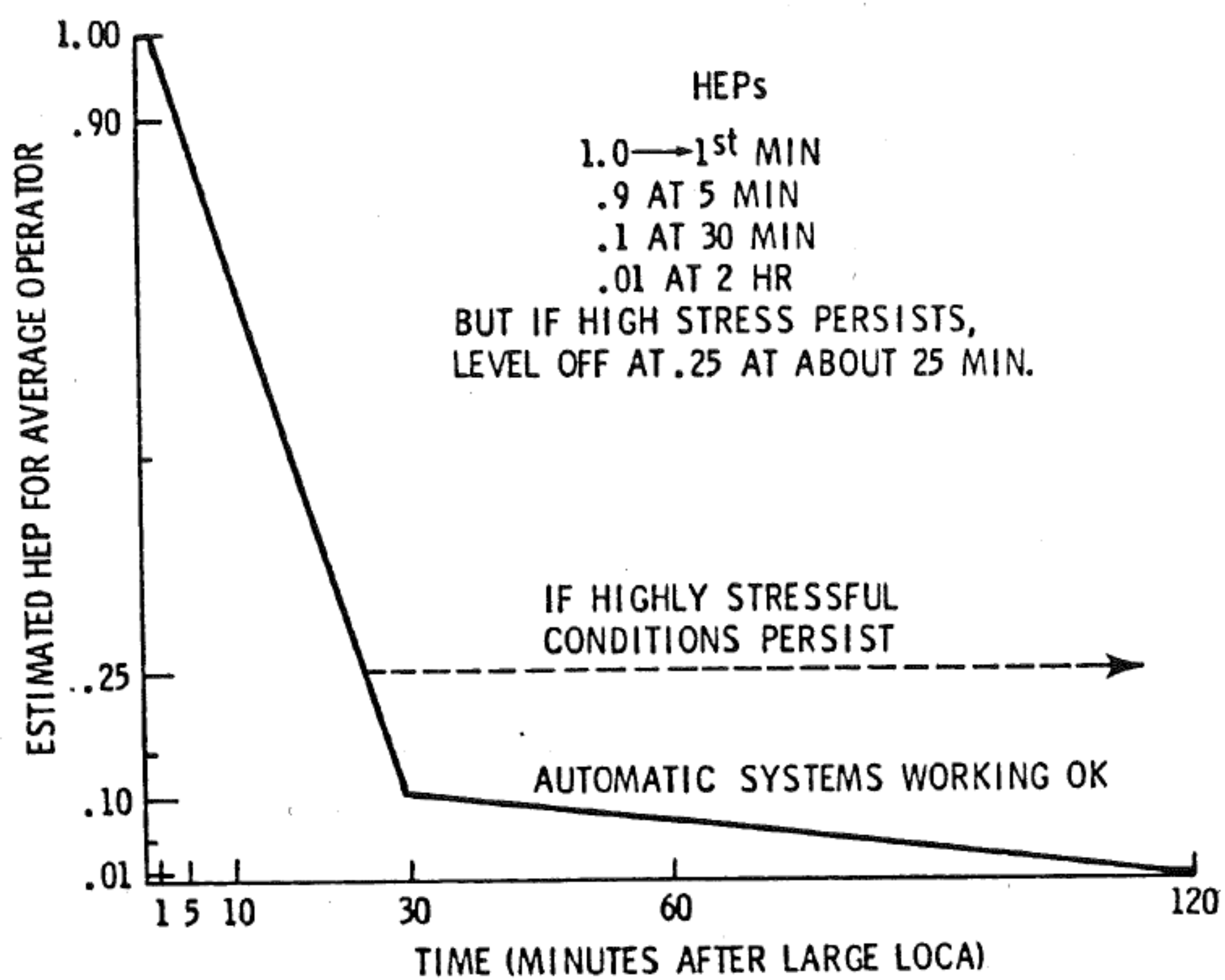


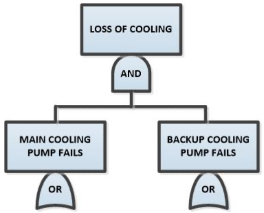
- The units for event A should be given on a frequency basis – the expected number of times maintenance occurs per year – not unavailability (fractional downtime per year)
- Event B should be the fractional number of times the RCIC or HPCI pump is opened for maintenance
- Event D – closing the isolation valve should be done with a high level of stress i.e., failure probability 0.25 versus 0.05



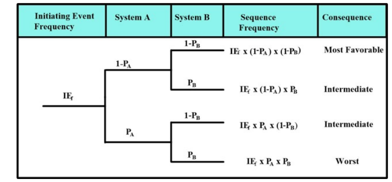
Estimated Human Performance after a highly stressful event such as a large LOCA

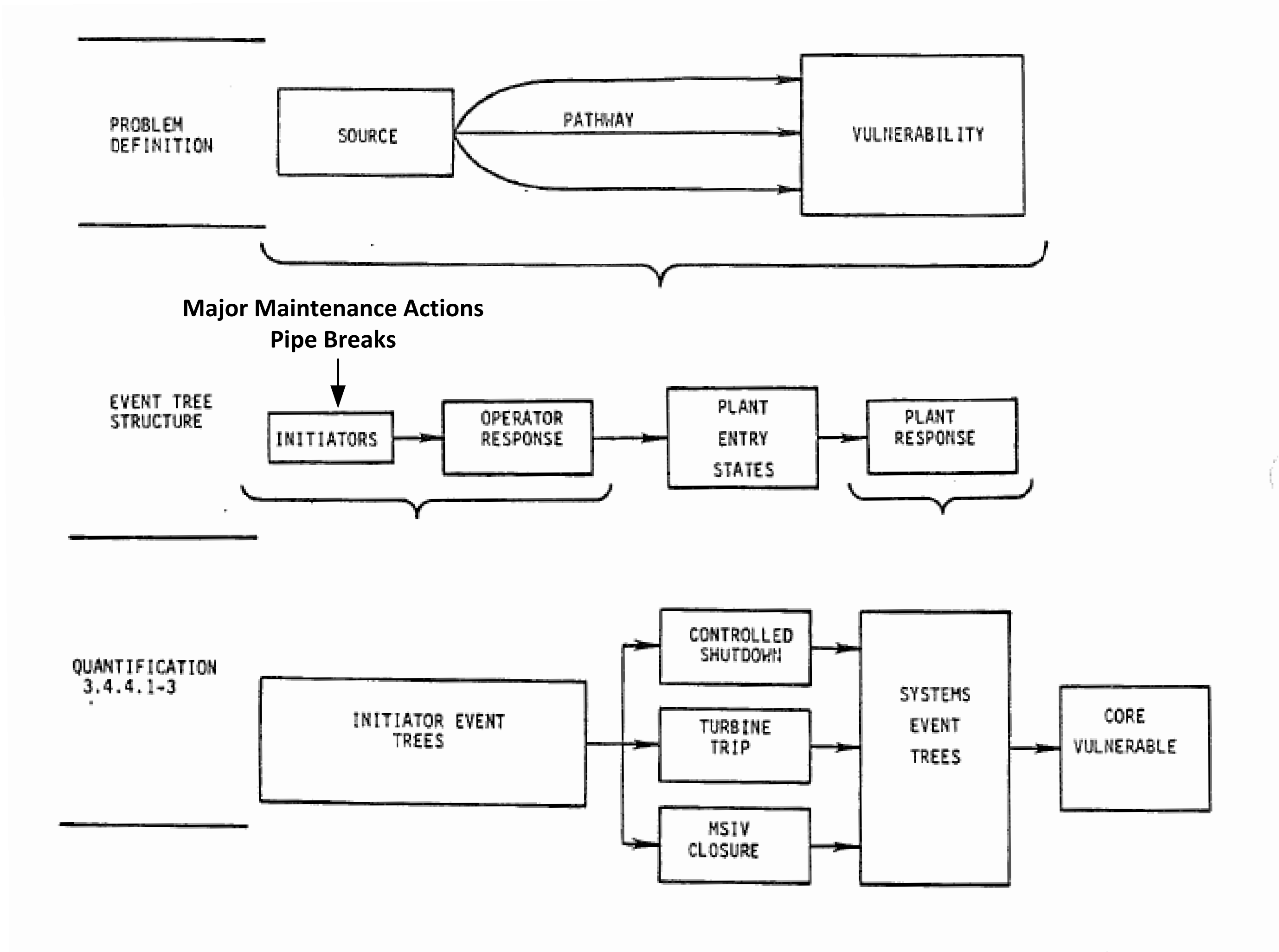


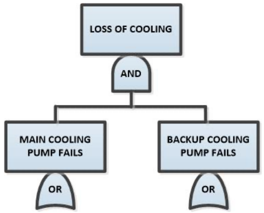




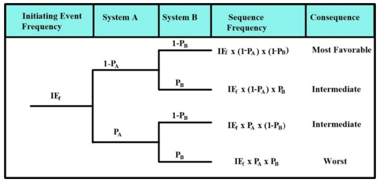
SAIC's Reanalysis



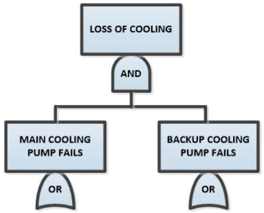




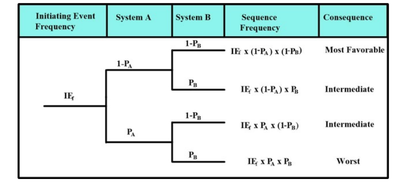
Vulnerability of Equipment to Flooding flood heights (final PRA) [Draft PRA]



- HPCI electrical panels (1’ 10”) [3’ 0”]
- RCIC electrical panels (1’ 9”) [6’ 0”]
- Core Spray electrical panels (3’ 10”) [6’ 0”]
- LPCI (RHR) electrical panels (3’ 10”) [6’ 0”]
- Recirculation pump MG-set fluid coupler water pump motor control centers (1’ 6”) [Trip causes a rise in reactor water level, MSIV closes, and the power conversion is lost as a heat sink] ←



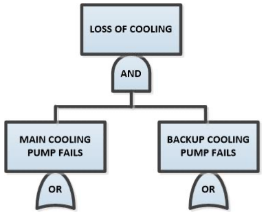
Internal Flooding Initiator Types: Source, Pathway, Flow Rates and Time to Critical Flood Depth



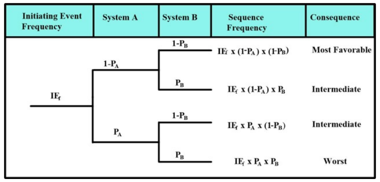
SOURCE	LOCATION	FLOW RATE gpm*	ELEVATION 8 FLOODING TIME, MINUTES* 3'-10" DEPTH
Suppression Pool	HPCI Pump Suction	9,600	17
	RCIC Pump Suction	1,500	110
	LPCI Pump Suction (Max/Large)**	17,000/8,500	9.4/19
	CS Pump Suction	13,000	12
	LPCI Pump Discharge	10,500 (1 Pump Runout)	15
	CS Pump Discharge	6,850 (1 Pump Runout)	23
Condensate Storage Tank (CST)	HPCI Pump Suction (Max/Large)**	12,000/6,000	13/27
	RCIC Pump Suction	2,100	76
	CS Pump Suction (Max/Large)**	12,000/6,000	13/27
	HPCI Pump Discharge	4,350 (Design)	37
Service Water	RHR Heat Exchanger	8,000 (Pump Runout)	20
WFPS	Rupture of 8" Pipe	4,000	40

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

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



Operator Action Interface Events Involved in Reactor Building Flood Sequences

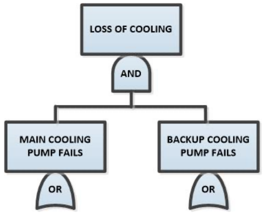


1. Event P -- Operator removes power from isolation valves.

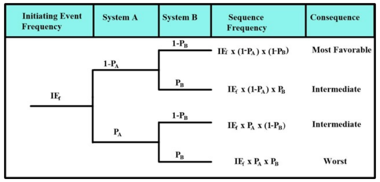
2. Event E -- Operator maintains Motor Control Center (MCC) isolation of the valves.

3. Event E -- Operator maintains control room isolation of the valves.

4. Event A -- Operator diagnoses and isolates flood within the time available for him to act.



Task Analysis -- HPCI pump maintenance and the operation of the isolation valve (event P)

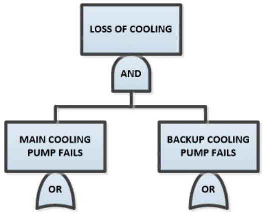


1. Maintenance on the high pressure core injection pump:

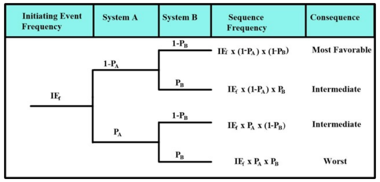
a. Remove power from isolation valve HVXXXX by **racking out breaker**.
b. Close steam isolation valve SHVXXXX and remove power.
c. Drain turbine.
d. Disconnect turbine drive shaft from pump.

2. Allow the pump to cool down to 80°F after closing isolation valve HVXXXX:

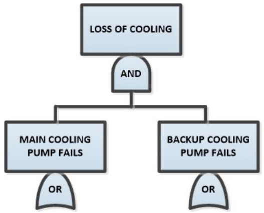
a. Disassemble pump, etc.
b. Carry out repair to pump, replace seals, etc.
c. Assemble pump, connect turbine drive shaft, etc.
d. **Rack in breakers** to valve HVXXXX and steam isolation valve. e. Open isolation valve HVXXX.



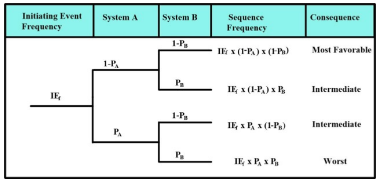
Human Error Probability for operator removes power from isolation valve (event P)



NUREG/CR-1278	Probability	Reference
1. Failure to *carry* out plant policy when there is no check on a person.	0.01(0.005 to 0.05)	p 20-31,Table 20-22, item 1
2. Error of Omission in Use of Written Procedures in *non-passive Tasks* with check off Long List ≥ 10 items	0.003 (0.001 to 0.01) Item 2	p 20-29,Table 20-20
3. Failure to follow established procedures or policies in valve changes or restoration	0.01 (0.005 to 0.05) Item 5	p 20-23,Table 20-15
4. Change or restore wrong MOV switch or circuit breakers in a group of similar appearing items.	0.003 (0.001 to 0.01) Item 7	p 20-21,Table 20-14,



Time Dependent human error probability single event versus multiple events – Event A



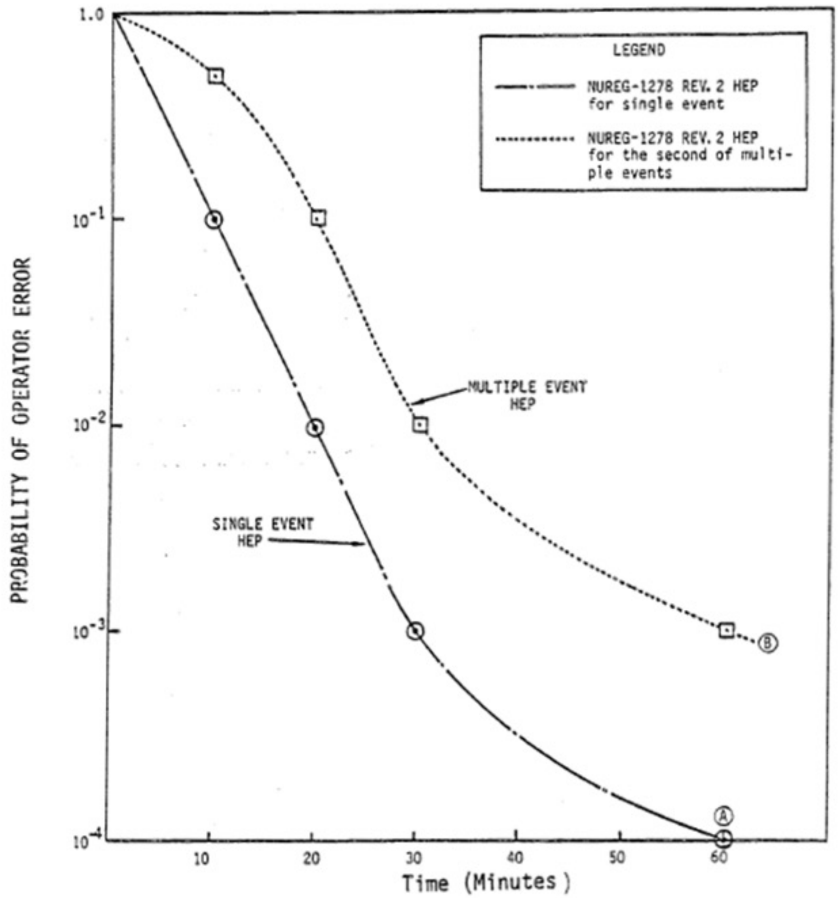
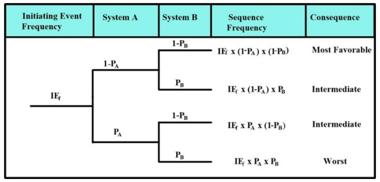
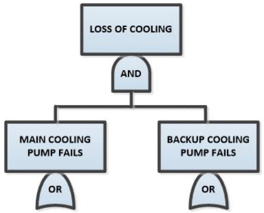


Figure G.4-1 Comparison of the HEPs Associated with Operator Actions for Singular Events and Coincident Multiple Events

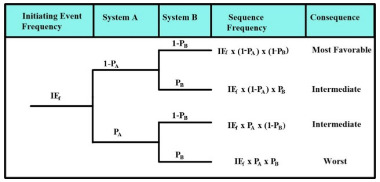


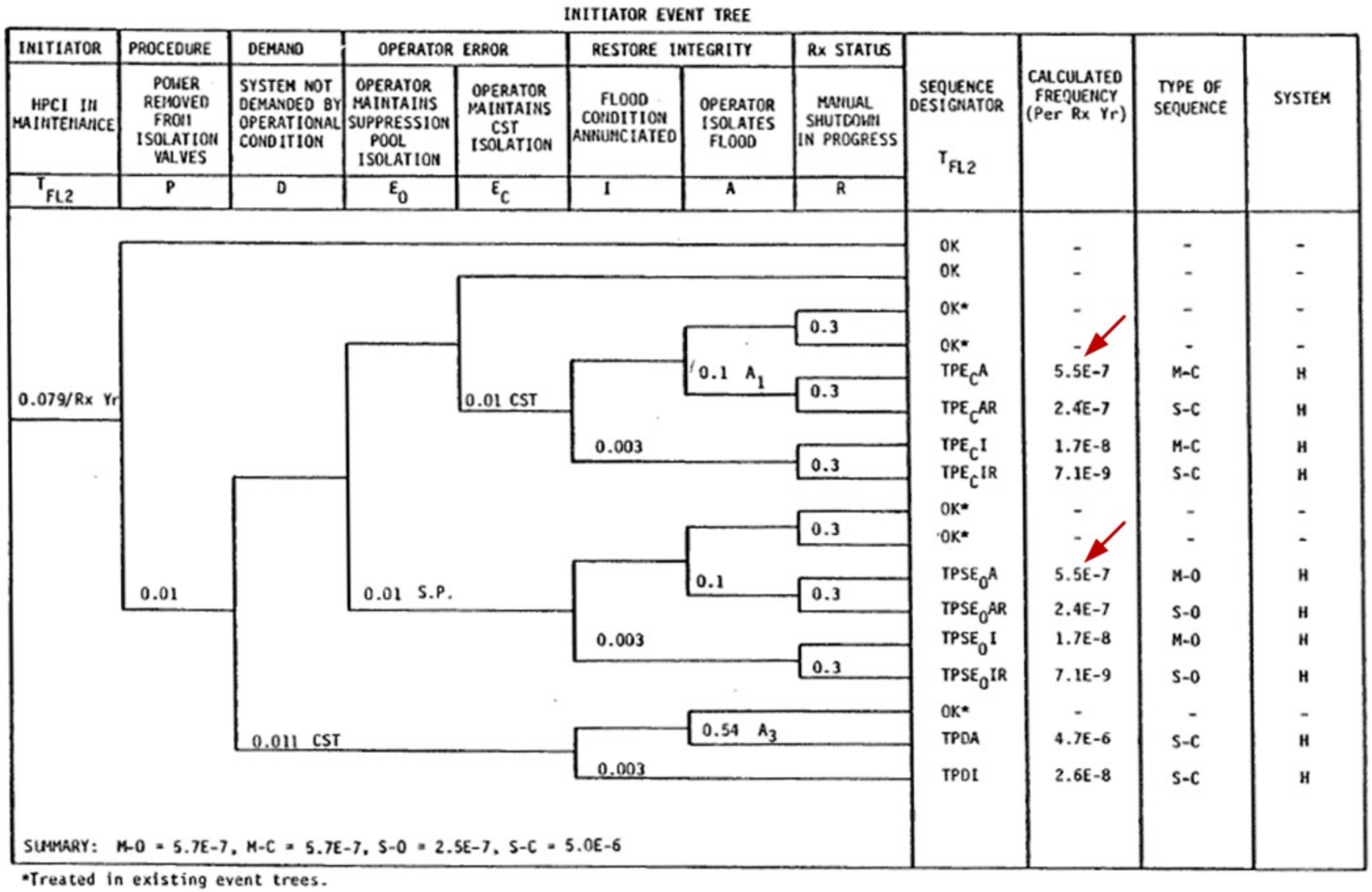
Flooding Sequence Initiators

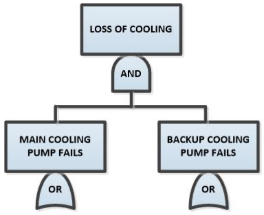
INITIATOR		INITIATOR FREQUENCY (per Rx Yr)	INITIATOR EVENT TREE FIGURE NO.
DESIGNATOR	DESCRIPTION		
Maintenance Related Initiators			
T_{FL1}	RCIC in maintenance	7.9E-2	3.4-21
T_{FL2}	HPCI in maintenance ←	7.9E-2	3.4-22
T_{FL3}	CS in maintenance	4.0E-2	3.4-23
T_{FL4}	LPCI in maintenance	8.4E-2	3.4-24
T_{FL5}	SW in maintenance	4.2E-2	3.4-25
Pipe Failure Related Initiators			
T_{FL6}	HPCI discharge break	3.5E-5	3.4-26
T_{FL7}	CS discharge break	6.9E-5	3.4-27
T_{FL8}	LPCI discharge break ←	2. E-4	3.4-28
T_{FL9}	SW discharge break	1.4E-4	3.4-29
T_{FL10}	WFPS break	1.2E-5	3.4-30
T_{FL11}	RCIC suction break (maximum flow rate)	1.8E-6	3.4-31
T_{FL12}	HPCI suction failure (maximum flow rate)	1.8E-6	3.4-32
T_{FL13}	HPCI suction failure (large break)	3.5E-6	3.4-33
T_{FL14}	CS suction failure (maximum flow rate)	2.5E-6	3.4-34
T_{FL15}	CS suction failure (large break)	4.9E-6	3.4-35
T_{FL16}	LPCI suction failure (maximum flow rate)	2.6E-6	3.4-36
T_{FL17}	LPCI suction failure (large break)	5.2E-6	3.4-37



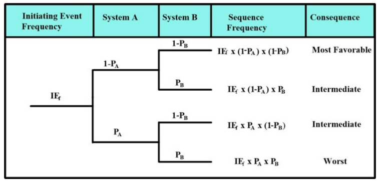
T_{FL2} Initiator Event Tree for Postulated Flooding Sequence by an error during HPCI in Maintenance

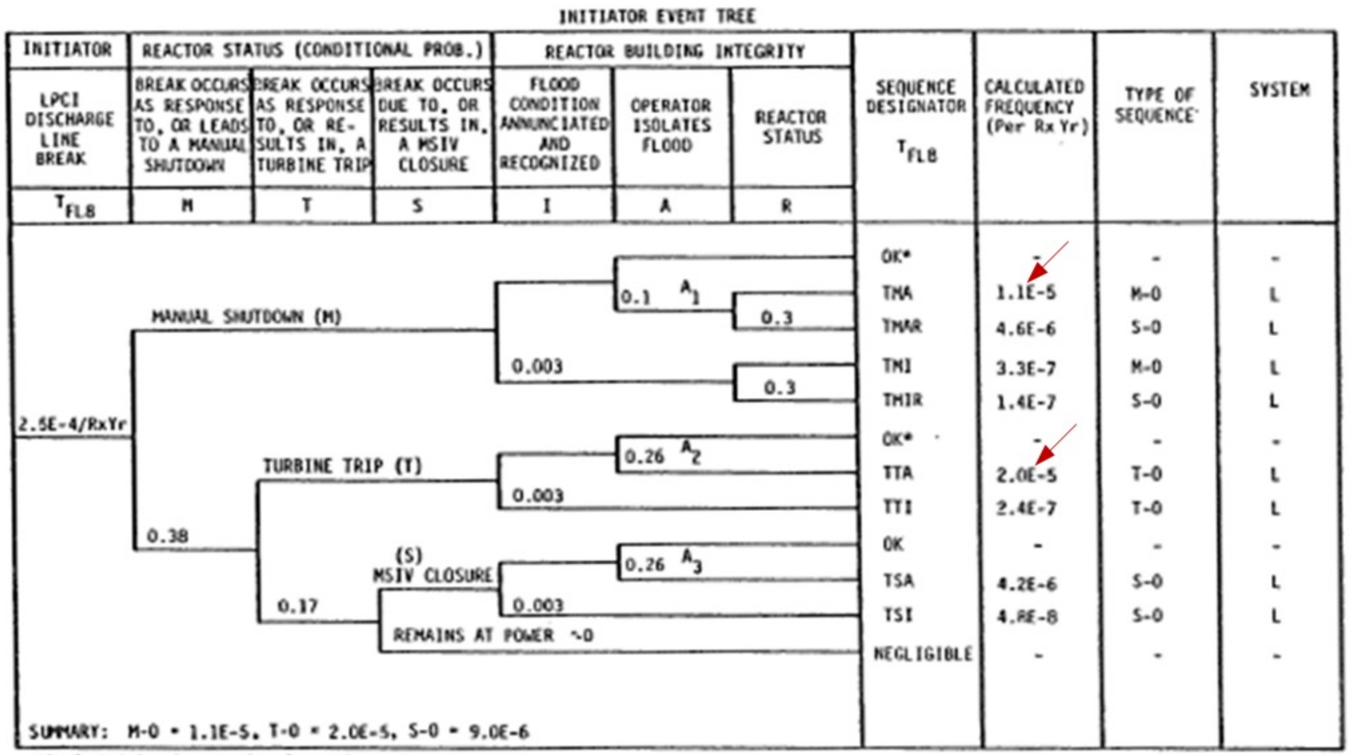




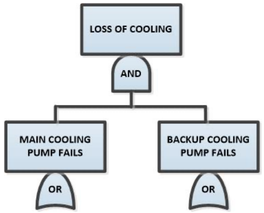


T_{FL8} Initiator Event Tree for Postulated Flooding Sequence by a service Water Line Break

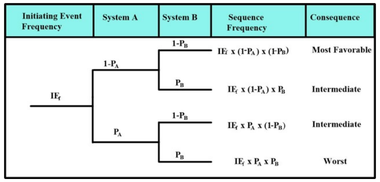




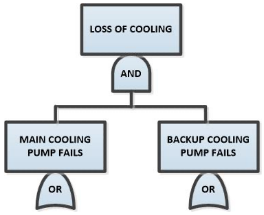
*Included in the previously evaluated event trees.



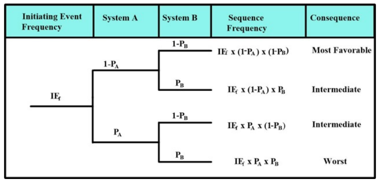
Summary of the initiator event tree results compiled to provide input to the systemic event trees

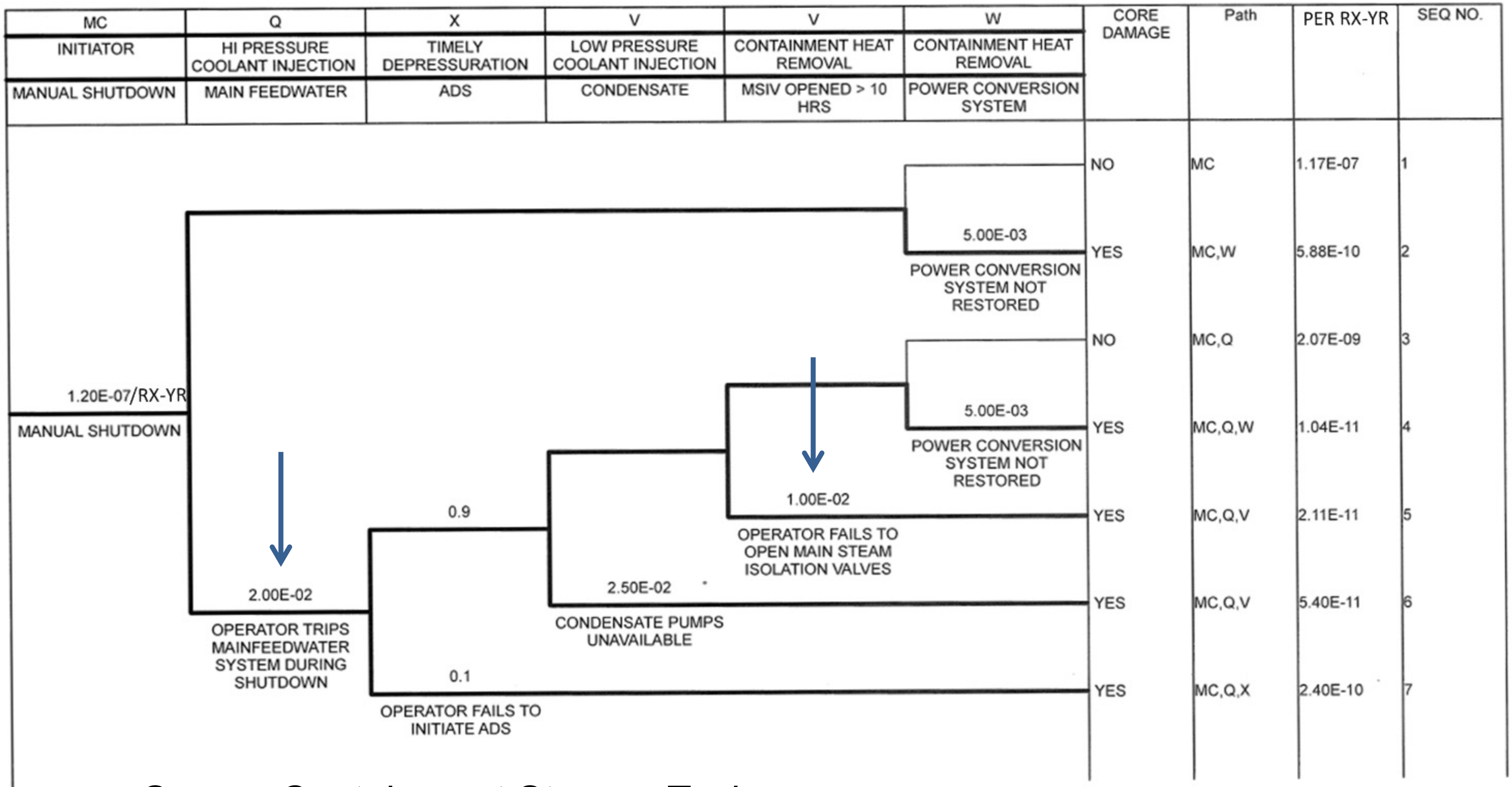


	INITIATOR	
DESIGNATOR	DESCRIPTION	CALCULATED FREQUENCY (PER RX YR)
M-0	Controlled Manual Shutdown with the Flood Source from the Suppression Pool, Service Water, or Fire Supp.	1.6E-5
M-C	Controlled Manual Shutdown with the Flood Source from the CST	8.2E-7
T-0	Turbine trip with the Flood Source from the Suppression Pool, Service Water, or Fire Supp.	2.2E-5
T-C	Turbine Trip with the Flood Source from the CST	3.4E-9
S-0	MSIV Closure with the Flood Source from the Suppression Pool , Service Water or Fire Supp.	1.7E-5
S-C	MSIV Closure with the Flood Source from the CST	5.5E-6

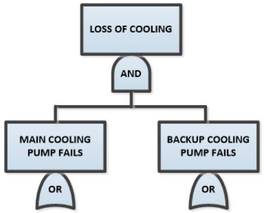


System event tree for manual shutdown with greater than 3’ 10” water in reactor building

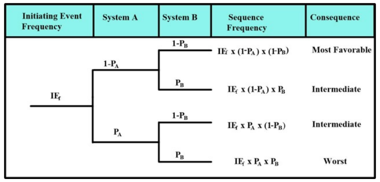


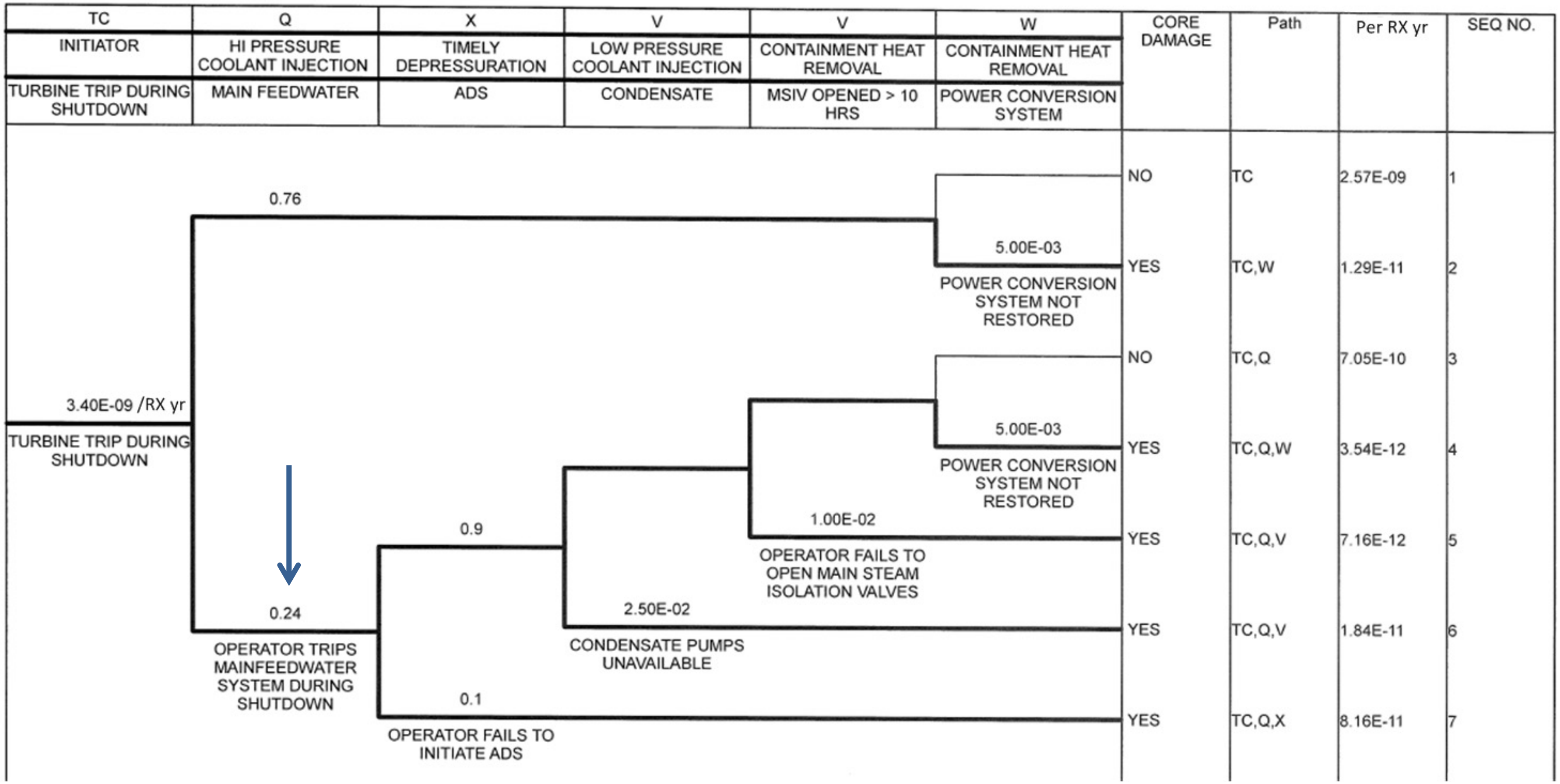


Source Containment Storage Tank

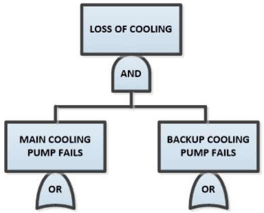


System event tree for turbine trips with greater than 3' 10" water in reactor building

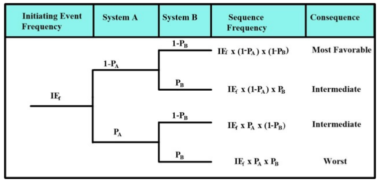


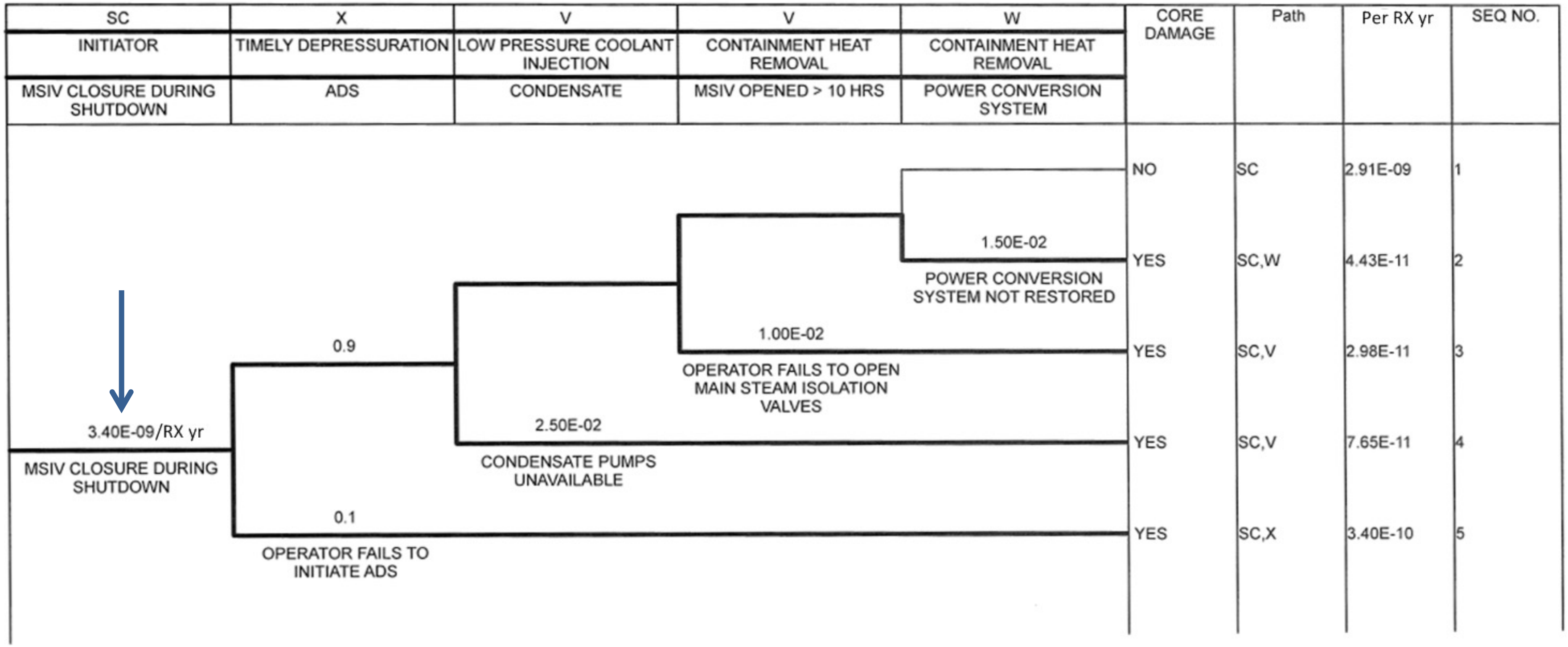


Source Containment Storage Tank

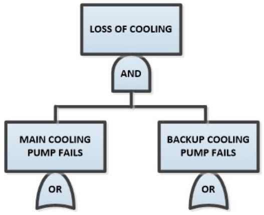


System event tree for MSIV closures with greater than 3’ 10” water in reactor building

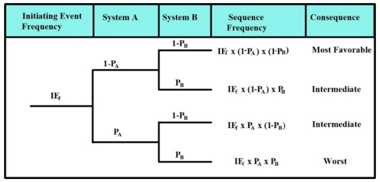




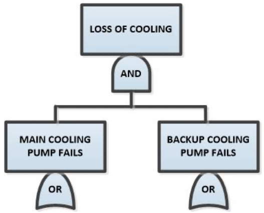
Source Containment Storage Tank



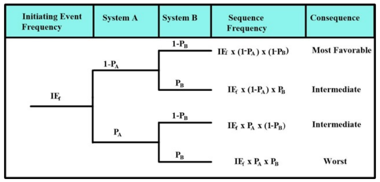
Summary of Mean Core Vulnerable State Frequency which lead to according to class

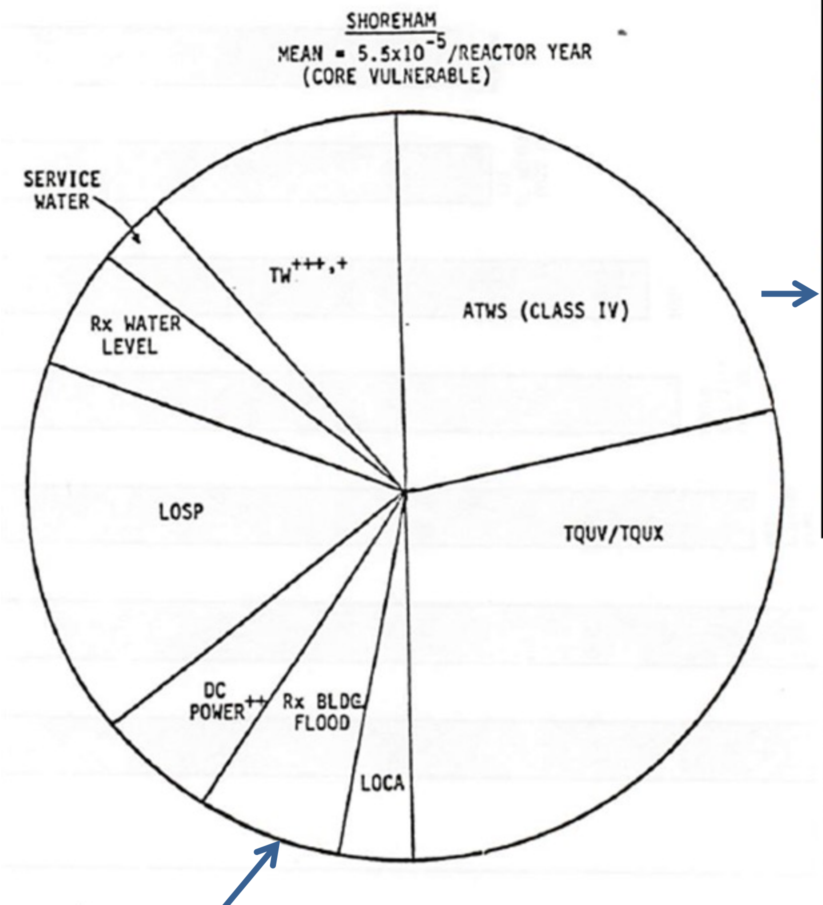


GENERALIZED CLASS	PHYSICAL BASIS FOR CLASSIFICATION	CLASS	ESTIMATED MEAN FREQUENCY OF CORE VULNERABILITY (PER REACTOR YEAR)
Loss of Coolant Makeup	Relatively fast core melt; containment intact at core melt and at low pressure	I	3.2E-5
Loss of Containment Heat Removal	Relatively slow core melt due to lower decay heat power: containment failed prior to core melt	II	8.5E-6
LOCA	Relatively fast core melt; containment intact at core melt, but at high internal pressure	III	1.0E-6
ATWS W/0 Poison Injection	Relatively fast core melt; containment fails prior to core melt due to overpressure	IV	1.4E-5
LOCA Outside Containment	Relatively fast core melt; containment failed from initiation of accident due to equipment failures	V	3.6E-8
TOTAL			5.5E-5



Summary of Mean Frequency which lead to core vulnerable state according to accident sequences





\+ LOSP separated out, ATWS Class I included.
++ Classes I and II.
+++ Anticipated transient and LOCAs only.

INITIATOR	PER YR	IMPORTANCE
ATWS	1.40E-05	25.50%
TRANSIENTS	1.30E-05	23.68%
LOSS OF OFFSITE POWER	1.10E-05	20.04%
SERVICE WATER	5.00E-06	9.11%
BLDG FLOOD	3.90E-06	7.10%
DC POWER BUS	2.90E-06	5.28%
RHR/REACTOR LEVEL	2.80E-06	5.10%
LOCA	2.30E-06	4.19%
TOTAL	5.49E-05	100.00%

T= Turbine Trip
Q=Feedwater
U=Reactor Core Isolation Cooling
X=Timely Depressurization
V=Core Spray
W=Residual Heat Removal System and Service Water