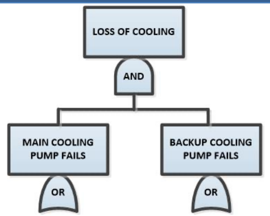
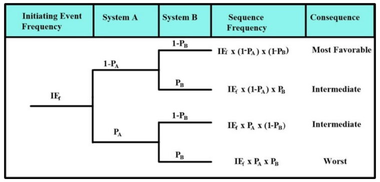
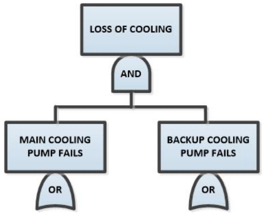


Reactor Safety Study
Session 3 Case Study

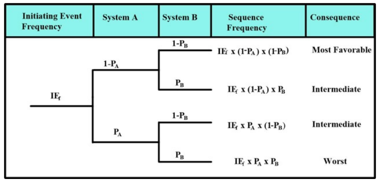


Describe the Reactor Safety Study a pioneering risk assessment study that used event trees and fault trees to generate reactor accident scenarios and compute the frequency and consequences of these scenarios

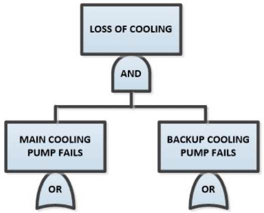
Howard Lambert
FTA Associates
2022



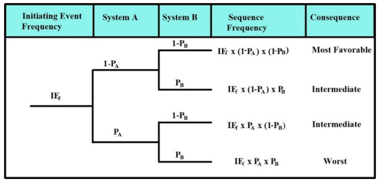
Reliability modeling techniques pertinent to nuclear power industry



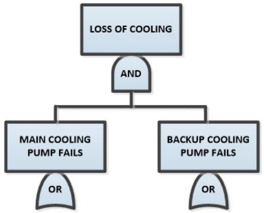
- Event-tree, fault-tree approach of Reactor Safety Study, WASH 1400
- Safeguards
  - Sabotage
  - Theft of Special Nuclear Material (SNM)



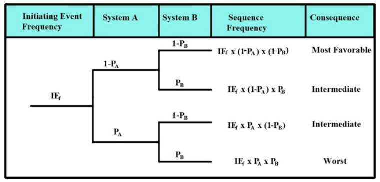
The Reactor Safety Study examined two reactor types



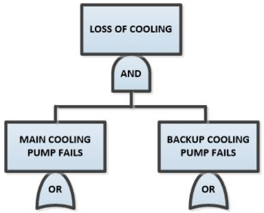
- Boiling water reactor (BWR)
- Pressurized water reactor (PWR)



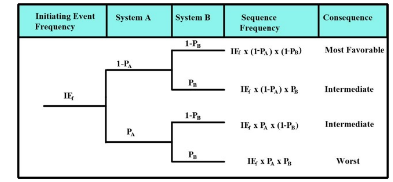
Two commercial types of nuclear power plants



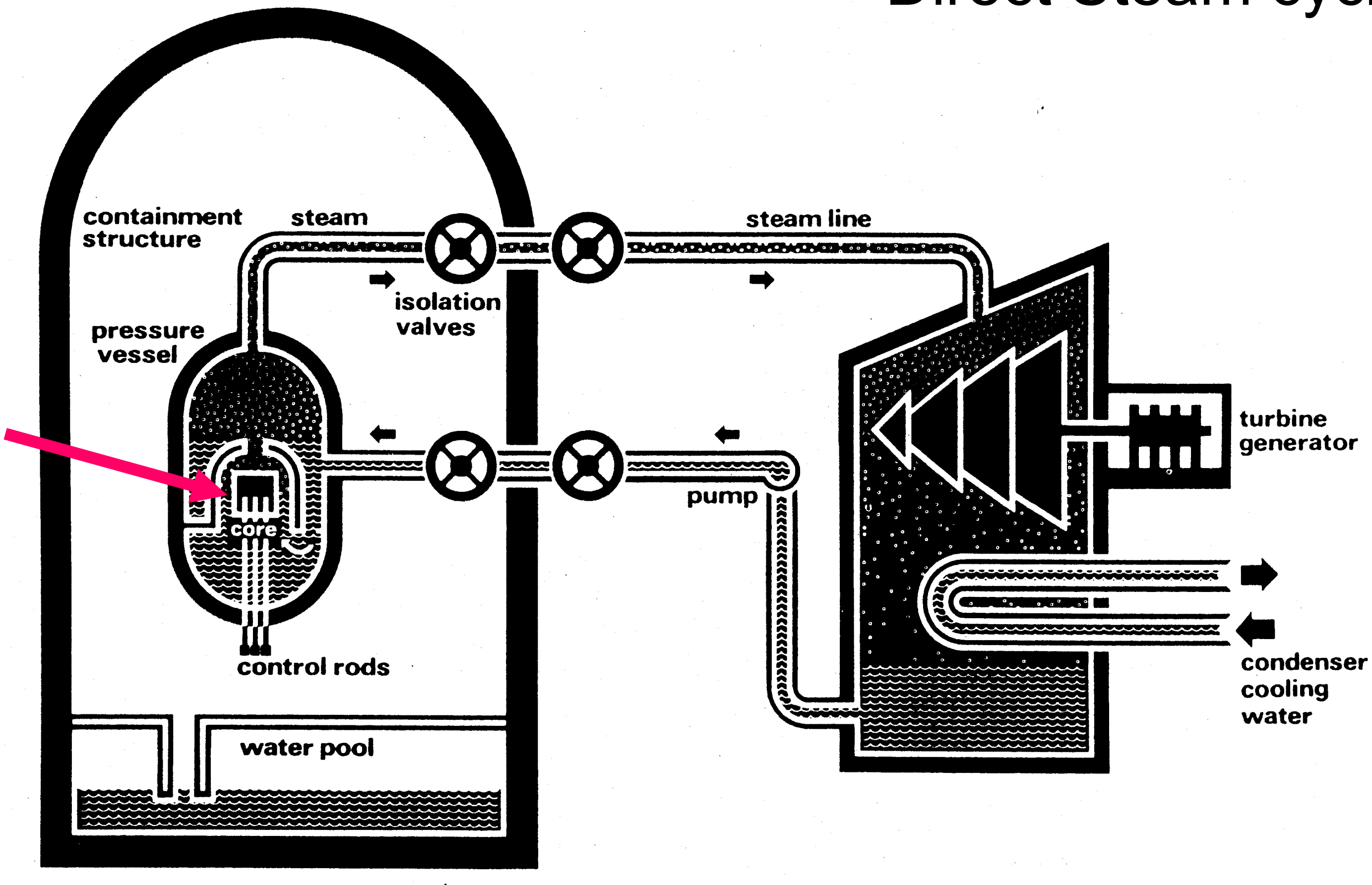
- The boiling water reactor (BWR) operates on direct cycle steam supplied to the turbine generator is produced in the reactor core. This system operates at 1000 psi and 545°F
- The pressurized water reactor (PWR) operates on an indirect cycle with a “liquid primary system” containing the reactor core which produces steam in the “secondary system” through the heat exchanger called the steam generator. The primary side operates at 2250 psi and 600ºF and the secondary system operates at 1000 psi and 550°F

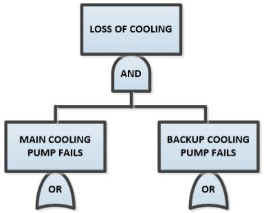


Boiling Water Reactor

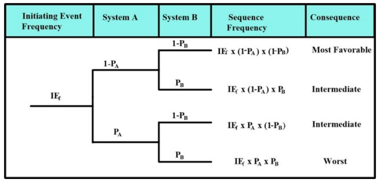


Direct Steam cycle

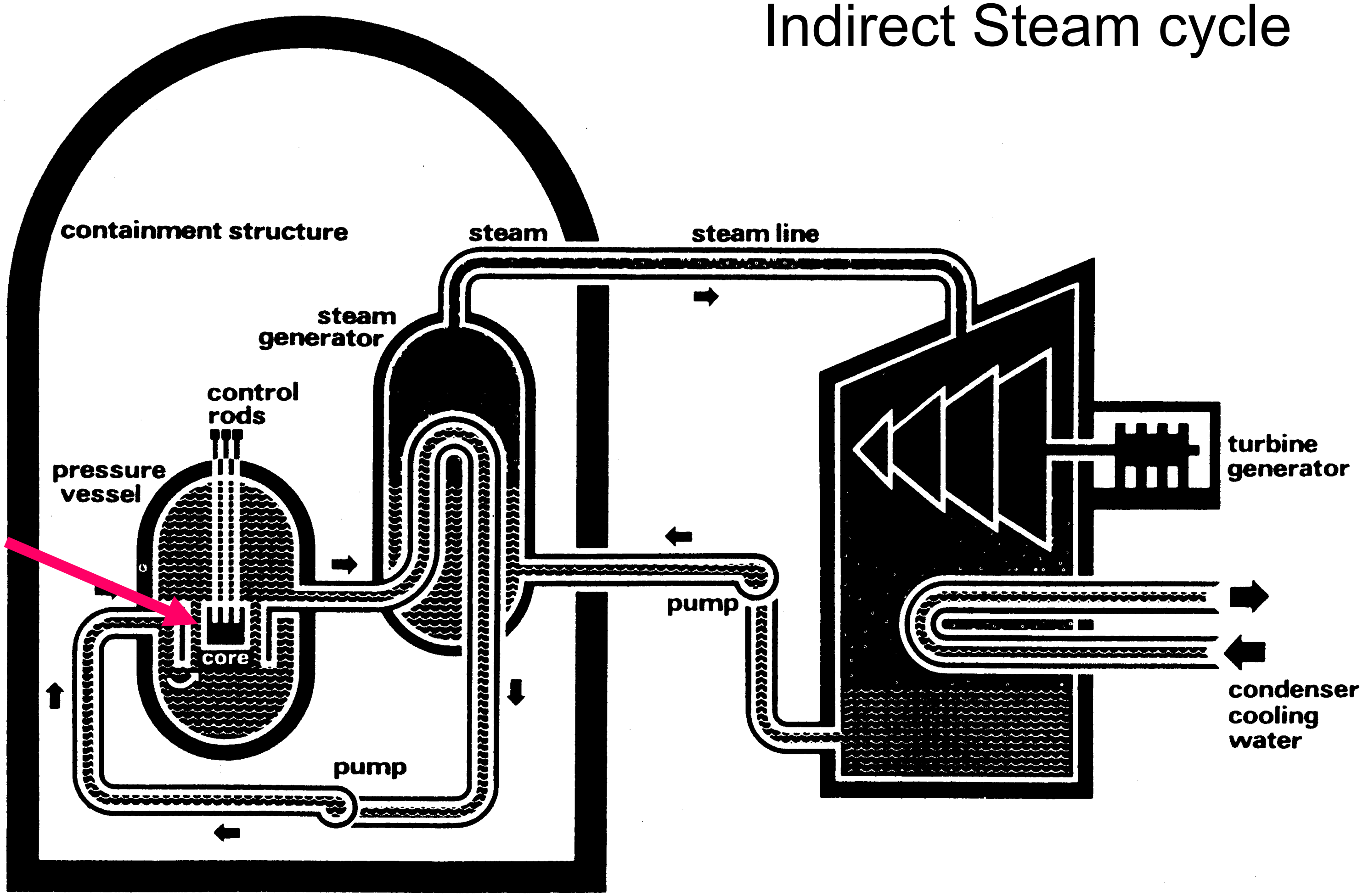


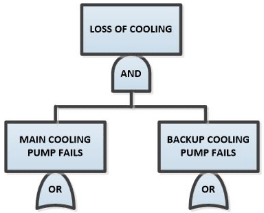


Pressurized Water Reactor

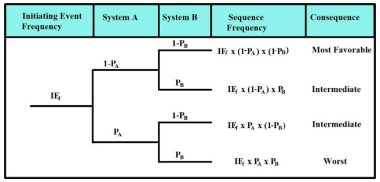


Indirect Steam cycle

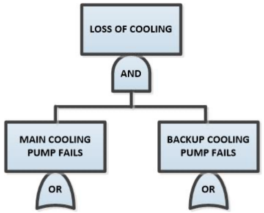




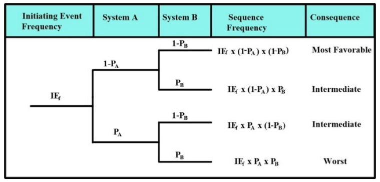
Purpose of Reactor Safety Study (RSS)



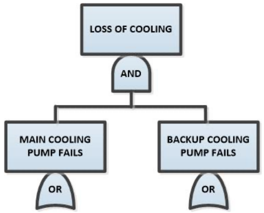
- Estimate risks to public from commercial nuclear power plants
- Compare these risks to other risks accepted by society
- WASH 1400 did not analyze entire nuclear fuel cycle



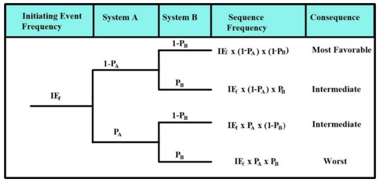
Public risks include



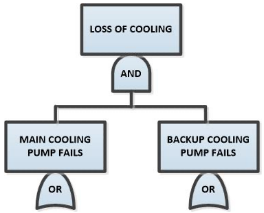
- Fatalities
- Illnesses
- Latent cancers
- Genetic effects
- Property damage



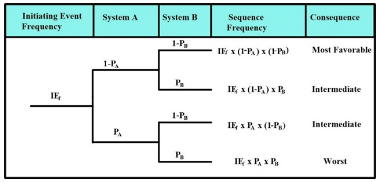
Location of radioactivity at a nuclear power plant



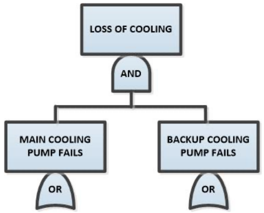
Location	Fraction of core inventory
Core (Most energetic)	1
Spent fuel storage (average)	1.6×10^{-1}
Shipping cask	2.7×10^{-3}
Waste gas storage tank	2.7×10^{-5}
Liquid waste storage tank	1.2×10^{-8}



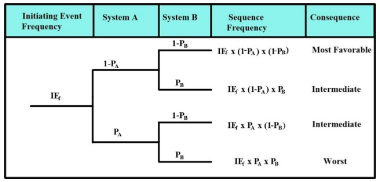
Fuel Melting Phenomena



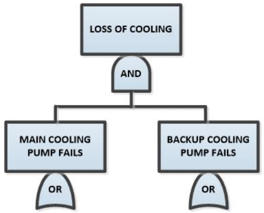
- Fuel melting (core damage) occurs as the result of imbalance between the heat generated by the fuel and the heat removed by the fuel
- Heat imbalances occur in two ways:
  1. Transient Events in which power level exceeds the capability to dissipate it
  2. Loss of coolant accident (LOCA) and emergency core cooling systems fail



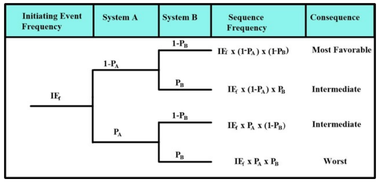
Engineered safeguard systems are safety systems used to mitigate the effects of transients and loss of coolant accidents, LOCAs



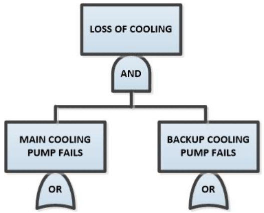
- The purpose of these systems are to:
  - Maintain the integrity of the cladding to prevent a fuel meltdown
  - Maintain the integrity of the containment to prevent release of fission products
  - Remove fission products, i.e., I^{131}, in event of a fuel meltdown



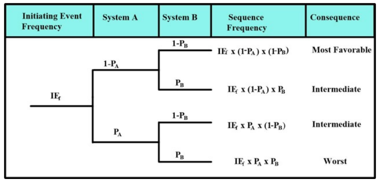
Engineered safeguard system functions



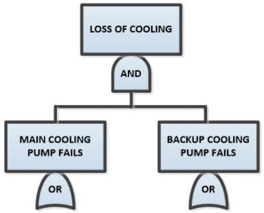
- Reactor trip—to terminate core power generation
- Emergency core cooling—to cool the core, keeping the release of radioactivity to the containment at low levels
- Post accident radioactivity removal—to remove radioactivity from the containment atmosphere
- Post accident heat removal—to remove heat from within the containment, thereby preventing over-pressurization
- Containment integrity—to prevent radioactivity from being dispersed into the environment



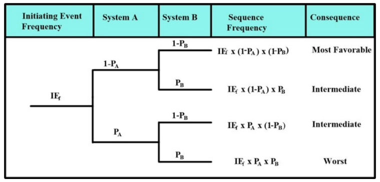
Event trees use in the Reactor Safety Study



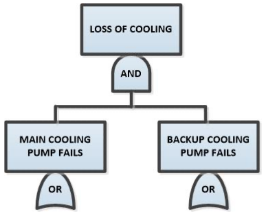
- Top level fault trees were abandoned to generate accident scenarios
- Used decision tree approach instead
- An event tree is an inductive logic diagram starts with a given initiating event and proceeds to define the possible outcomes of such and event considering the success and failure of engineered safeguards systems
- Applicable to internal events only – did not consider external events in a meaningful way



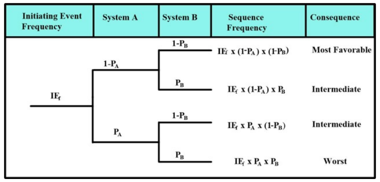
Event trees in WASH 1400 accomplished the following



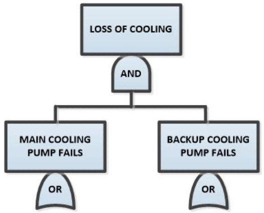
- Provided a basis in defining accident scenarios
- Depicting the relationships of success and failure of safety related systems associated with various accident consequences
- Provides a means for defining top events to system fault trees



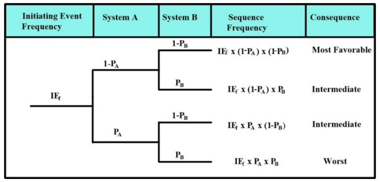
Initiating events leading to core melt

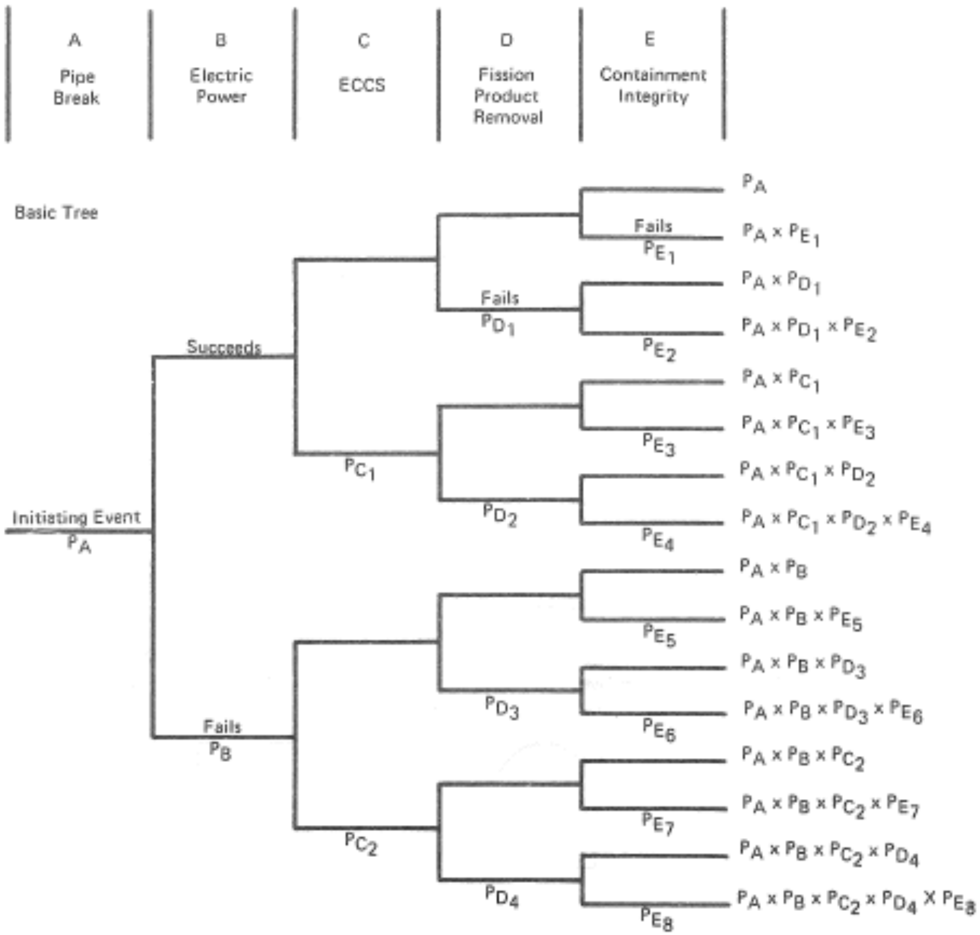


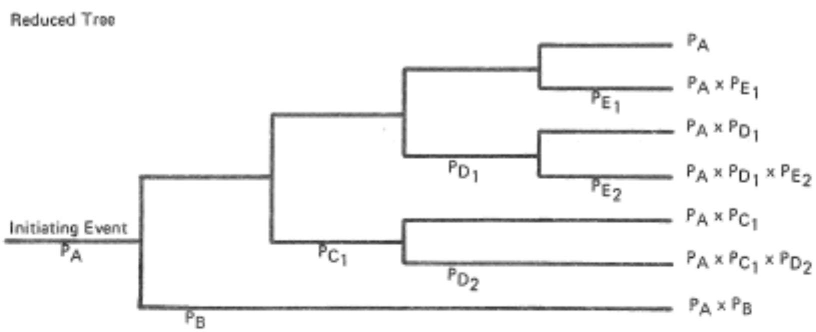
- Pipe breaks (three categories by size)
  - Large (6” to approximately 3’)
  - Intermediate (2” to 6”
  - Small (1/2” to 2”)
- Reactor pressure vessel ruptures (the more general term is an excessive LOCA)
- Steam generator ruptures
- Ruptures between systems interfacing with the reactor coolant system
- Transient events (e.g., loss of offsite power; turbine trip, etc.)



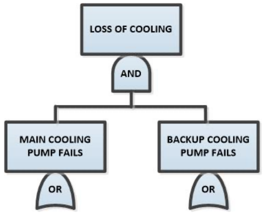
Simplified Event Trees for a Large LOCA



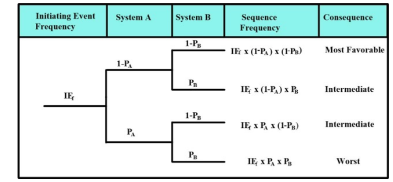


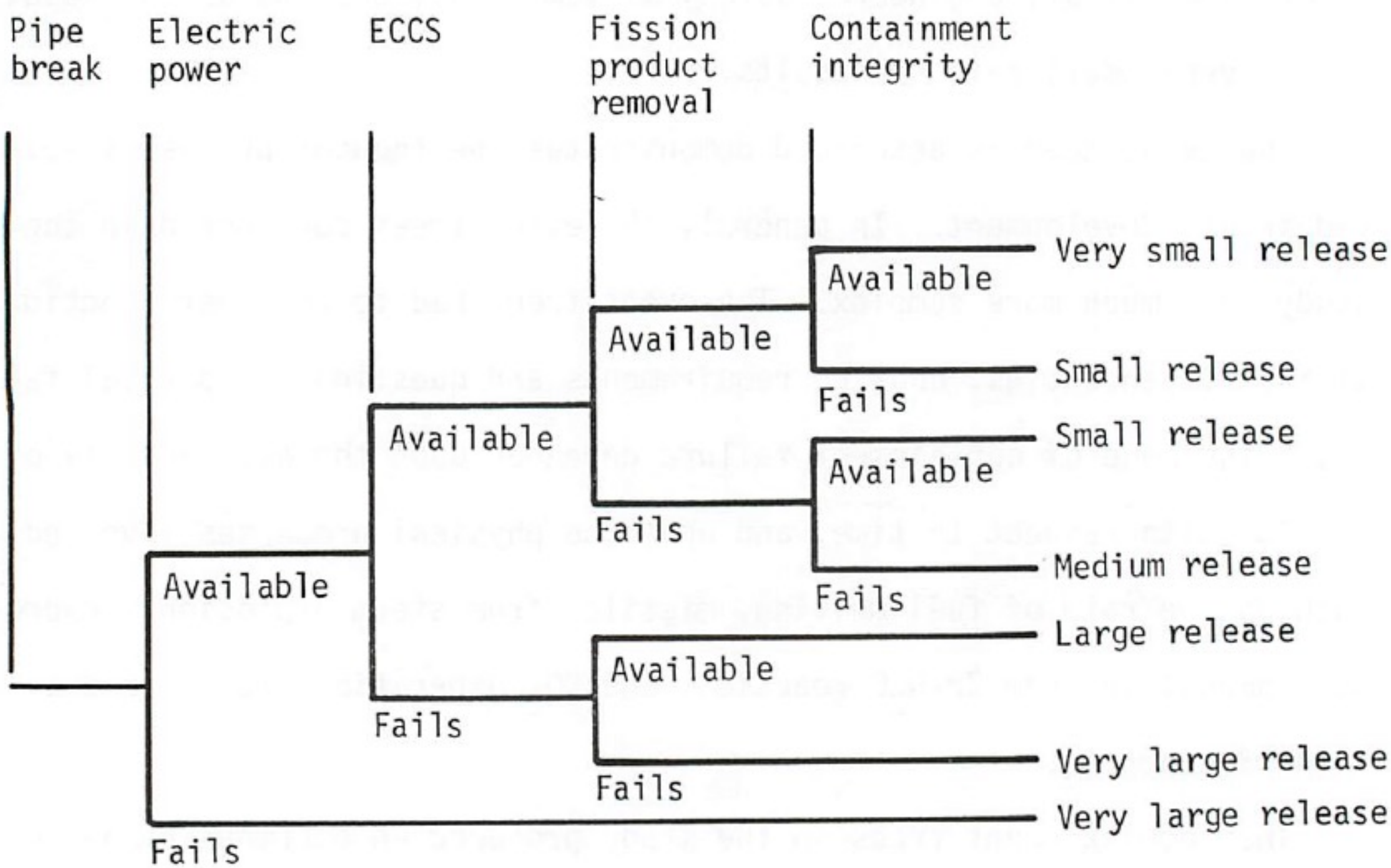


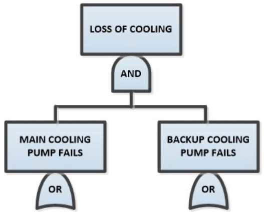
Note - Since the probability of failure, P, is generally less than 0.1, the probability of success (1-P) is always close to 1. Thus, the probability associated with the upper (success) branches in the tree is assumed to be 1.



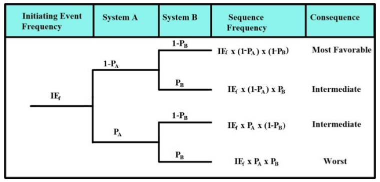
Simplified event tree for a local in a typical nuclear power plant







Event trees then generated accident sequences of the form



Accident sequence = Initiating event x System failure x Containment failure mode

AS A B C

Probability (AS) = P(A) x P (B|A) x P(C|A·B)

P(B|A) is determined by fault tree analysis (conditional probability of core damage given the initiating event)

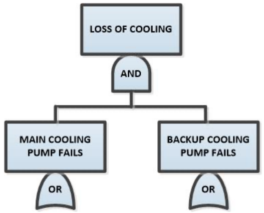
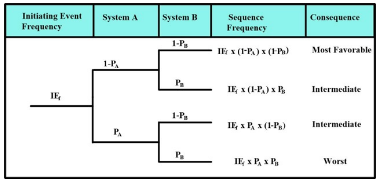
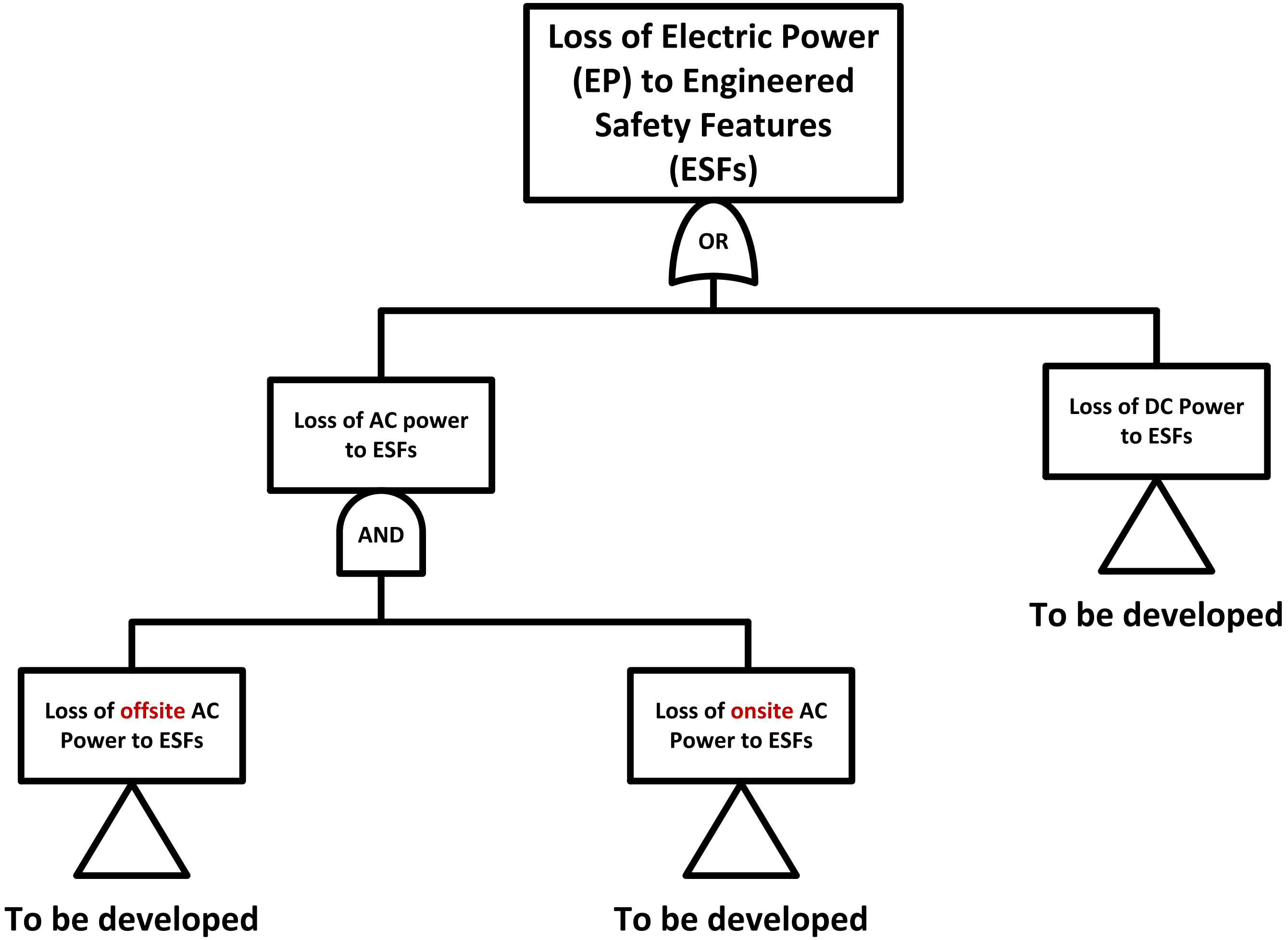
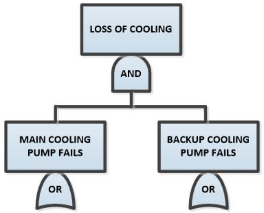


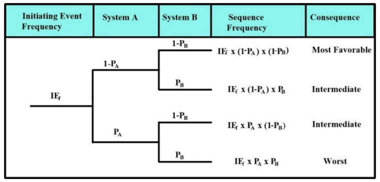
Illustration of fault tree development







The Reliability Quantification Techniques of Wash 1400 Centered Around Evaluation of these Sequences



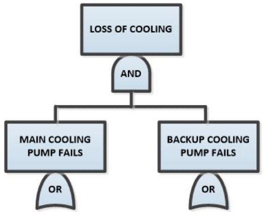
$$P(AS) = \underline{P(A)}\ \underline{P(B|A)}\ \underline{P(C|A \cdot B)}$$

Data On

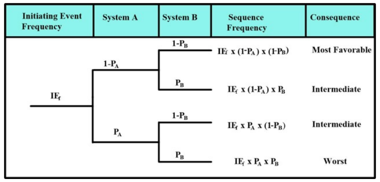
1) Pipe Breaks
2) Catastrophic Rupture of Pressure
3) Vessel
4) System Interface Conditions
5) Transient Events

Safety Systems (S) Unavailability, Quantification Of Fault Trees

Engineering Judgement On Accident Phenomenology



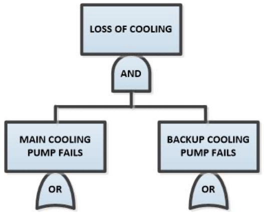
System Unavailability



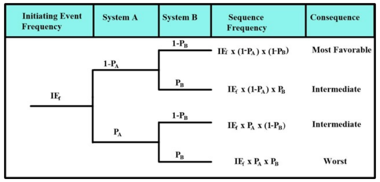
Factors That Could Cause These Safety Systems To Fail Upon Demand

1) Undetected Failures For Extended Periods of Time Caused by Human Error or Hardware Faults

2) System Downtime due to Testing or Maintenance

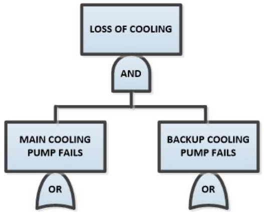


Factors that contribute to component unavailability

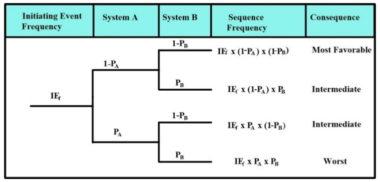


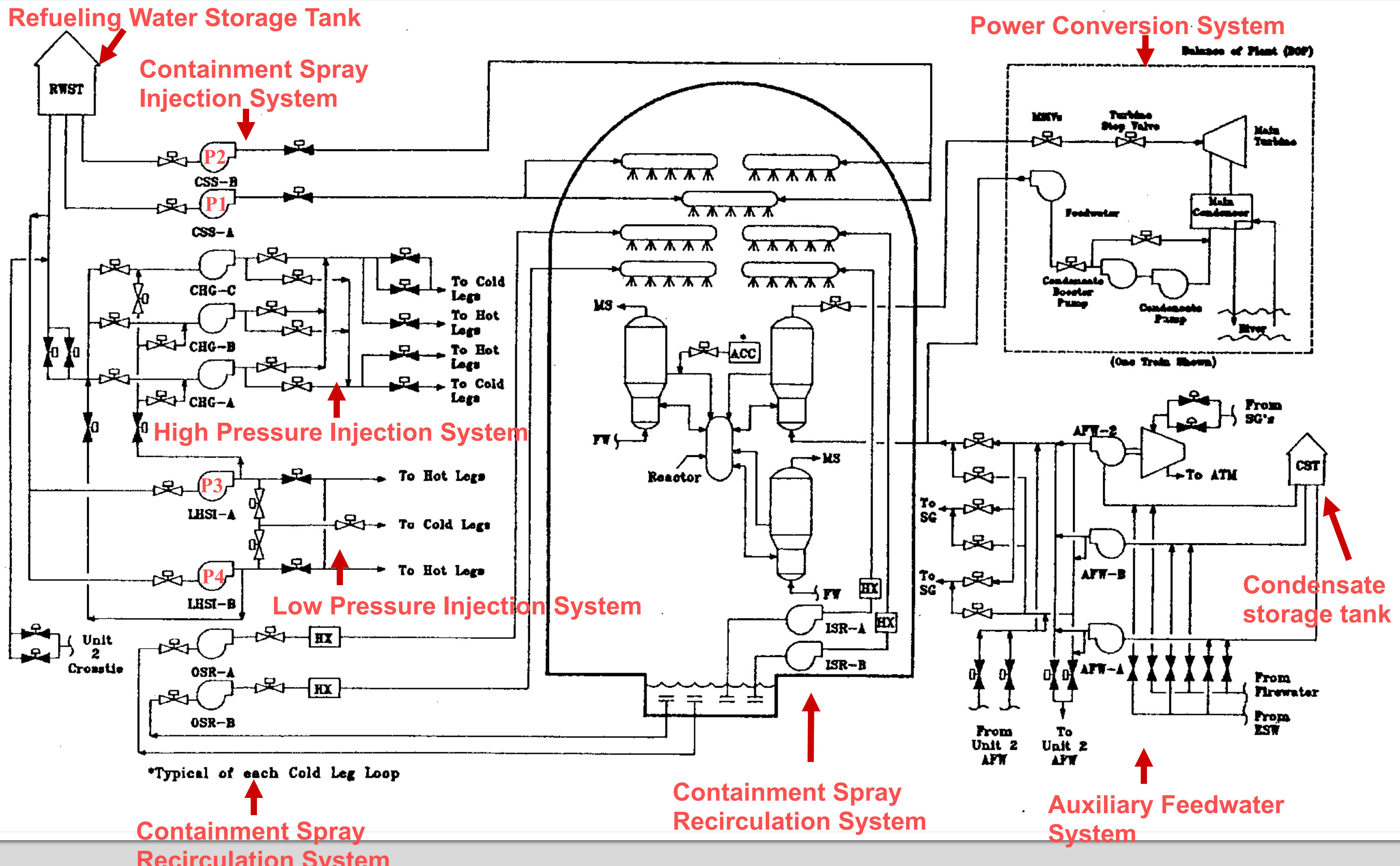
The Fault Trees in the Reactor Safety Study Revealed Four Major Factors that Contributed to the Downtime (i.e. Unavailability) of the Engineered Safeguard Systems --

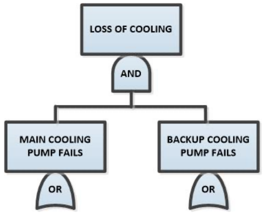
1. Random Hardware Failures
2. Periodic Testing
3. Maintenance
4. Human Error



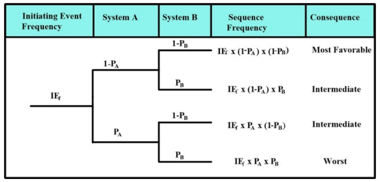
Surry, Unit One -- Nuclear Power Plant



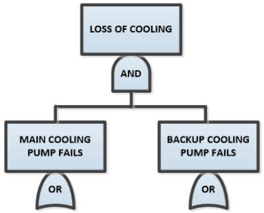




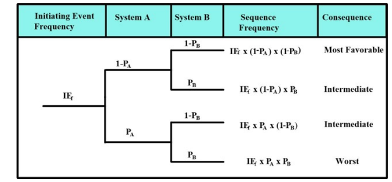
Containment Spray and Low Pressure Injection and Recirculation Systems

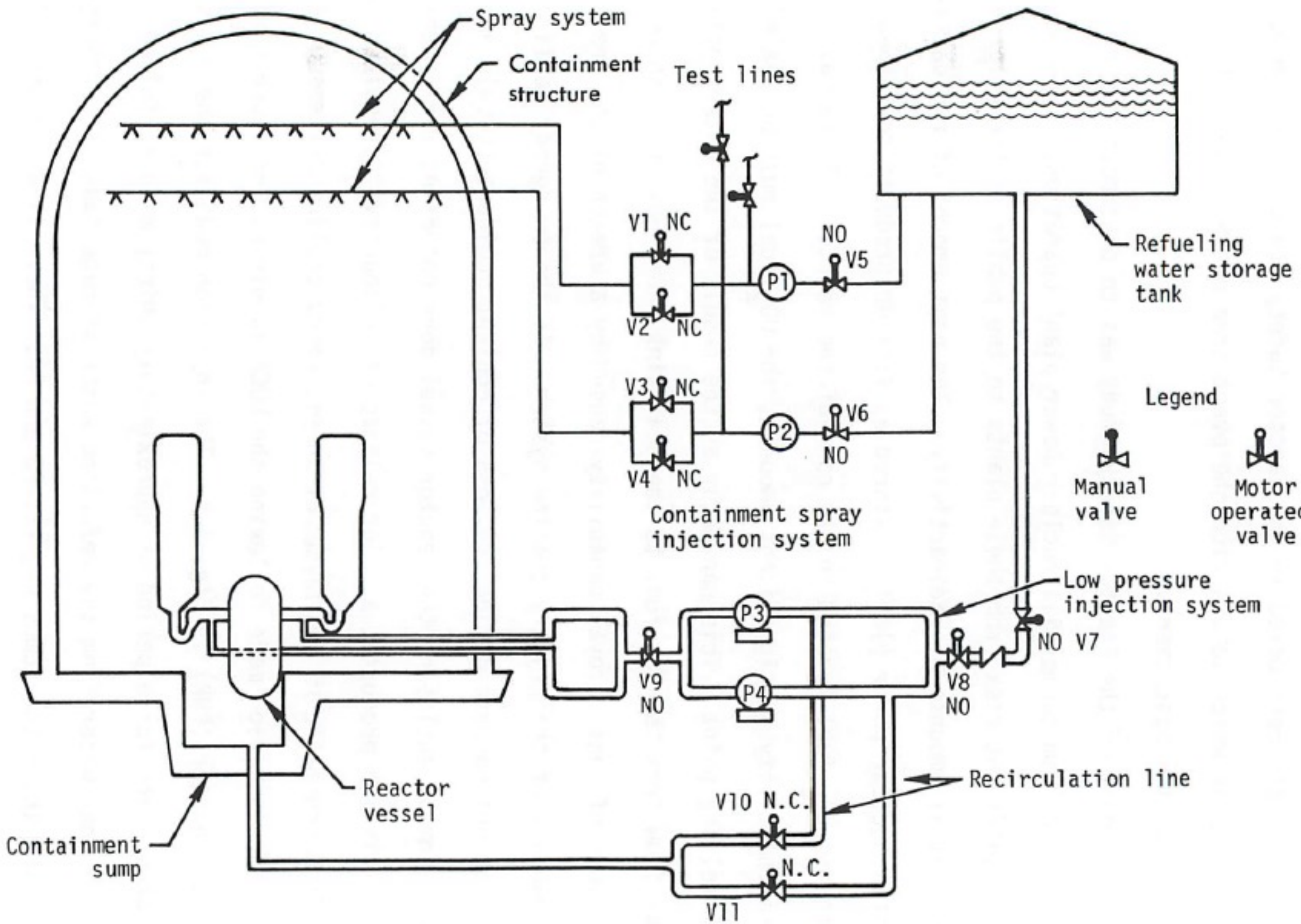


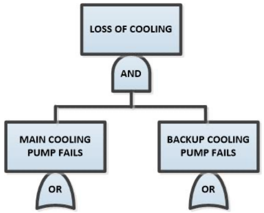
- To illustrate the calculations the Low Pressure Injection System (LPIS) and Containment Spray Injection System (CSIS) is chosen



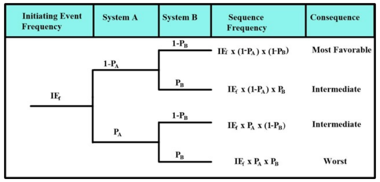
Containment Spray and Low Pressure Injection and Recirculation Systems





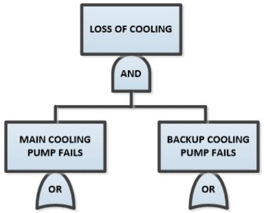


Hardware Contribution

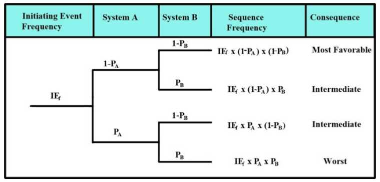


Point Estimates Based on Log Normal Distribution

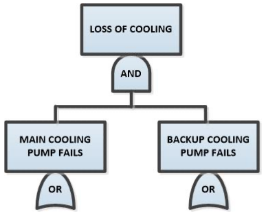
- Q Pump (Failure to Start) – **10^{-3}/Demand**
- Q Pump (Failure to Run, Give Start) – **3x 10^{-5} /HR.**
- Q Valve (Motor Operated, Failure to Open or Close) – **10^{-3} /Demand**
- Q Valve (Inadvertently Opens or Closes AT **t>0)** – **10^{-6} / Per Hour**



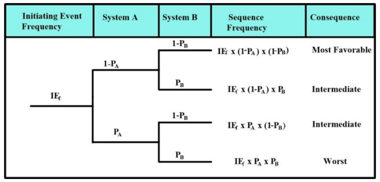
Maintenance Contribution



- Maintenance Performed on LPIS and CSIS Pumps Ranged From 1 Month to 12 Months, Log Normal Mean of 4.5 Months – Duration of Maintenance Act Between 30 Minutes and 24 Hours, LN Mean of 7.1 Hours.



Maintenance Contribution (continued)



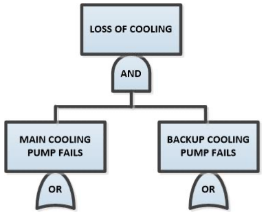
- M= F(ACTS Per Month X

  T (Hours Per Month)

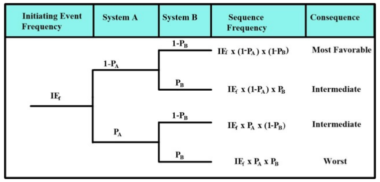
  720 (Hours Per Month)

- M = (1/4.5) (7.1)/(720)

  = 2.2 X 10^{-3} (Unavailability due to maintenance)



Testing Contribution



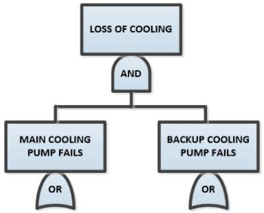
- Tech Specs. Require Testing of LPIS and CSIS Pumps at a Frequency of Once a Month – If Tests Last More Than 4 Hours, Plant Shutdown Required:

- LN Mean= 1.4 HR

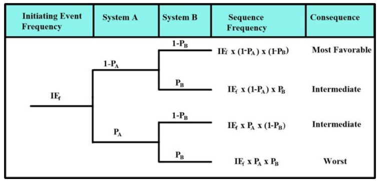
  $T = f \times t / 72$

  $= (1) \times (1.4) / 720$

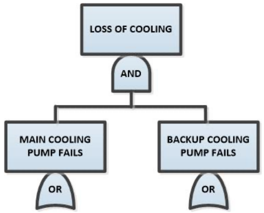
  $= 1.9 \times 10^{-3}$ (Unavailability due to Testing)



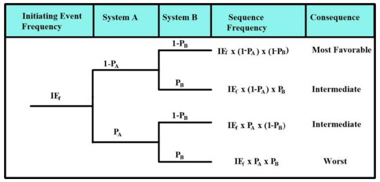
Human Error Contribution

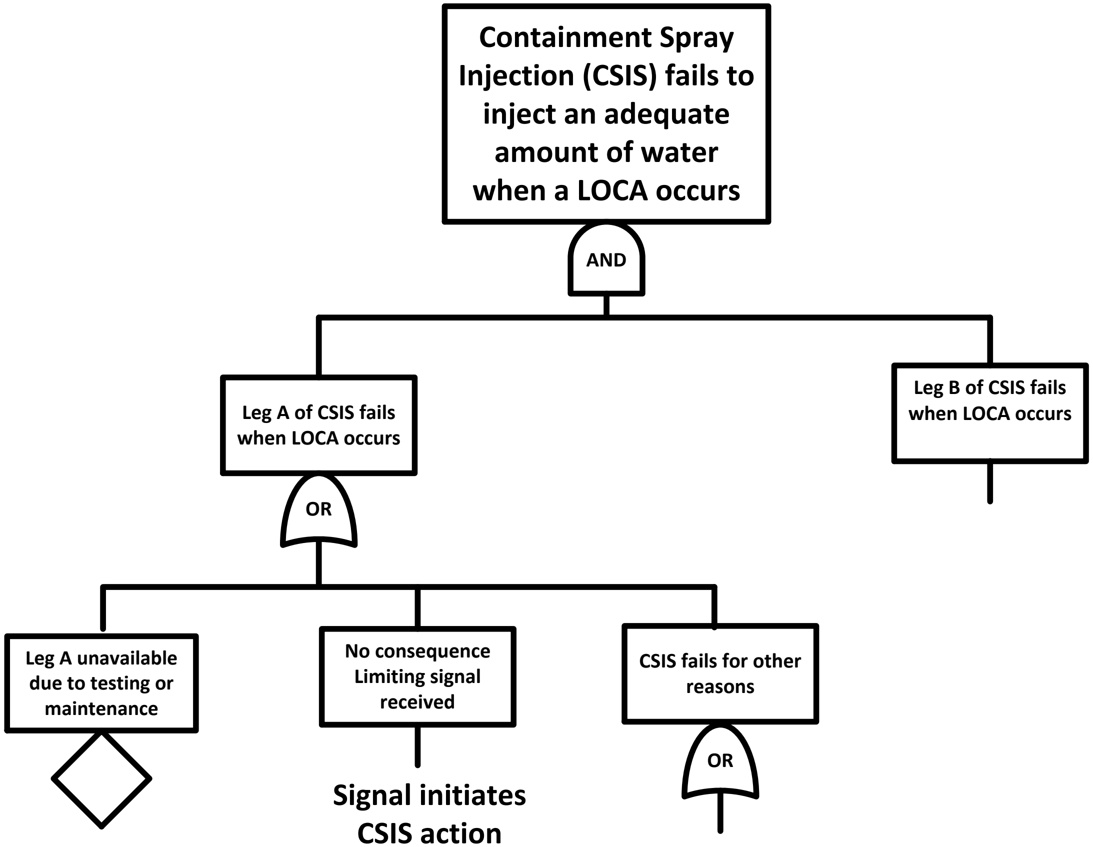


- Concept of Coupling Introduced In Quantifying Human Error– Four Levels of Coupling (i.e. Statistical Dependence)
- 1.) No Coupling e.g., P(A) x P(B)
- 2.) Loose Coupling SQRT[P(A) x P(B) x P(A)]
- 3.) Tight Coupling Min[P(A),P(B)]
- 4.) Complete Coupling e.g., P(A)



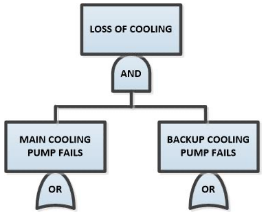
Containment Spray Injection Fault Tree



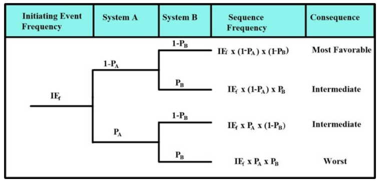


1. Hardware Related Faults e.g., pump 1 fails to start, MOV fails to open

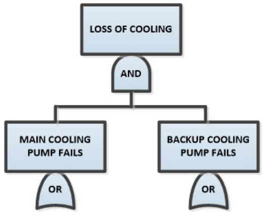
2. Human Error



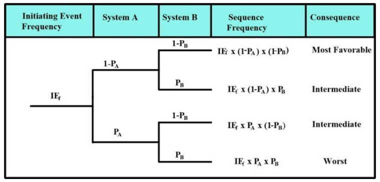
Min Cut Set Listing For CSIS



- 1. All Sensors in CLCS (Consequence Limiting Control System) miscalibrated
- 2. Both Manual Valves Left Open After Test
- 3.Leg A Down Due to Maintenance or testing <u>and</u> Leg B Fails when LOCA Occurs
- 4.Leg B Down Due to Maintenance or Testing <u>and</u> Leg A Fails When LOCA Occurs
- 5. Leg A Fails <u>and</u> Leg B Fails due to <u>Independent</u> Hardware Faults or Human Error



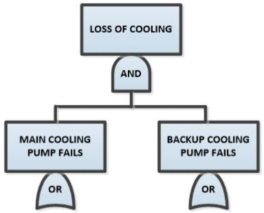
System Unavailability



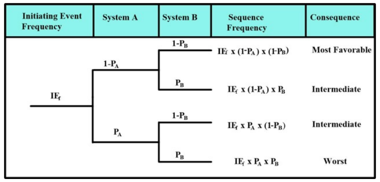
- $S_{LEGA} = Q_A + M_A + T_A$

<u>Contribution</u>

- $S = Q_A \cdot Q_B$ } Hardware
- $+Q_A (M_B + T_B) + Q_B (M_A + T_A)$ } Test and Maintenance
- $+Q_{CC}$ } Common Cause
- $+ Q_{SINGLES}$ } Singles



CSIS Unavailability Calculation

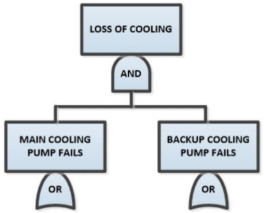


- $Q_{HARDWARE} = (1.8 \times 10^{-2})^{2} = 3.2 \times 10^{-4}$
- $Q_{TEST} + Q_{MAINTENANCE}$
- $2\,(1.9 \times 10^{-3} + 2.2 \times 10^{-3})\ (1.8 \times 10^{-2}) = 1.5 \times 10^{-4}$
- $Q_{COMMON\ CAUSE} = 1 \times 10^{-3} + 0.9 \times 10^{-3} = 1.9 \times 10^{-3}$

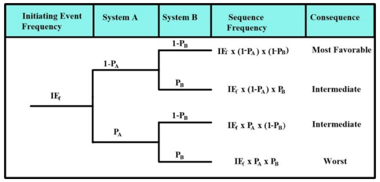
$$\sum = 1.9 \times 10^{-3}$$

Both Valves Open

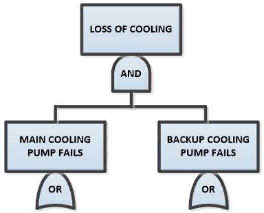
Miscalibration of Sensors



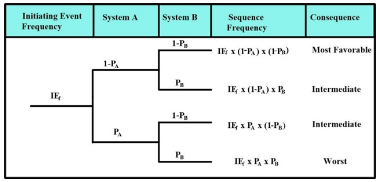
Contributions to System Unavailability for Various Engineered Safeguard Systems

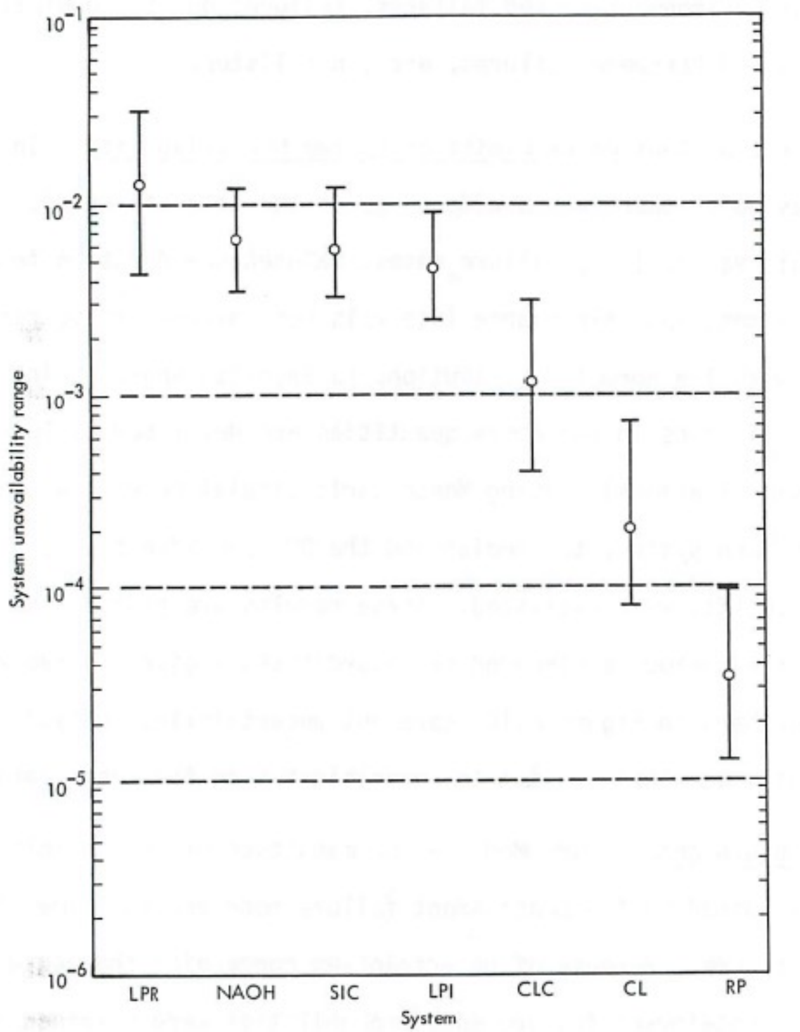


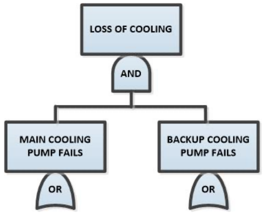
System	Hardware	Test & Maintenance	Human
Low Pressure Recirculation System (LPR)	14%		75%
Sodium Hydroxide System (NaOH)		75%	18%
Low Pressure Injection System (LPIS)	51%	20%	53%
Consequence Limiting Control System (CLCS)			91%
Containment Leakage (CL)	65%		
Reactor Protection (RP)	44%	33%	
Safety Injection Control System (SICS)	51%	38%	



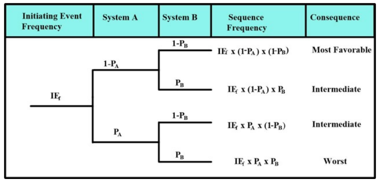
Characteristic System Results Unavailability Range

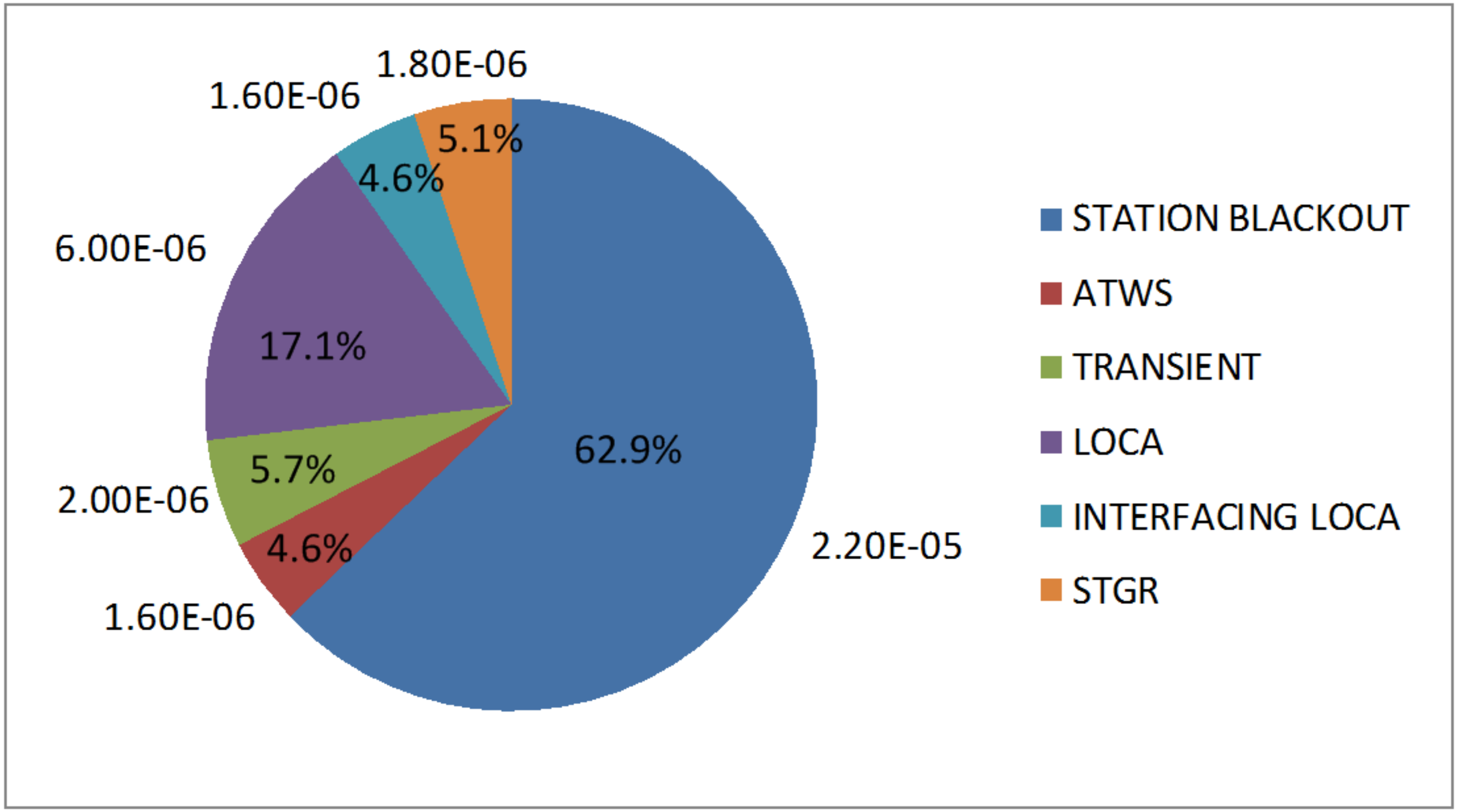




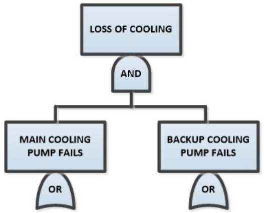


Contribution of initiating events to mean annual Core Melt Frequency – Surry NUREG 1150

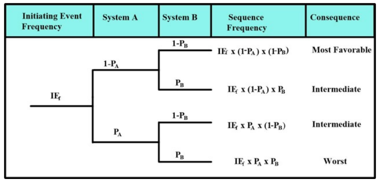


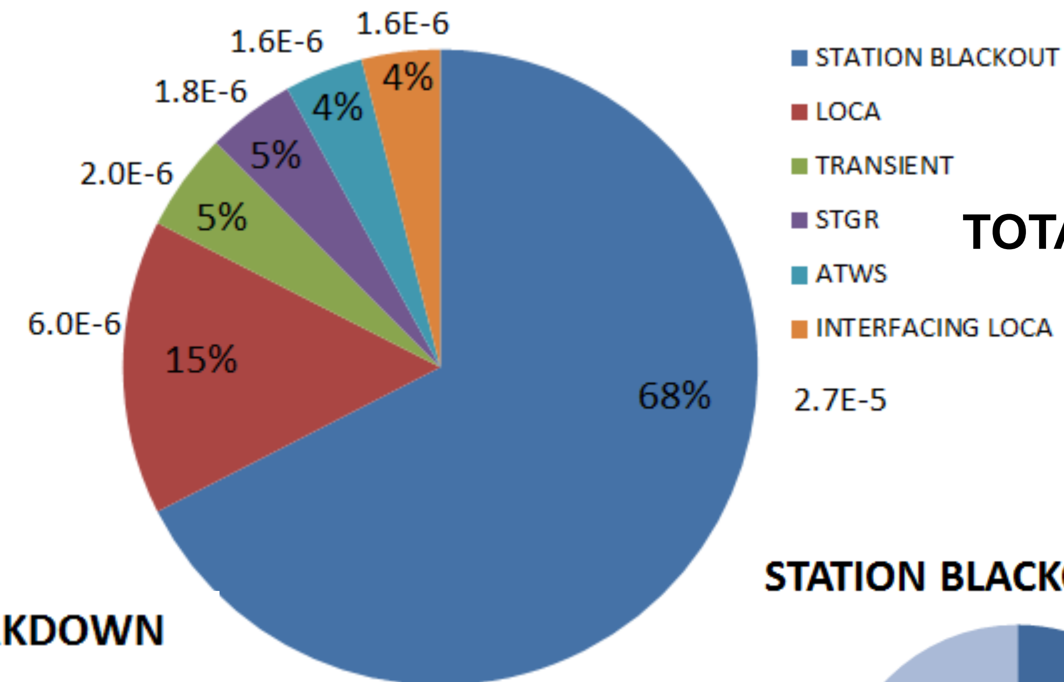


Total core melt frequency = 4.0 E-5 per year



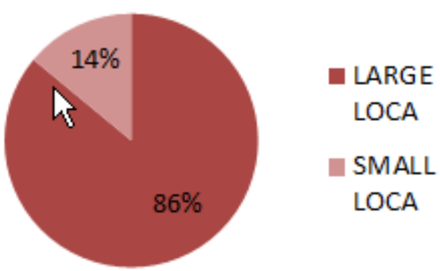
Contribution of initiating events to mean annual Core Melt Frequency – Surry NUREG 1150



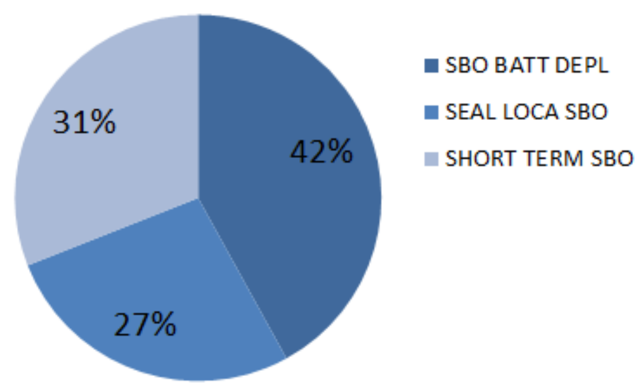


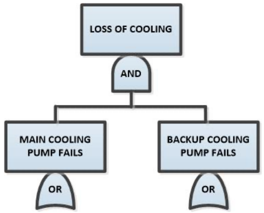
TOTAL = 4.0 E-5 YR-1

LOCA BREAKDOWN

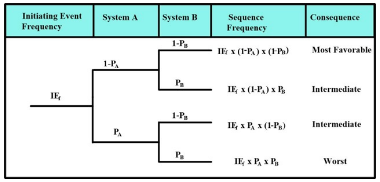


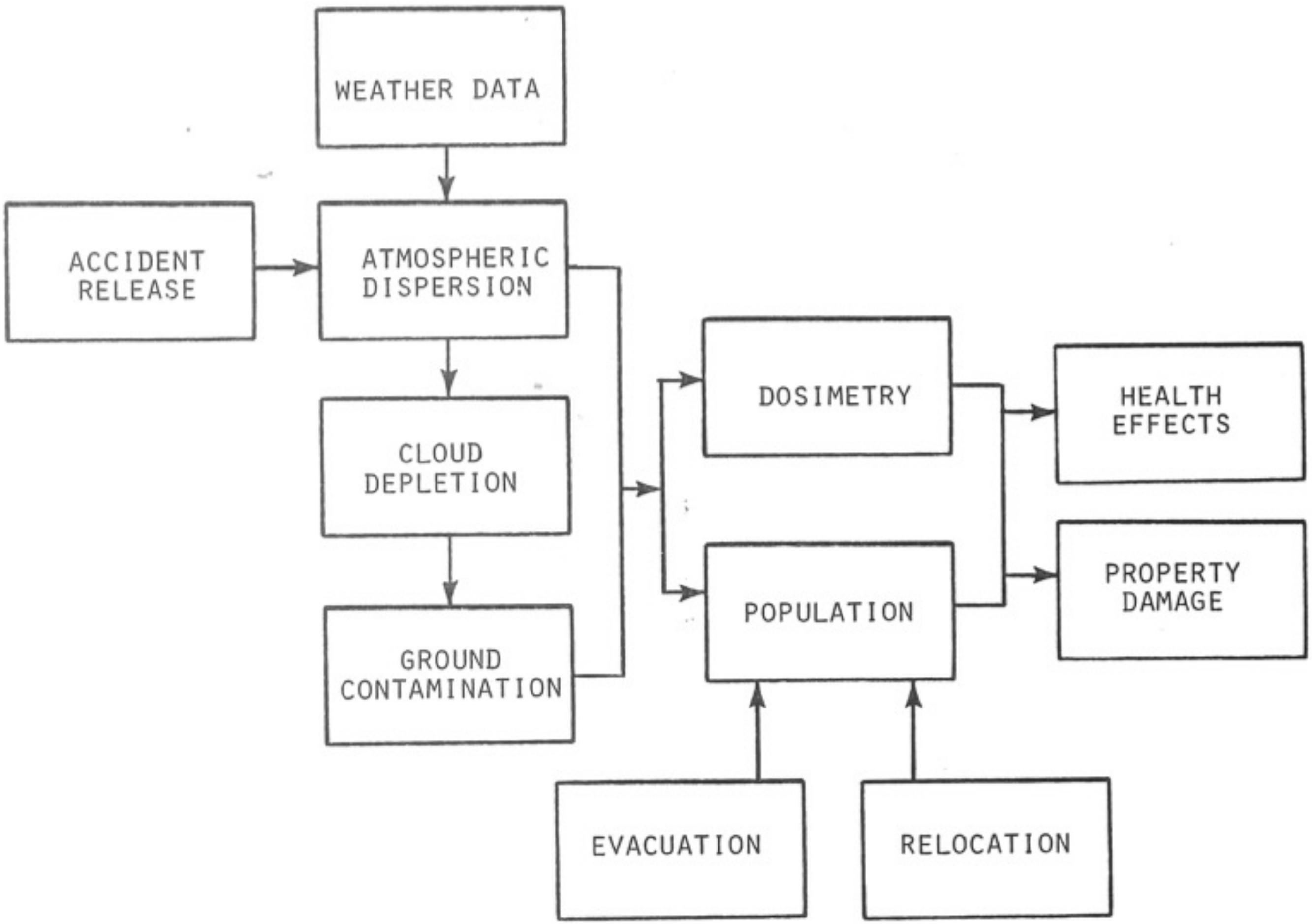
STATION BLACKOUT BREAKDOWN

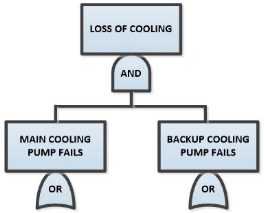




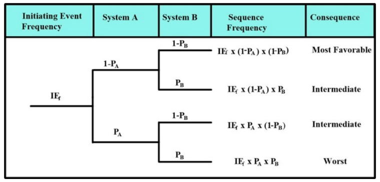
Consequence Model
Reactor Safety Study



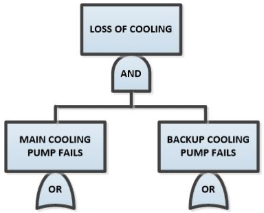




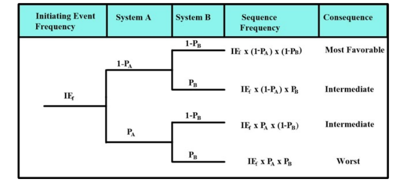
Magnitude of Release Following Core Damage



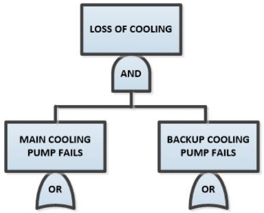
- There Must Be a Core Meltdown For A Major Release Of Radiation. Two Important Factors Contributed To The Magnitude Of Release After Core Damage
- 1. The Containment Failure Mode
- 2. The time at Which Containment Failure Occurs



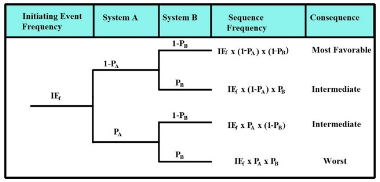
Containment Failure Modes (Fuel In Molten State) RSS



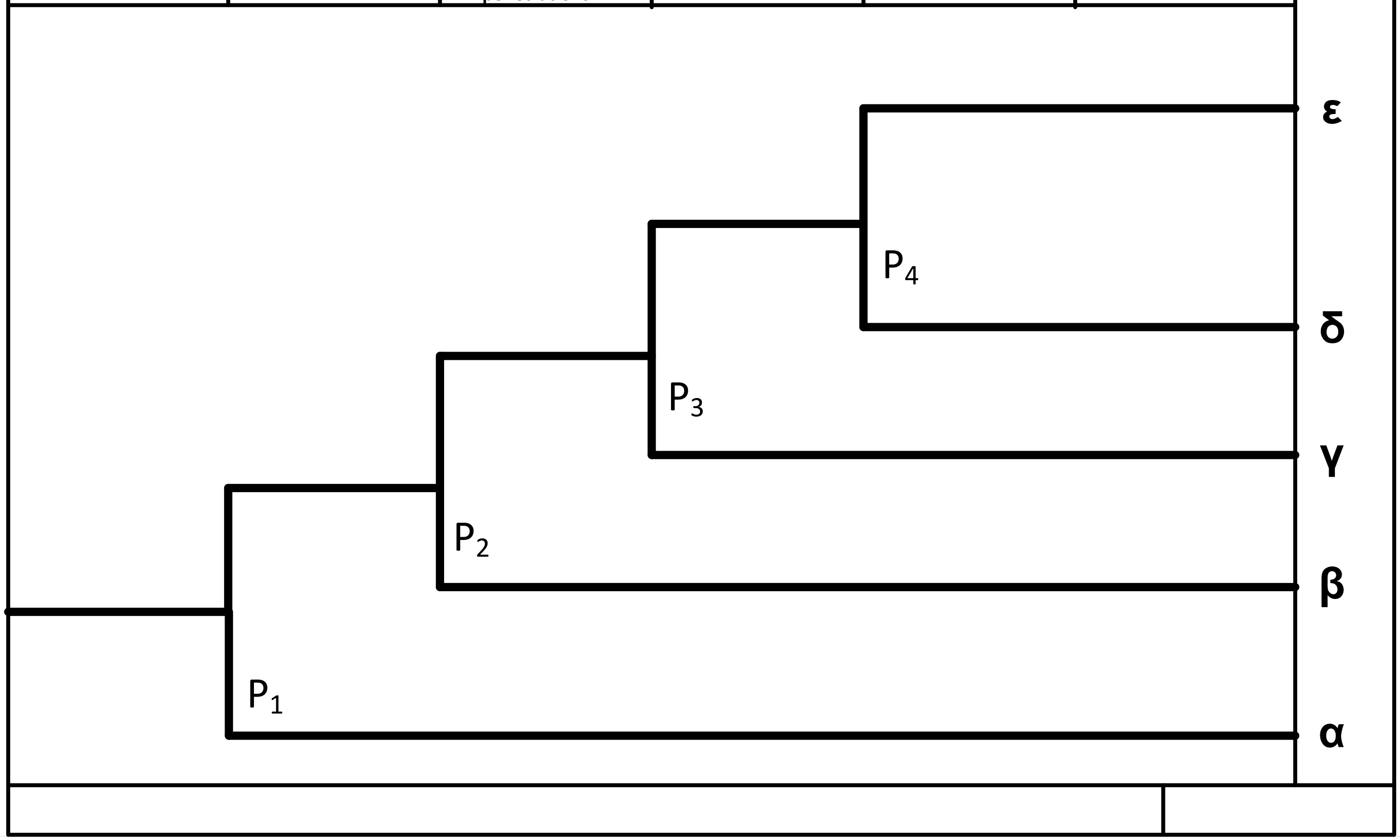
1. Containment Rupture Due to Steam Explosion α
2. Primary Failure Of Containment, I.E. Containment Fails To Isolate β
3. Containment Rupture Due to Hydrogen Combustion Υ
4. Containment Rupture Due to Over Pressurization δ
5. Containment Rupture By Melt Through ε



Containment Event Tree with end state probabilities RSS



PDS	α	β	γ	δ	ε	Seq Name
Plant Damage State	**Containment rupture due to a reactor vessel steam explosion**	**Containment failure resulting from inadequate isolation of containment openings and penetrations**	**Containment failure due to γ hydrogen burning**	**Containment failure due to overpressure**	**Containment vessel melt-through**	



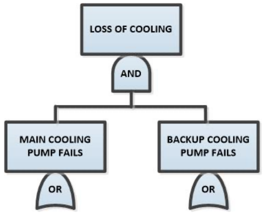
$P_\varepsilon = (1 - P_1)(1 - P_2)(1 - P_3)(1 - P_4)$

$P_\delta = (1 - P_1)(1 - P_2)(1 - P_3)P_4$

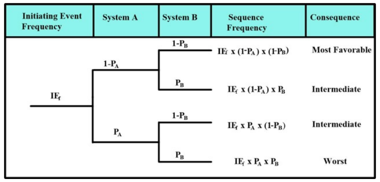
$P_\gamma = (1 - P_1)(1 - P_2)P_3$

$P_\beta = (1 - P_1)P_2$

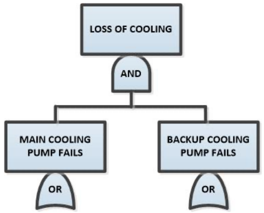
$P_\alpha = P_1$



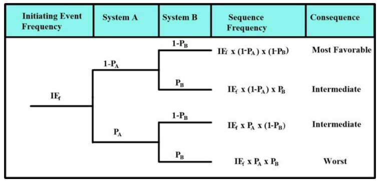
Consequence Modeling RSS



- 1. Key Sequences were Chosen For The Large LOCA
- 2. Battelle CORRAL Computer Code Determined The Isotopic Composition and Amount of Radionuclides Released For Various Accident Chains for These Key Sequences
- 3. Other Small LOCA, Transient and Other Event Tree Sequences were Grouped with These Key Sequences
- 4. Accident Sequence Were Then Grouped Into Representative Release Categories, 9 For PWR and 5 for BWR



Consequence Modeling RSS

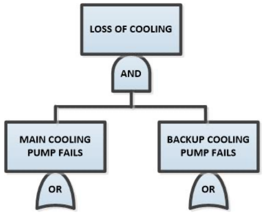


- 5. The Dose to the Population for Each Release Category was then Determined By Three Models

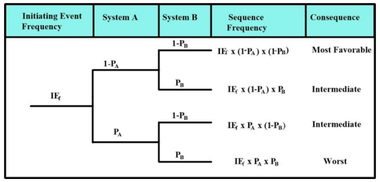
    1. Atmospheric Dispersion Model

    2. Population Model

    3. Health Effect and Property Damage Model



Consequence Modeling RSS



- 6. For Overall Risk Assessment, Histogram Were Generated, Probability of Release Versus Release Category by the Relation;

  P (Release Category) = P(Release) x P(Weather) x P (Population)

- 7. Consequences considered were

  1. Fatalities
  2. Injuries
  3. Long-Term Health Effects
  4. Property Damage

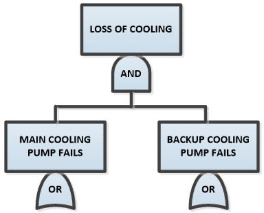
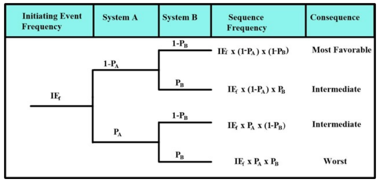


Table 5-1 Summary of Accidents Involving Core



RELEASE CATEGORY	PROBABILITY per Reactor-Yr	TIME OF RELEASE (Hr)	DURATION OF RELEASE (Hr)	WARNING TIME FOR EVACUATION (Hr)	ELEVATION OF RELEASE (Meters)	CONTAINMENT ENERGY RELEASE (10^6 Btu/Hr)	FRACTION OF CORE INVENTORY RELEASED[a] Xe-Kr	Org. I	I	Cs-Rb	Te-Sb	Ba-Sr	Ru[b]	La[c]
PWR 1	9×10^{-7}	2.5	0.5	1.0	25	520[d]	0.9	6×10^{-3}	0.7	0.4	0.4	0.05	0.4	3×10^{-3}
PWR 2	8×10^{-6}	2.5	0.5	1.0	0	170	0.9	7×10^{-3}	0.7	0.5	0.3	0.06	0.02	4×10^{-3}
PWR 3	4×10^{-6}	5.0	1.5	2.0	0	6	0.8	6×10^{-3}	0.2	0.2	0.3	0.02	0.03	3×10^{-3}
PWR 4	5×10^{-7}	2.0	3.0	2.0	0	1	0.6	2×10^{-3}	0.09	0.04	0.03	5×10^{-3}	3×10^{-3}	4×10^{-4}
PWR 5	7×10^{-7}	2.0	4.0	1.0	0	0.3	0.3	2×10^{-3}	0.03	9×10^{-3}	5×10^{-3}	1×10^{-3}	6×10^{-4}	7×10^{-5}
PWR 6	6×10^{-6}	12.0	10.0	1.0	0	N/A	0.3	2×10^{-3}	8×10^{-4}	8×10^{-4}	1×10^{-3}	9×10^{-5}	7×10^{-5}	1×10^{-5}
PWR 7	4×10^{-5}	10.0	10.0	1.0	0	N/A	6×10^{-3}	2×10^{-5}	2×10^{-5}	1×10^{-5}	2×10^{-5}	1×10^{-6}	1×10^{-6}	2×10^{-7}
PWR 8	4×10^{-5}	0.5	0.5	N/A	0	N/A	2×10^{-3}	5×10^{-6}	1×10^{-4}	5×10^{-4}	1×10^{-6}	1×10^{-8}	0	0
PWR 9	4×10^{-4}	0.5	0.5	N/A	0	N/A	3×10^{-6}	7×10^{-9}	1×10^{-7}	6×10^{-7}	1×10^{-9}	1×10^{-11}	0	0
BWR 1	1×10^{-6}	2.0	2.0	1.5	25	130	1.0	7×10^{-3}	0.40	0.40	0.70	0.05	0.5	5×10^{-3}
BWR 2	6×10^{-6}	30.0	3.0	2.0	0	30	1.0	7×10^{-3}	0.90	0.50	0.30	0.10	0.03	4×10^{-3}
BWR 3	2×10^{-5}	30.0	3.0	2.0	25	20	1.0	7×10^{-3}	0.10	0.10	0.30	0.01	0.02	3×10^{-3}
BWR 4	2×10^{-6}	5.0	2.0	2.0	25	N/A	0.6	7×10^{-4}	8×10^{-4}	5×10^{-3}	4×10^{-3}	6×10^{-4}	6×10^{-4}	1×10^{-4}
BWR 5	1×10^{-4}	3.5	5.0	N/A	150	N/A	5×10^{-4}	2×10^{-9}	6×10^{-11}	4×10^{-9}	8×10^{-12}	8×10^{-14}	0	0

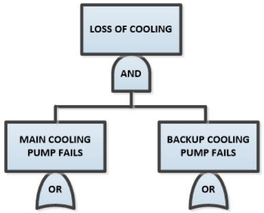
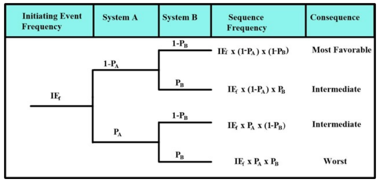


Table 5-2 PWR Dominant Accident Sequences **vs.** Release Categories



RELEASE CATEGORIES							Core Melt	No Core Melt	
	1	2	3	4	5	6	7	8	9
LARGE LOCA A	AB-α 1×10^{-11} AF-α 1×10^{-10} ACD-α 5×10^{-11} AG-α 9×10^{-11}	AB-γ 1×10^{-10} AB-δ 4×10^{-11} AHF-γ 2×10^{-11}	AD-α 2×10^{-8} AH-α 1×10^{-8} AF-δ 1×10^{-8} AG-δ 9×10^{-9}	ACD-β 1×10^{-11}	AD-β 4×10^{-9} AH-β 3×10^{-9}	AB-ι 1×10^{-9} AHF-ε 1×10^{-10} ADF-ε 2×10^{-10}	AD-ε 2×10^{-6} AH-ε 1×10^{-6}	A-β 2×10^{-7}	A 1×10^{-4}
A Probabilities	2×10^{-9}	1×10^{-8}	1×10^{-7}	1×10^{-8}	4×10^{-8}	3×10^{-7}	3×10^{-6}	1×10^{-5}	1×10^{-4}
SMALL LOCA S_1	S_1B-α 3×10^{-11} S_1CD-α 7×10^{-11} S_1F-α 3×10^{-10} S_1G-α 3×10^{-10}	S_1B-γ 4×10^{-10} S_1B-δ 1×10^{-10} S_1HF-γ 6×10^{-11}	S_1D-α 3×10^{-8} S_1H-α 3×10^{-8} S_1F-δ 3×10^{-8} S_1G-δ 3×10^{-8}	S_1CD-β 1×10^{-11}	S_1H-β 5×10^{-9} S_1D-β 6×10^{-9}	S_1DF-ε 3×10^{-10} S_1B-ε 2×10^{-9} S_1HF-ε 4×10^{-10}	S_1D-ε 3×10^{-6} S_1H-ε 3×10^{-6}	S_1-β 6×10^{-7}	S_1 3×10^{-4}
S_1 Probabilities	3×10^{-9}	2×10^{-8}	2×10^{-7}	3×10^{-8}	8×10^{-8}	6×10^{-7}	6×10^{-6}	3×10^{-5}	3×10^{-4}
SMALL LOCA S_2	S_2B-α 1×10^{-10} S_2F-α 1×10^{-9} S_2CD-α 2×10^{-10} S_2G-α 9×10^{-10} S_2C-α 2×10^{-8}	S_2B-γ 1×10^{-9} S_2HF-γ 2×10^{-10} S_2B-δ 4×10^{-10}	S_2D-α 9×10^{-8} S_2H-α 6×10^{-8} S_2F-δ 1×10^{-7} S_2C-δ 2×10^{-6} S_2G-δ 9×10^{-8}	S_2DG-β 1×10^{-12}	S_2D-β 2×10^{-8} S_2H-β 1×10^{-8}	S_2B-ε 8×10^{-9} S_2CD-ε 2×10^{-8} S_2HF-ε 1×10^{-9}	S_2D-ε 9×10^{-6} S_2H-ε 6×10^{-6}		
S_2 Probabilities	1×10^{-7}	3×10^{-7}	3×10^{-6}	3×10^{-7}	3×10^{-7}	2×10^{-6}	2×10^{-5}		

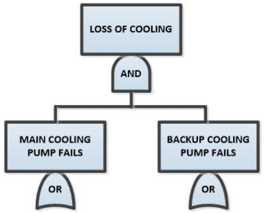
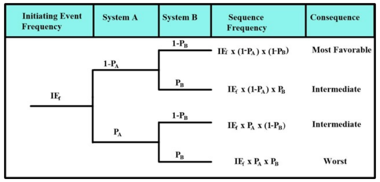
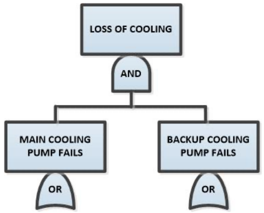


Table 5-2 PWR Dominant Accident Sequences **vs.** Release Categories continued

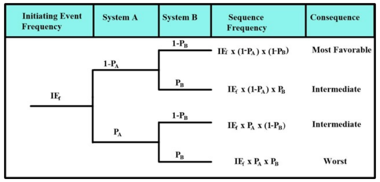


Release Category	1	2	3	4	5	6	7	8	9
REACTOR VESSEL RUPTURE - R	RC-α 2×10^{-12}	RC-γ 3×10^{-11} RF-δ 1×10^{-11} RC-δ 1×10^{-12}	R-α 1×10^{-9}				R-ε 1×10^{-7}		
R Probabilities	2×10^{-11}	1×10^{-10}	1×10^{-9}	2×10^{-10}	1×10^{-9}	1×10^{-8}	1×10^{-7}		
INTERFACING SYSTEMS LOCA (CHECK VALVE) - V		V 4×10^{-6}							
V Probabilities	4×10^{-7}	4×10^{-6}	4×10^{-7}	4×10^{-8}					
TRANSIENT EVENT - T	TMLB'-α 3×10^{-8}	TMLB'-γ 7×10^{-7} TMLB'-δ 2×10^{-6}	TML-α 6×10^{-8} TKQ-α 3×10^{-8} TKMQ-α 1×10^{-8}		TML-β 3×10^{-10} TKQ-β 3×10^{-10}	TMLB'-ε 6×10^{-7}	TML-ε 6×10^{-6} TKQ-ε 3×10^{-6} TKMQ-ε 1×10^{-6}		
T Probabilities	3×10^{-7}	3×10^{-6}	4×10^{-7}	7×10^{-8}	2×10^{-7}	2×10^{-6}	1×10^{-5}		
(Σ) SUMMATION OF ALL ACCIDENT SEQUENCES PER RELEASE CATEGORY									
MEDIAN (50% VALUE)	9×10^{-7}	8×10^{-6}	4×10^{-6}	5×10^{-7}	7×10^{-7}	6×10^{-6}	4×10^{-5}	4×10^{-5}	4×10^{-4}
LOWER BOUND (5% VALUE)	9×10^{-8}	8×10^{-7}	6×10^{-7}	9×10^{-8}	2×10^{-7}	2×10^{-6}	1×10^{-5}	4×10^{-6}	4×10^{-5}
UPPER BOUND (95% VALUE)	9×10^{-6}	8×10^{-5}	4×10^{-5}	5×10^{-6}	4×10^{-6}	2×10^{-5}	2×10^{-4}	4×10^{-4}	4×10^{-3}

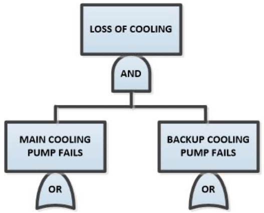
Note: The probabilities for each release category for each event tree and the Σ for all accident sequences are the median values of the dominant accident sequences summed by Monte Carlo simulation plus a 10% contribution from the adjacen release category probability.



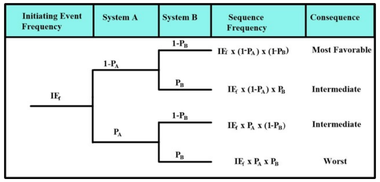
Key to PWR Accident Sequence Symbols



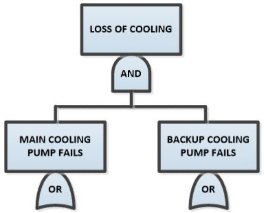
- A – Intermediate to Large LOCA.
- B – Failure of Electric Power to ESFs.
- B' – Failure to recover either onsite or offsite electric power within about 1 to 3 hours following an initiating transient which is a loss of offsite AC power.
- C – Failure of the containment spray injection system.
- D - Failure of the emergency core cooling injection system.
- F – Failure of the containment heat removal system.



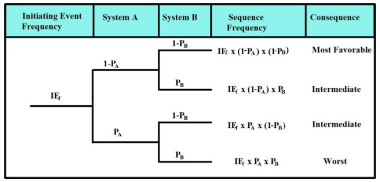
Key to PWR Accident Sequence Symbols Continued



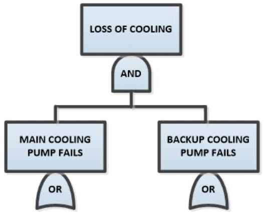
- G – Failure of the containment heat removal system.
- H – Failure of the emergency corer cooling recirculation system.
- K – Failure of the reactor protection system.
- L – Failure of the secondary system steam relief valves and the auxiliary feedwater system.
- M – Failure of the secondary system steam relief valves and the power conversion system.



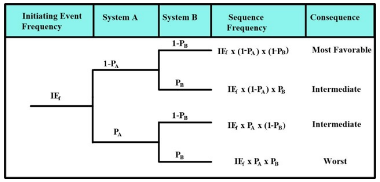
Key to PWR Accident Sequence Symbols Continued



- Q – Failure of the primary system safety relief valves to reclose after opening.
- R – Massive rupture of the reactor vessel (also called excessive LOCA).
- S_1 – A small LOCA with an equivalent diameter of about 2 to 6 inches.
- S_2 – A small LOCA with an equivalent diameter of about ½ to 2 inches.
- T – Transient event.



Key to PWR Accident Sequence Symbols Continued



- V – LPIS check valve failure.
- α - Containment rupture due to a reactor vessel steam explosion.
- β - Containment failure resulting from inadequate isolation of containment openings and penetrations.
- ϒ - Containment failure due to hydrogen burning.
- δ - Containment failure due to overpressure.
- ε - Containment vessel melt-through.

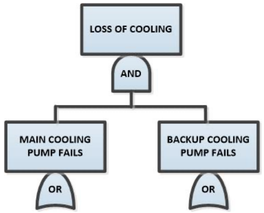
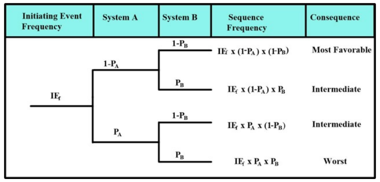
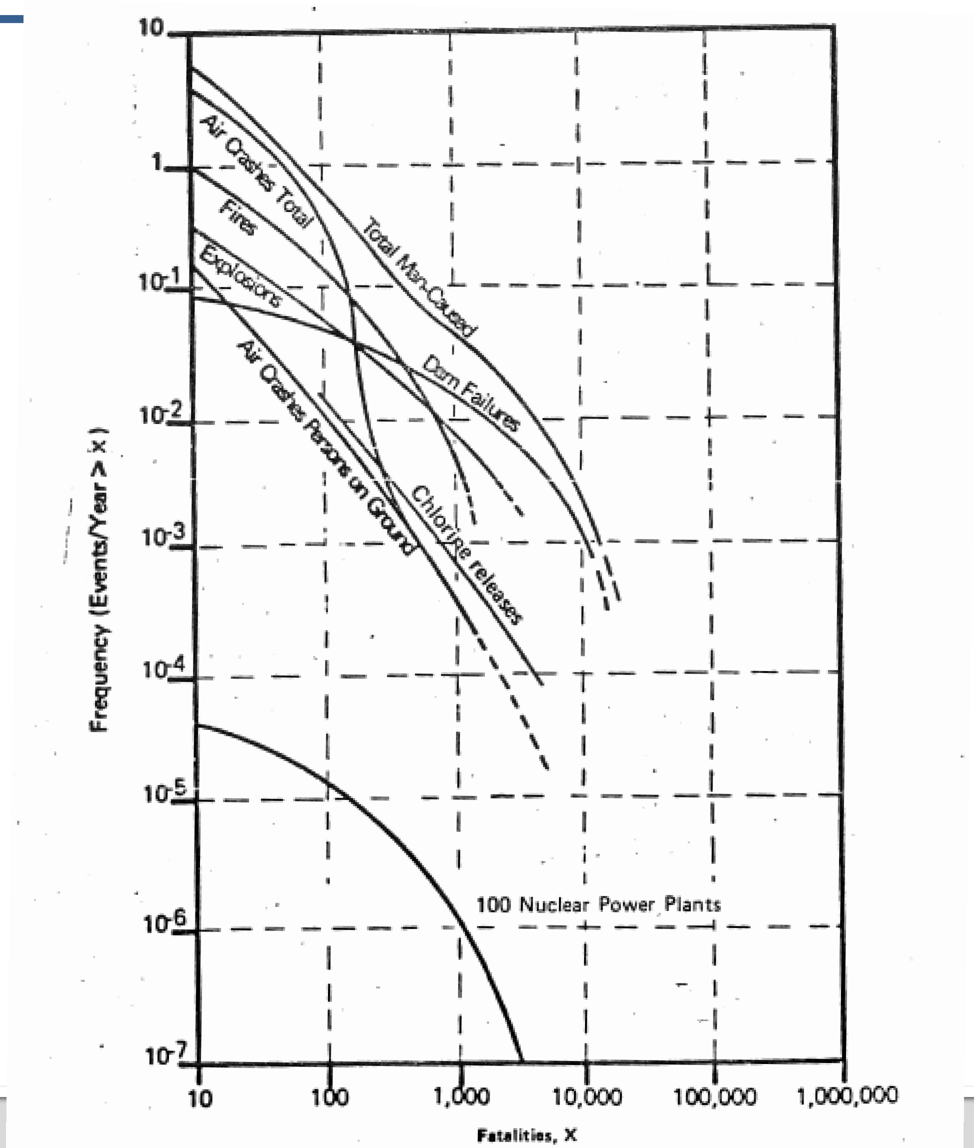
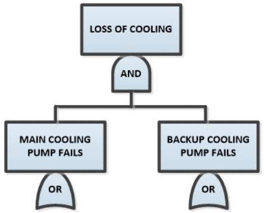


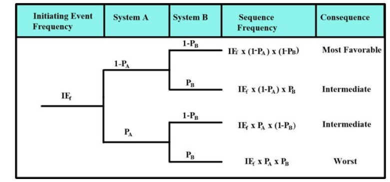
Figure 6-1 Frequency of Man-Caused Events Involving Fatalities

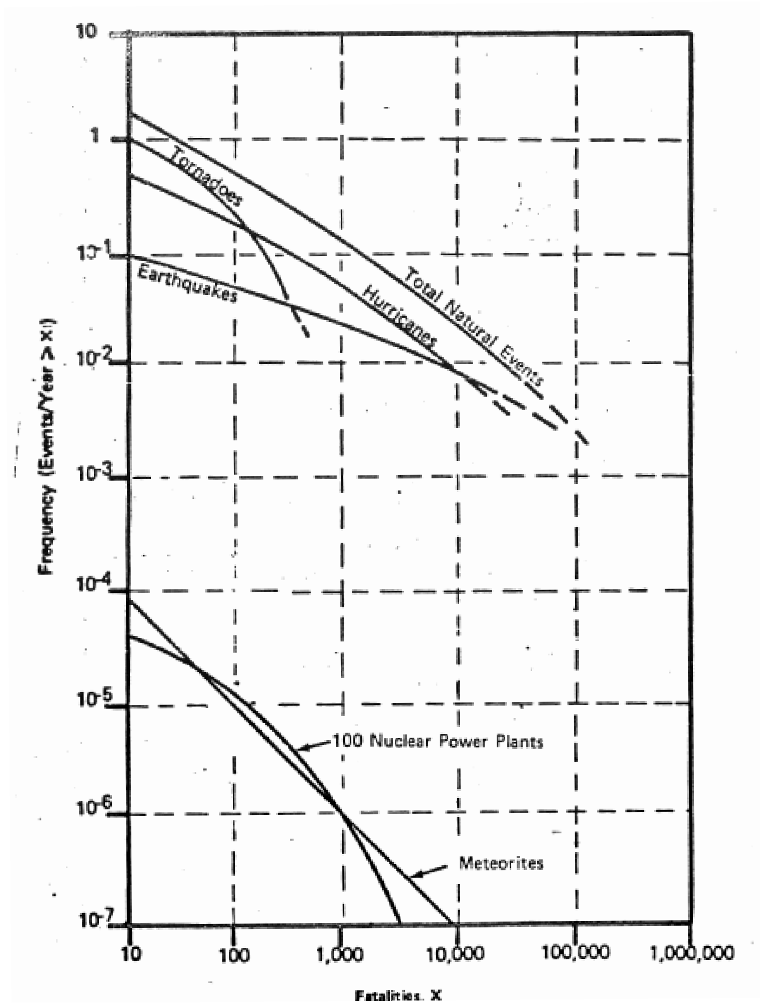


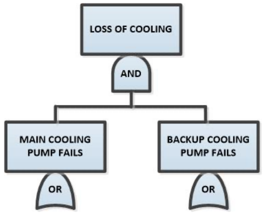




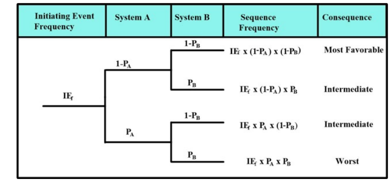
Frequency of Natural Events Involving Fatalities

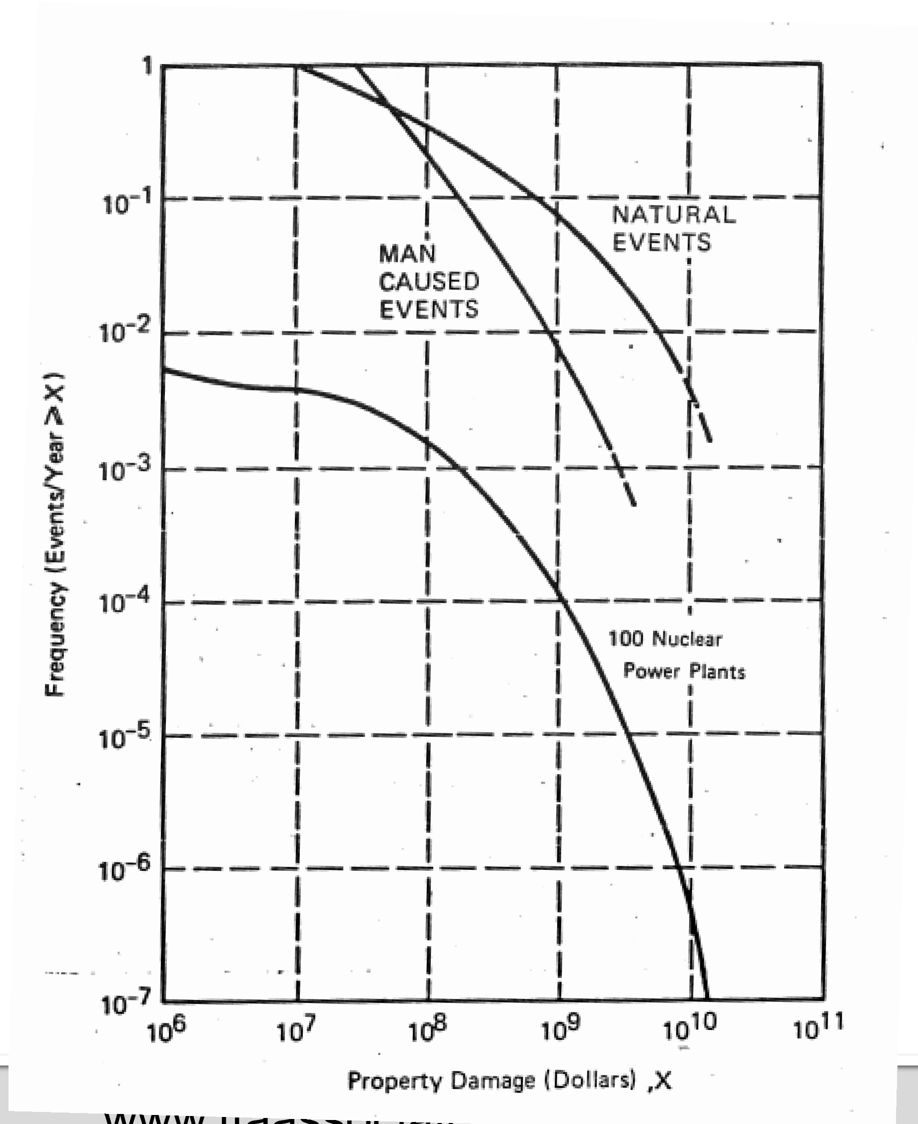


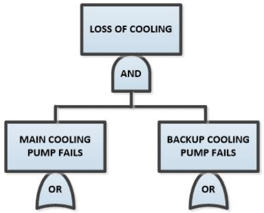




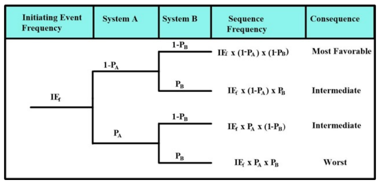
Frequency of Natural Events Involving Property Damage



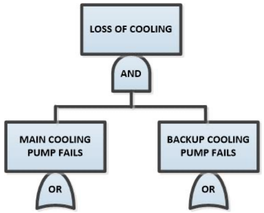




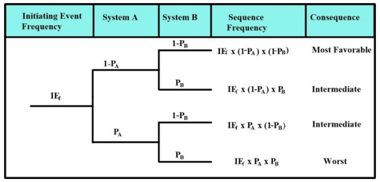
Review Group (chaired by Prof. Harold Lewis) findings and recommendations



- WASH-1400 methodology is sound and both can and should be used by NRC to make regulatory process more rational and to more effectively align its resources to risk.
- Do not believe that fires, earthquake and human accident initiation contribute insignificantly to the overall risk
- Doubt the completeness of the analysis
- Unable to assess the accuracy of the absolute probabilities but believe that uncertainties are understated – use of invented statistical techniques
- Report is inscrutable impairing both its usefulness and quality of peer review
- Executive Summary is a poor description of report



Commission response to peer review findings



- **The following are excerpts from the NRC statement of January 18, 1979:**
  - *" … the Commission has reexamined its views regarding the study in light of the Review Group critique."*
  - *"The Commission withdraws any explicit or implicit part endorsement of the Executive Summary."*
  - *"… the Commission does not regard as reliable the Reactor Safety Study's numerical estimates of the overall risk of reactor accidents."*