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SUSTAINABLE ENERGY

Prof. Michael W. Golay

Nuclear Engineering Dept.

PROBABILISTIC RISK ANALYSIS

INTRODUCTION OF THE BASIC ELEMENTS OF PROBABILISTIC RISK (PRA) ANALYSES

- Fault Trees
- Risk
- Data
- Uncertainties
- Nuclear Power Plant PRA Structure
- Typical Results

THE PRE-PRA ERA (prior to 1975)

- Management of (unquantified at the time) uncertainty was always a concern.
- Defense-in-depth and safety margins became embedded in the regulations.
- “Defense-in-Depth is an element of the NRC’s safety philosophy that employs successive compensatory measures to prevent accidents or mitigate damage if a malfunction, accident, or naturally caused event occurs at a nuclear facility.” [Commission’s White Paper, February, 1999]
- Design Basis Accidents are postulated accidents that a nuclear facility must be designed and built to withstand without loss to the systems, structures, and components necessary to assure public health and safety.

TECHNOLOGICAL RISK ASSESSMENT

- Study the system as an integrated socio-technical system.

  Probabilistic Risk Assessment (PRA) supports Risk Management by answering the questions:

  - What can go wrong? (accident sequences or scenarios)
  - How likely are these scenarios?
  - What are their consequences?

$$\text{Risk} = \text{Expected consequences} = \sum_{\text{Sequences},i} \text{Prob}_i * \text{Consequence}_i$$

DEFINITION OF RISK

Event Risk ≡ Vector (Set) of Expected Consequences From an Event

For an Event of Type i, the Associated Risk Vector, $\vec{R}_i$

$\vec{R}_i = \langle \vec{C}_i \rangle =$ (Probability of Event, i) * (Set of Consequences' of Event, i)

= [(Frequency of Event, i) * (Time Interval of Interest)] * (Set of Consequences of Event, i)

CORE DAMAGE RISK DUE TO N DIFFERENT CORE DAMAGE EVENTS

$$\vec{R}_{total} = \sum_{i=1}^{N} \vec{R}_i = \sum_{i=1}^{N} p_i \begin{bmatrix} \text{Consequence}_{1,\, i} \\ \Downarrow \\ \text{Consequence}_{M,\, i} \end{bmatrix}$$

Total Risk is the Sum Over All Possible Events of the Risks Associated with Each Event, Respectively

RISK CALCULATION

$$\overrightarrow{\text{Risk}} = \sum_{\substack{\text{i, All Event} \\ \text{Sequences}}} \overline{C}_i p_i = \langle \overline{C} \rangle = \begin{bmatrix} \langle C_a \rangle \\ \langle C_b \rangle \\ \downarrow \\ \langle C_n \rangle \end{bmatrix}$$

$\overline{C}_i$ = Vector of consequences associated with the i^{th} event sequence

p_i = Probability of the i^{th} event sequence

$\langle \overline{C} \rangle$ = Mean, or expected, consequence vector

$\langle C_a \rangle$ = Mean, or expected, consequence of type a, summed over all event sequences

EXAMPLE

$$\overline{C}_i = \begin{bmatrix} \text{Offsite acute fatalities due to event i} \\ \text{Offsite latent fatalities due to event i} \\ \text{Onsite acture fatalities due to event i} \\ \text{Onsite latent fatalities due to event i} \\ \text{Offsite property loss due to event i} \\ \text{Onsite property loss due to event i} \\ \text{Costs to other NPPs due to event i} \end{bmatrix}$$

THE HAZARD
(some fission-product isotopes)

Isotope	Half-Life	Volatility	Health Hazard
^{131}I	8 d	Gaseous	External whole-body radiation; internal irradiation of thyroid; high toxicity
^{89}Sr	54 y	Moderately volatile	Bones and lungs
^{106}Ru	1 y	Highly volatile	Kidneys
^{137}Cs	33 y	Highly volatile	Internal hazard to whole body

DECAY HEAT

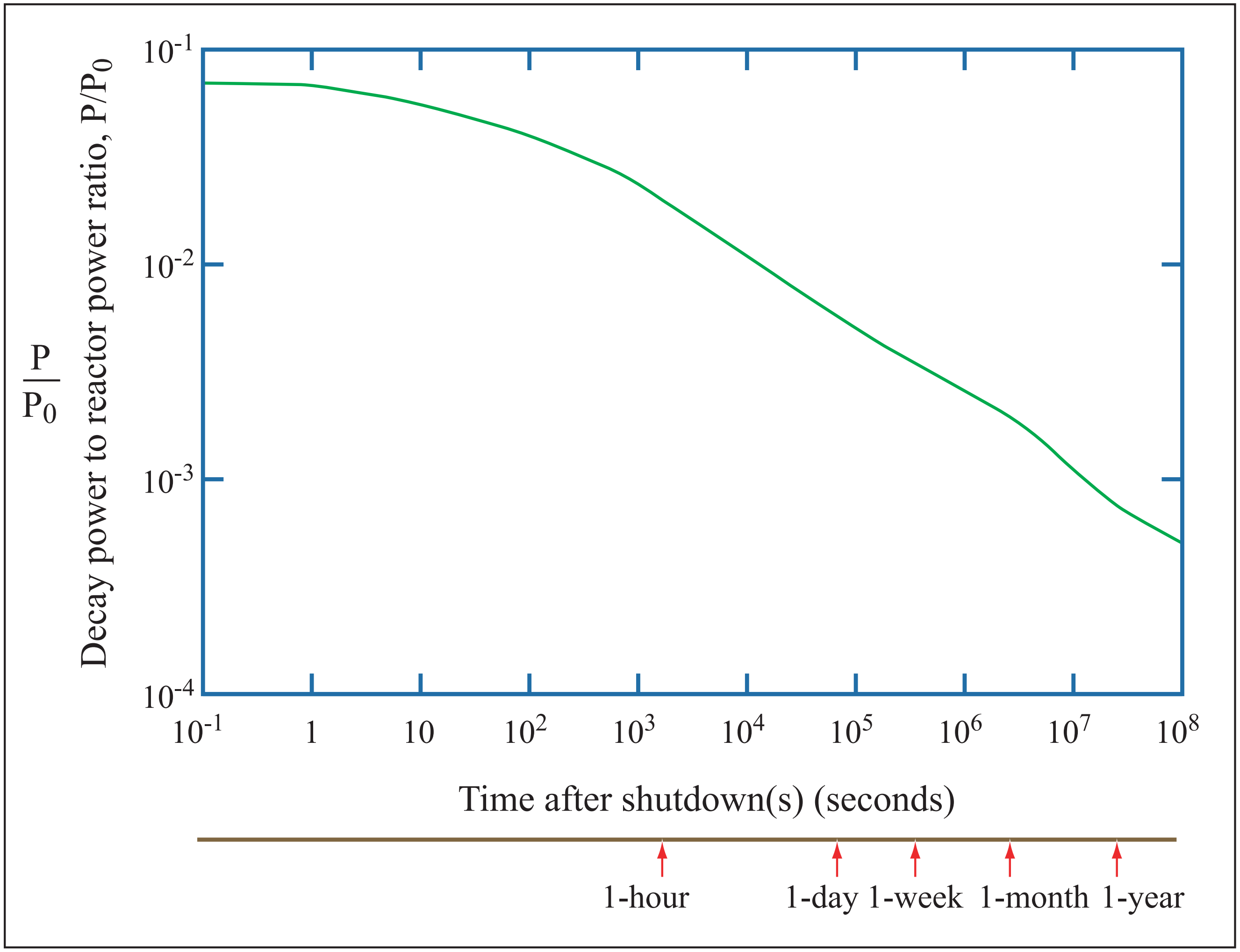


Image by MIT OpenCourseWare. Adapted from Todreas & Kazimi, *Nuclear Systems Volume I: Thermal Hydraulic Fundamentals*.

THE FARMER LINE

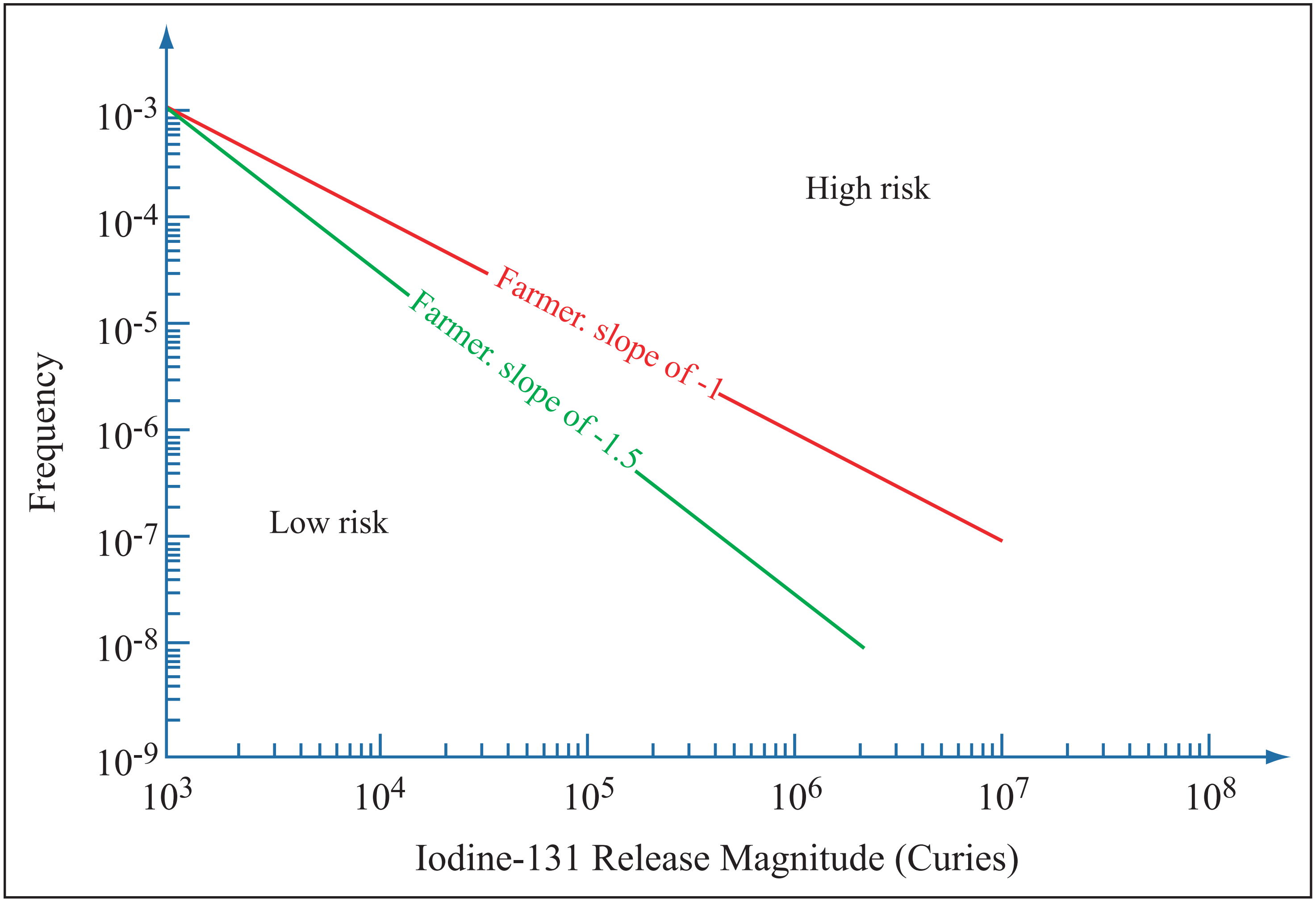


Image by MIT OpenCourseWare.

CRITICAL SAFETY FUNCTIONS HARDWARE / TRAINING / PROCEDURES / CULTURE

KEEP FISSION PRODUCTS WITHIN THE FUEL

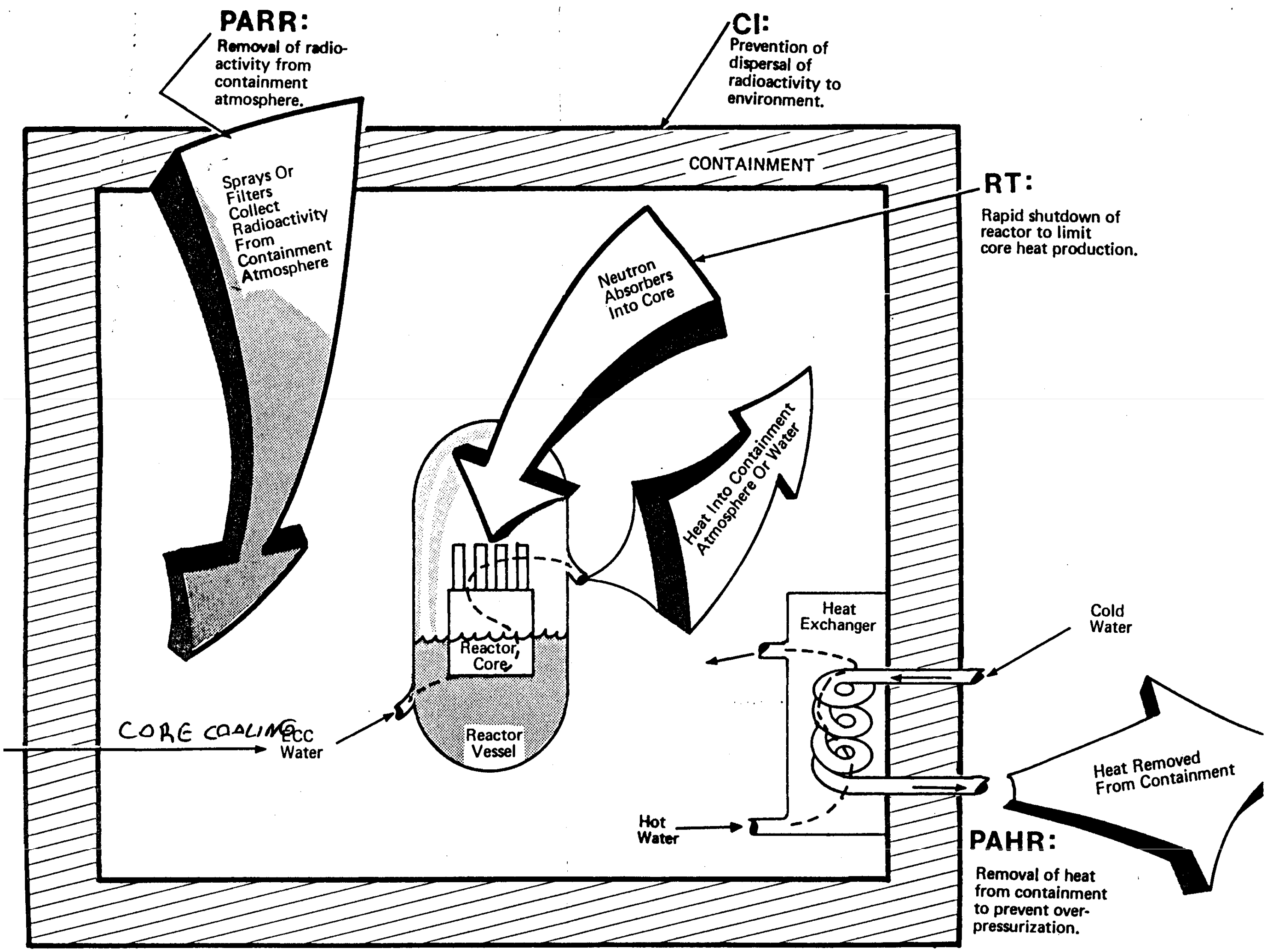
- Control Reactor Power
  - Control reactivity additions
  - Shutdown reliably
- Cool the Reactor and Spent Fuel
  - Maintain coolant inventory
  - Maintain coolant flow
  - Maintain coolant heat sinks

KEEP RADIOACTIVE MATERIAL OUT OF THE BIOSPHERE

- Maintain Containment Integrity
  - Prevent over-pressurization
  - Prevent over-heating
  - Prevent containment bypass
- Capture Material Within Containment
  - Scrubbing
  - Deposition
  - Chemical capture

SHIELD PERSONNEL FROM RADIATION

EMERGENCY SAFETY FUNCTIONS



Reactor Safety Study, WASH-1400

REACTOR SAFETY STUDY (WASH-1400; 1975)

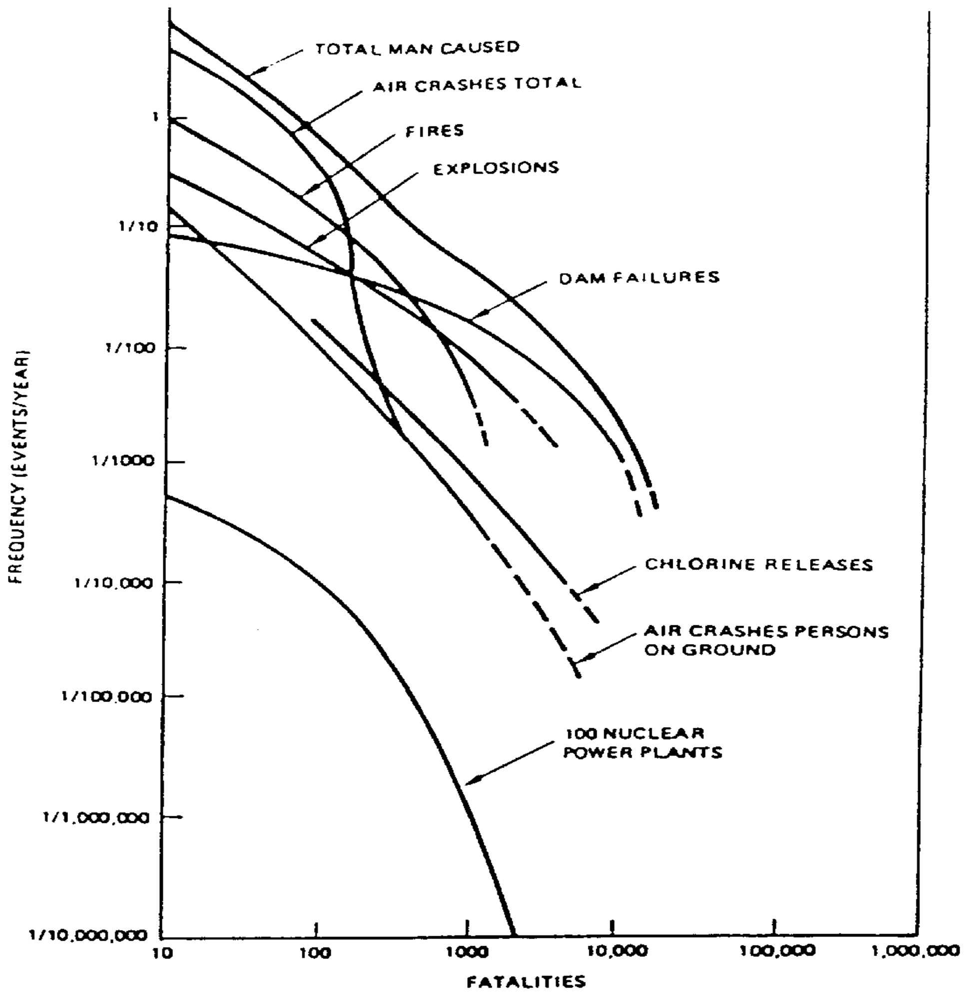
Prior Beliefs:

1. Protect against large LOCA.
2. CDF is low (about once every 100 million years, 10-8 per reactor year) .
3. Consequences of accidents would be disastrous.

Major Findings:

1. Dominant contributors: Small LOCAs and Transients.
2. CDF higher than earlier believed (best estimate: 5x10-5, once every 20,000 years; upper bound: 3x10-4 per reactor year, once every 3,333 years).
3. Consequences significantly smaller.
4. Support systems and operator actions very important.

RISK CURVES



Frequency of Fatalities Due to Man-Caused Events (RSS)

Source: Reactor Safety Study, Nuclear Regulatory Commission, WASH-1400.

RISK ASSESSMENT REVIEW GROUP

- “We are unable to define whether the overall probability of a core melt given in WASH-1400 is high or low, but we are certain that the error bands are understated.”

- WASH-1400 is "inscrutable."

- "…the fault -tree/event-tree methodology is sound, and both can and should be more widely used by NRC."

- "PSA methods should be used to deal with generic safety issues, to formulate new regulatory requirements, to assess and revalidate existing regulatory requirements, and to evaluate new designs."

COMMISSION ACTIONS (Jan. 18, 1979)

- "…the Commission has reexamined its views regarding the Study in light of the Review Group's critique."
- "The Commission withdraws any explicit or implicit past endorsement of the Executive Summary."
- "…the Commission does not regard as reliable the Reactor Safety Study's numerical estimate of the overall risk of reactor accidents."

NPP: END STATES

- Various states of degradation of the reactor core.
- Release of radioactivity from the containment.
- Individual risk.
- Numbers of early and latent deaths.
- Number of injuries.
- Land contamination.

NPP: INITIATING EVENTS

- Transients
  - Loss of offsite power
  - Turbine trip
  - Others
- Loss-of-Coolant Accidents (LOCAs)
  - Small LOCA
  - Medium LOCA
  - Large LOCA

LOSS-OF-OFFSITE-POWER EVENT TREE

LOOP	Secondary Heat Removal	Bleed & Feed	Recirc.	Core

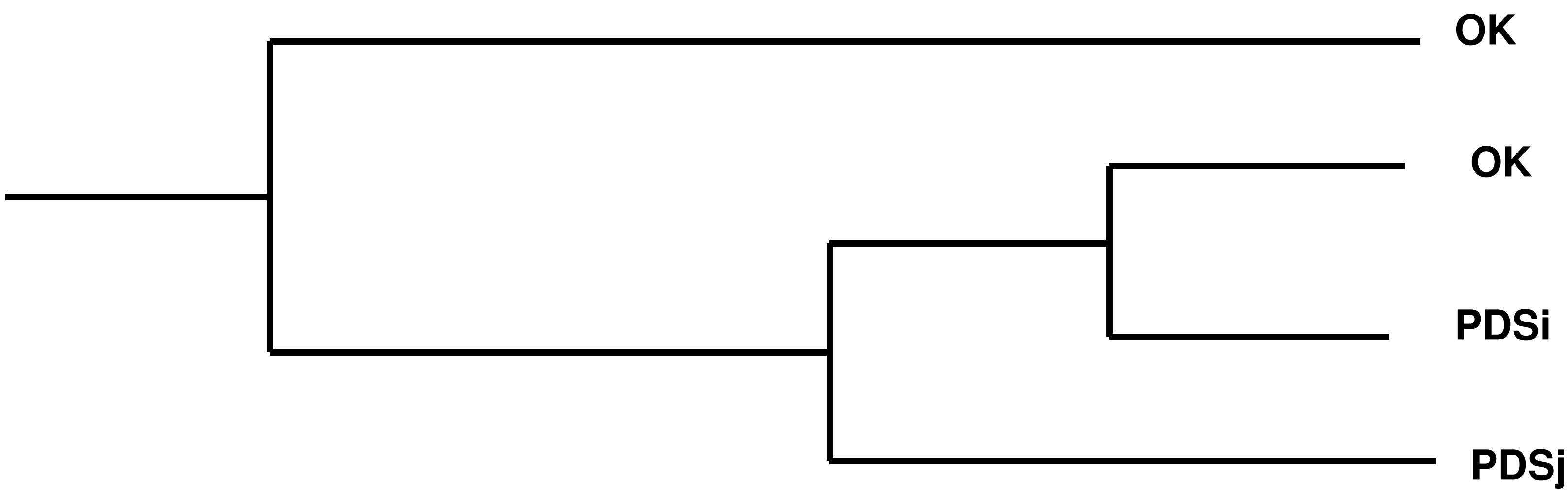
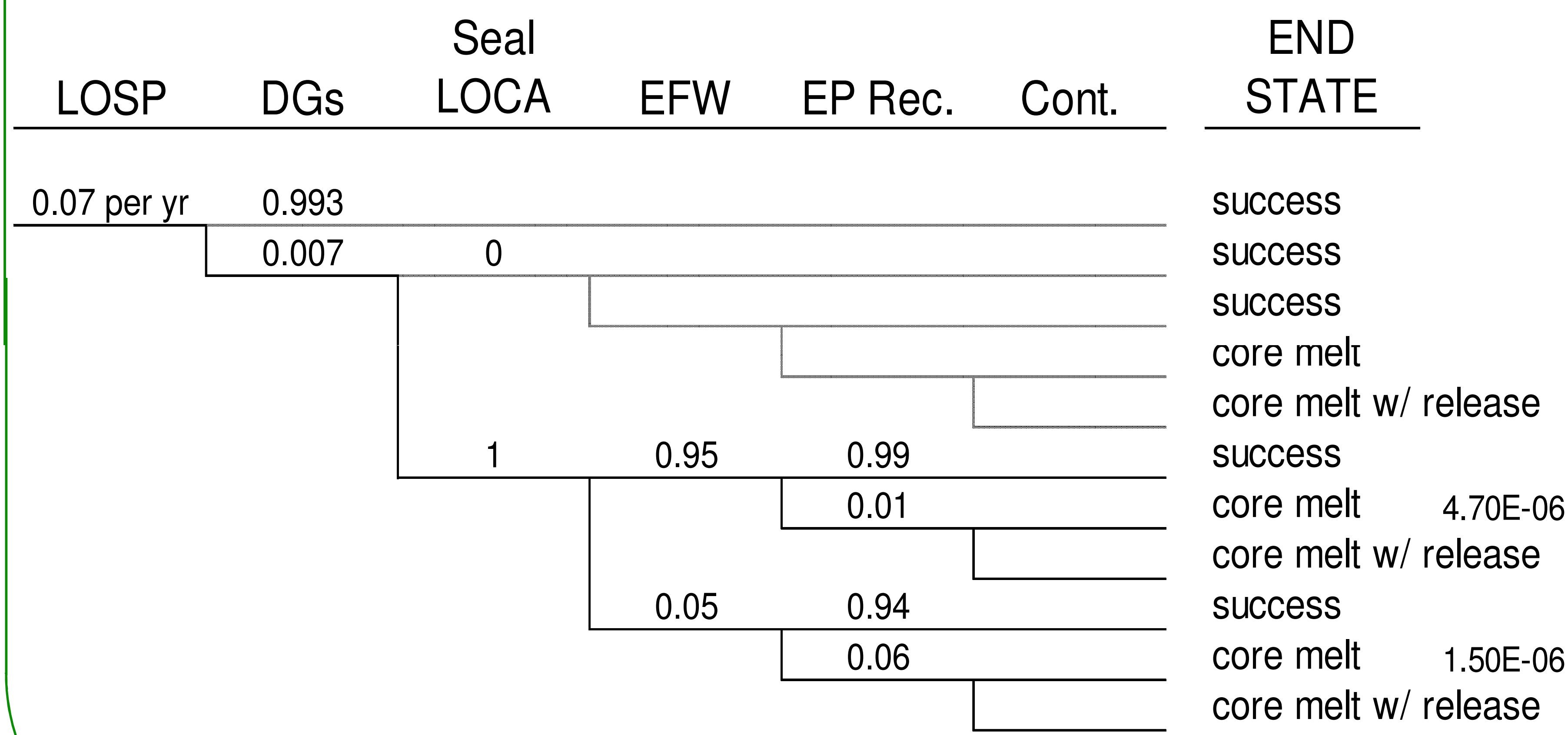


ILLUSTRATION EVENT TREE: Station Blackout Sequences



Courtesy of K. Kiper. Used with permission.

From: K. Kiper, MIT Lecture, 2006

PRA MODEL OVERVIEW AND SUBSIDIARY OBJECTIVES

CDF
10^{-4}/ry

LERF
10^{-5}/ry

QHOs

Level I

PLANT MODEL

Results

Accident sequences leading to plant damage states

Level II

CONTAINMENT MODEL

Results

Containment failure/release sequences

Level III

SITE/CONSEQUENCE MODEL

Results

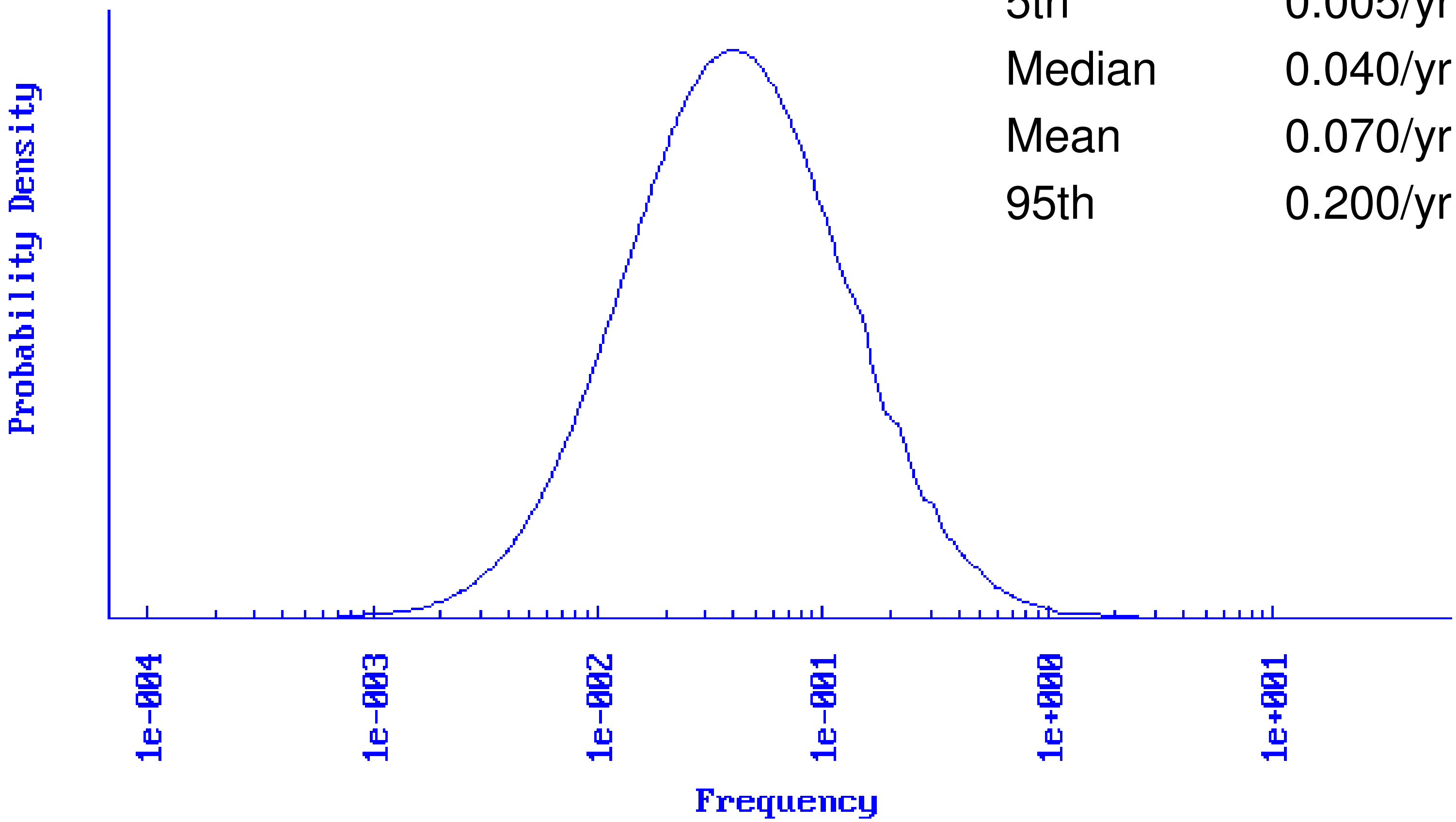
Public health effects

PLANT MODE
At-power Operation
Shutdown / Transition Evolutions

SCOPE
Internal Events
External Events

Uncertainties

LOSP DISTRIBUTION

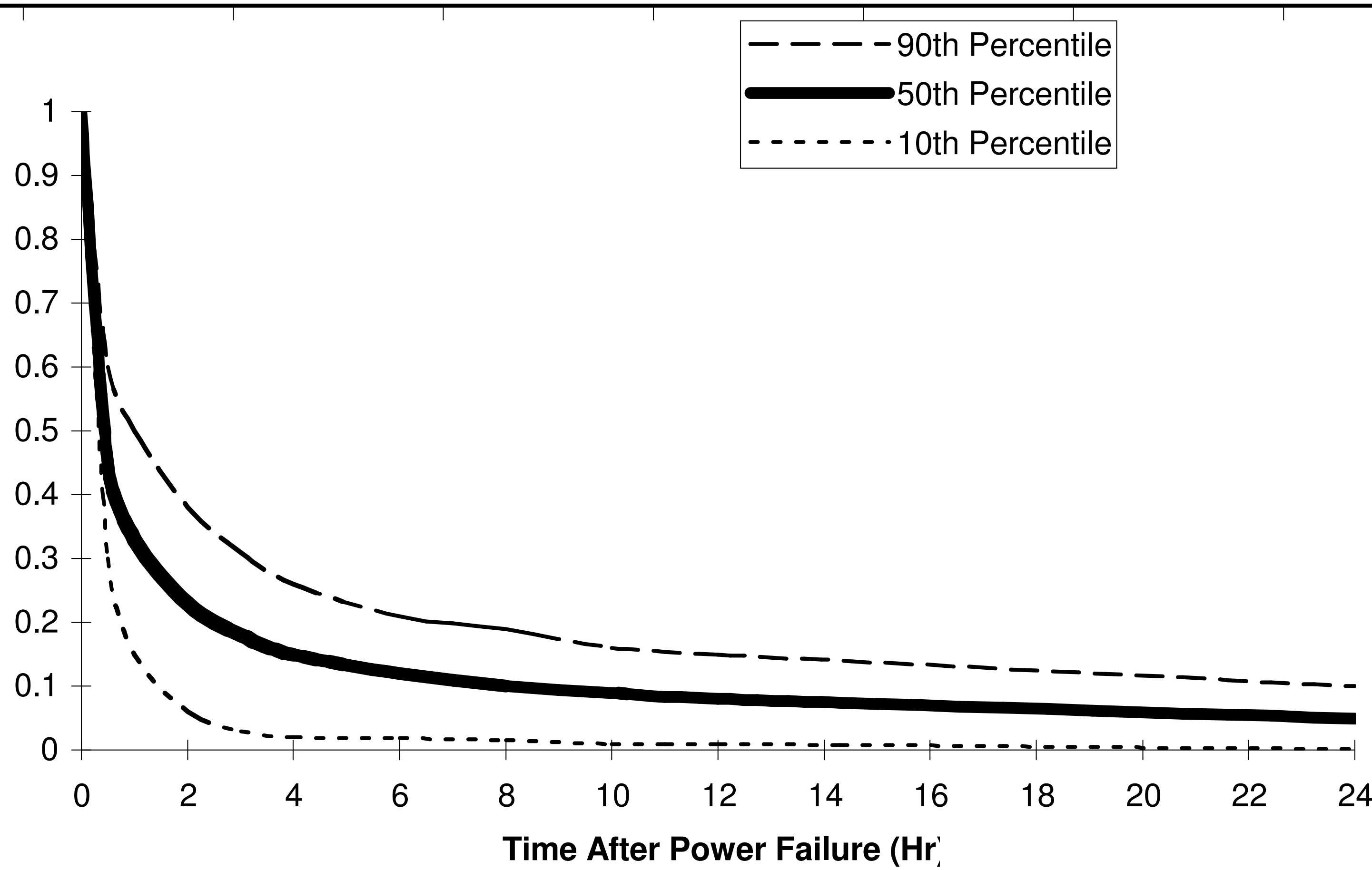


Epistemic Uncertainties

5th	0.005/yr	(200 yr)
Median	0.040/yr	(25 yr)
Mean	0.070/yr	(14 yr)
95th	0.200/yr	(5 yr)

From: K. Kiper, MIT Lecture, 2006

OFFSITE POWER RECOVERY CURVES



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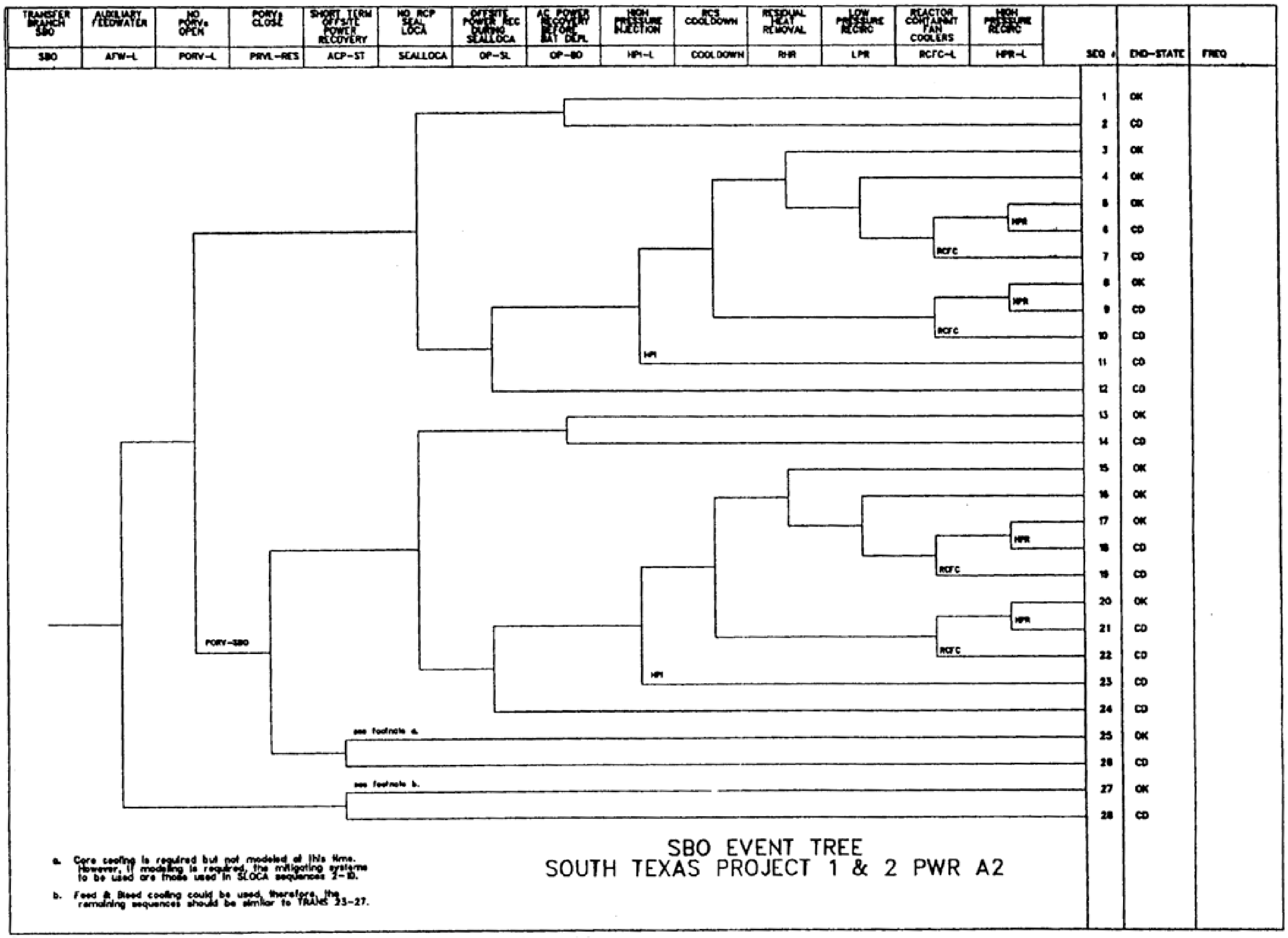
From: K. Kiper, MIT Lecture, 2006

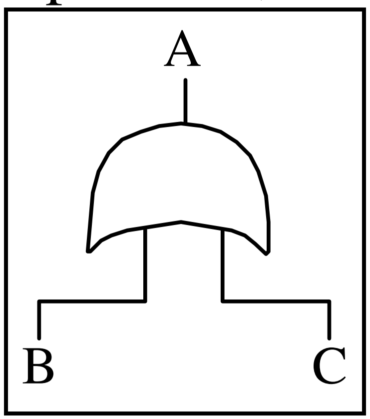
SOUTH TEXAS PROJECT 1 & 2 PWR A2 STATION BLACKOUT EVENT TREE



South Texas Project 1 & 2, Rev 2QA, Fig. 2-2, p. 2-7.

LOGIC SYMBOLS ("GATES")

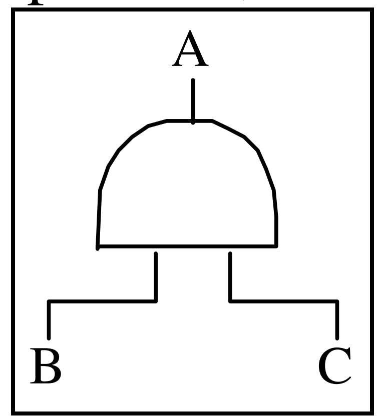
Operation, OR



Meaning:

Event A occurs when either event B or C occurs

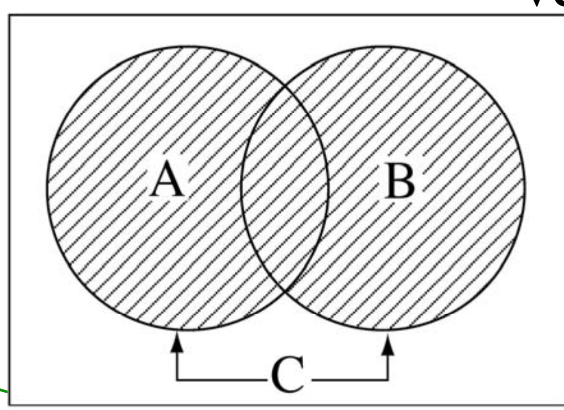
Operation, AND

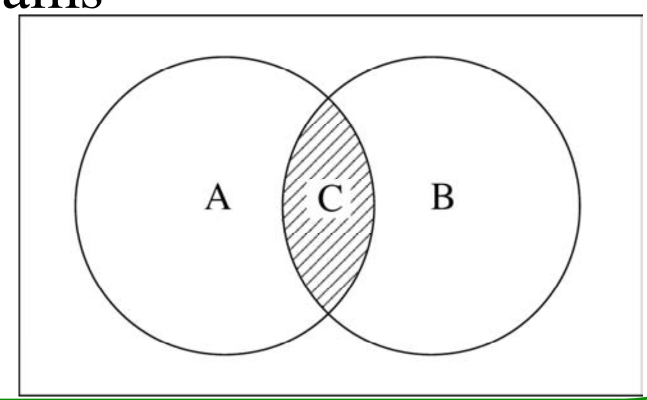


Meaning:

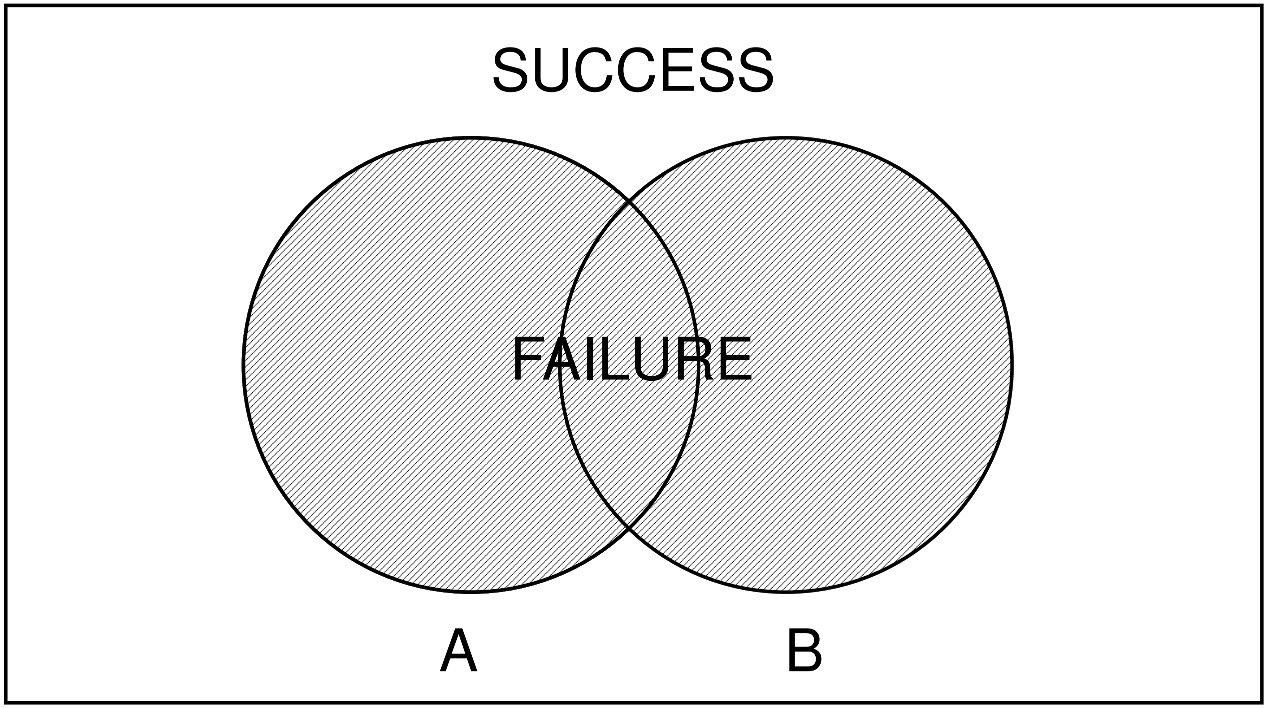
Event A occurs when both events B and C occur

Venn Diagrams





CONSIDER SYSTEM MINIMAL CUT SETS A & B

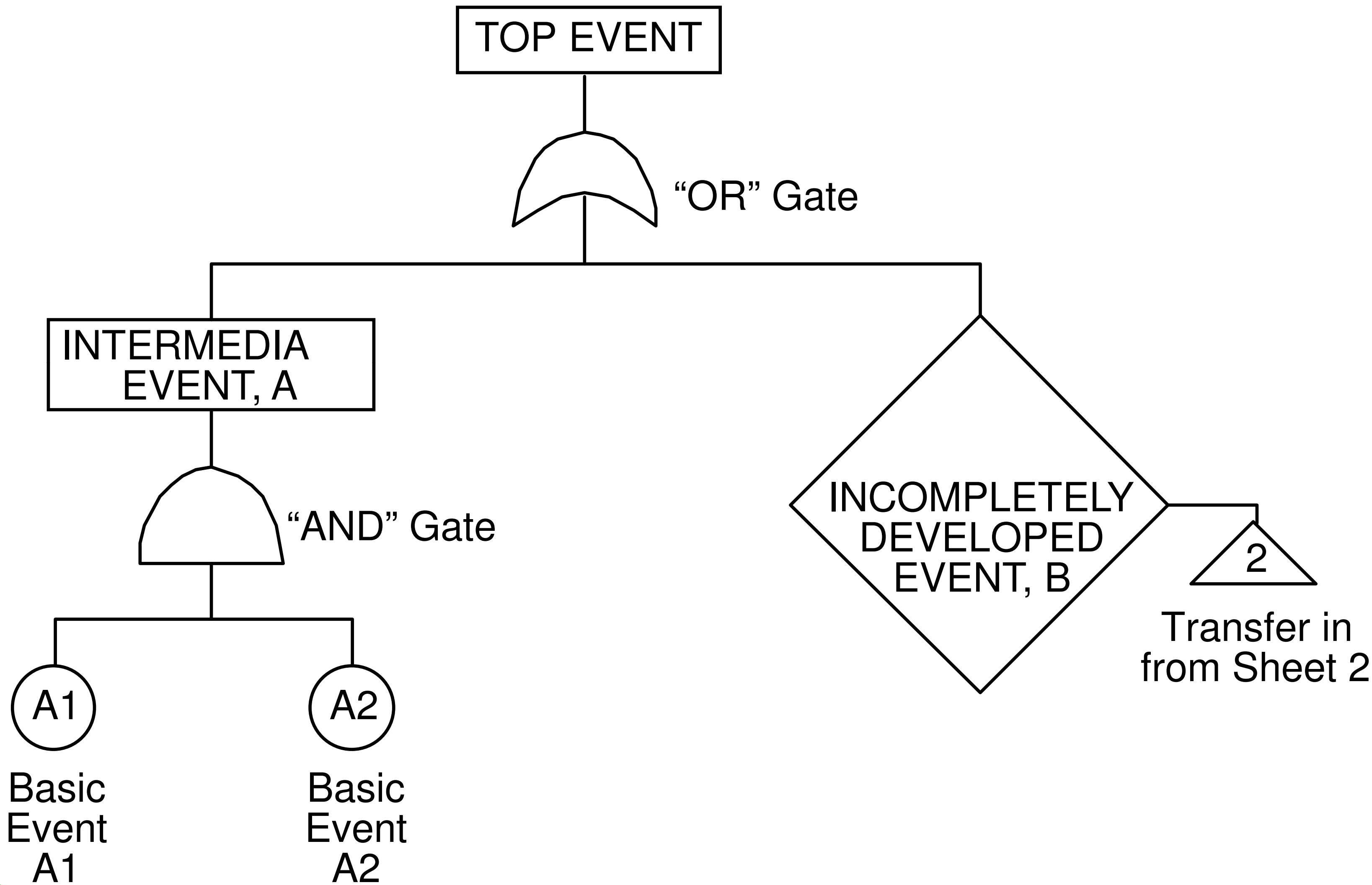


Prob Failure $= Prob_A + Prob_B$ - [Prob (B/A) $Prob_A$]
$= Prob_A + Prob_B - (Prob_A * Prob_B)$
if A & B are independent

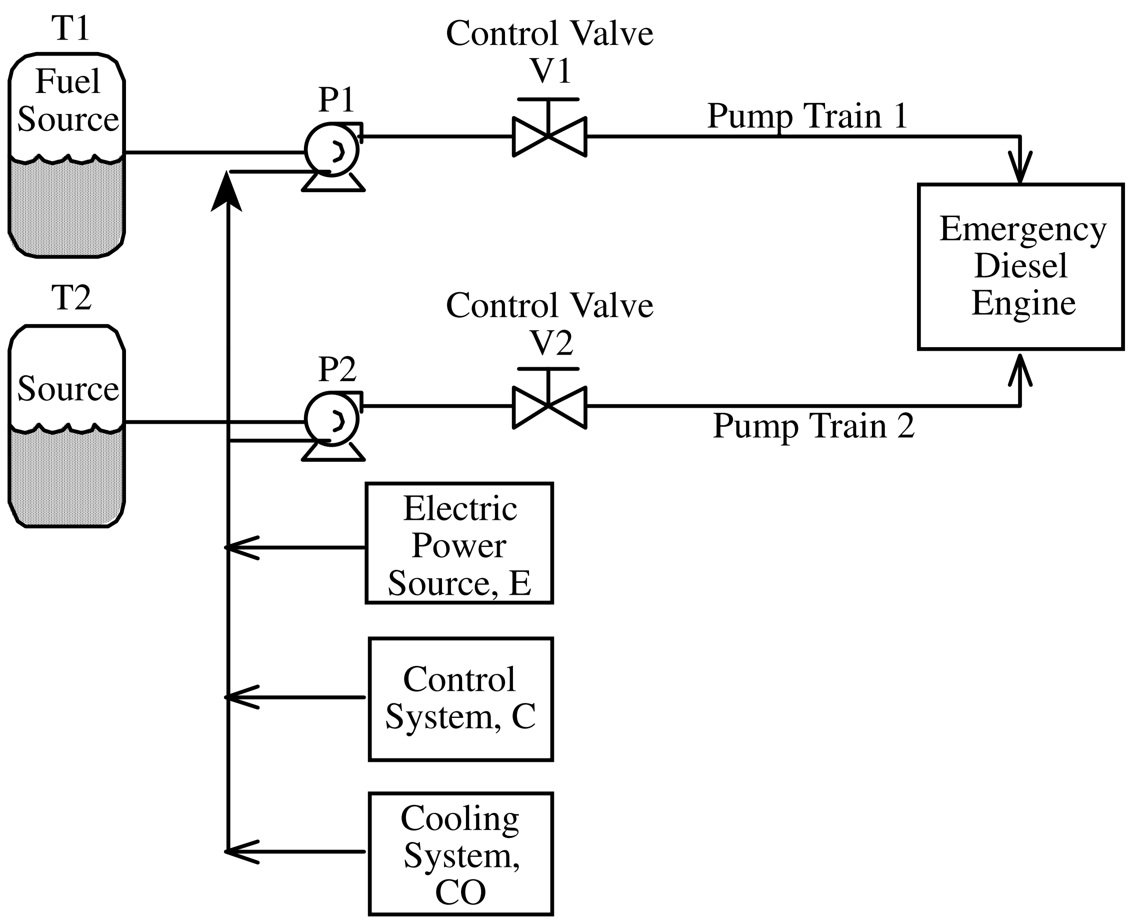
For a Good System:

$Prob_A$, $Prob_B << 1$ and $Prob_A * Prob_B << Prob_A$ or $Prob_B$, and
Prob Failure $\leq Prob_A + Prob_B$ (rare event approximation)

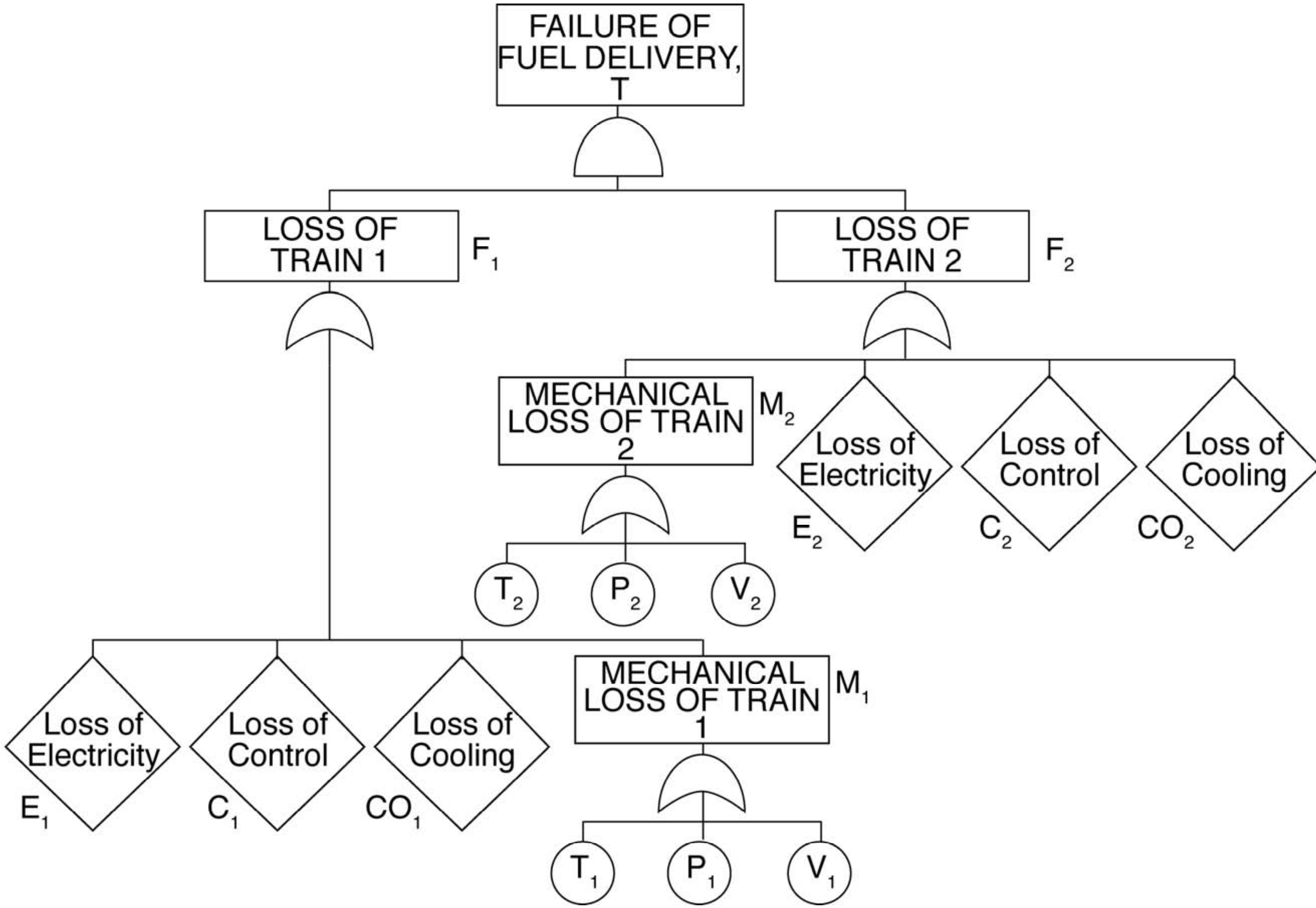
ILLUSTRATION OF ELEMENT OF FAULT TREE ELEMENTS



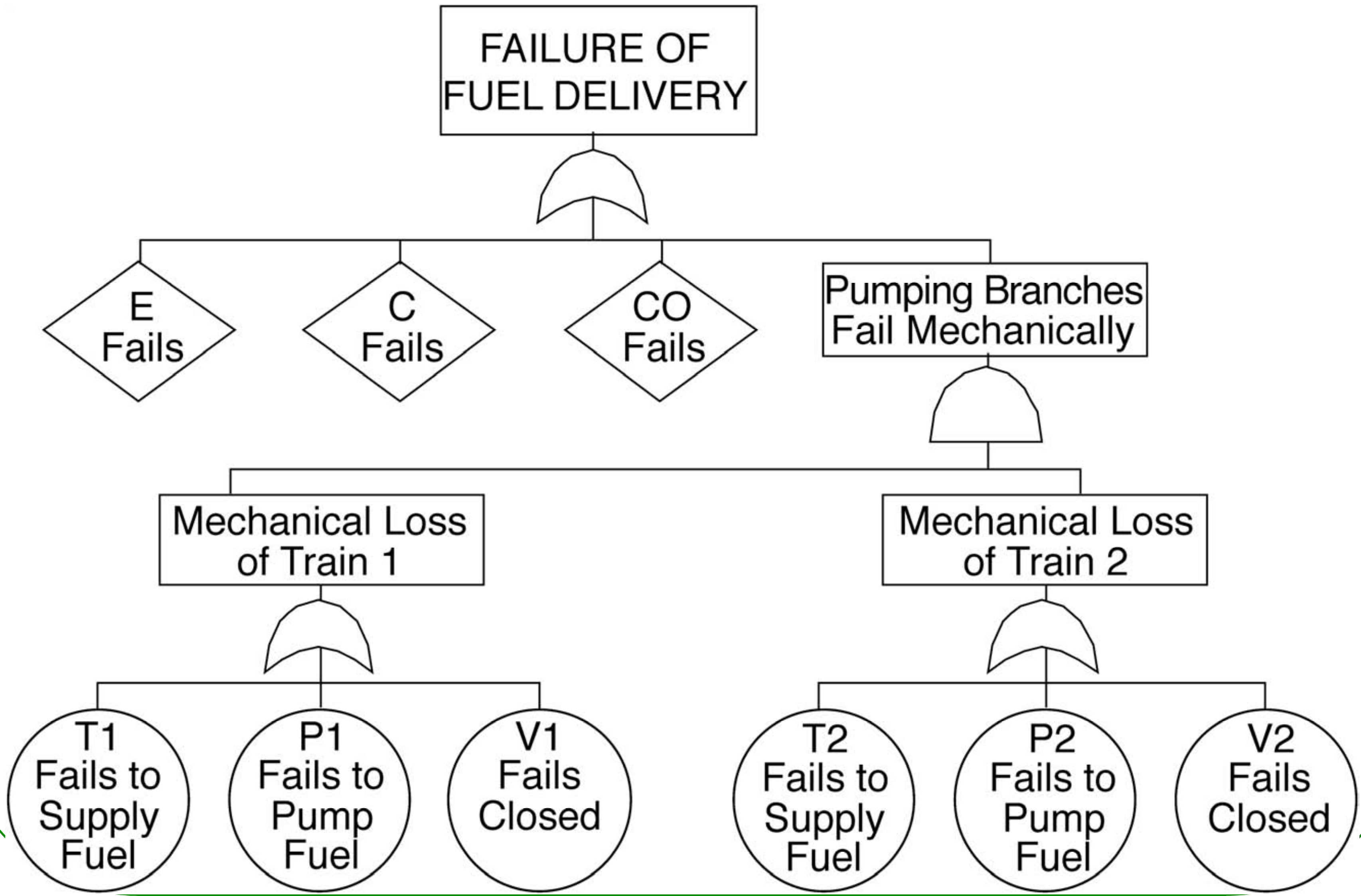
AN EXAMPLE OF A PUMPING SYSTEM



FAULT TREE FOR THE FUEL PUMPING SYSTEM



FAULT TREE FOR THE FUEL PUMPING SYSTEM



CUT SETS AND MINIMAL CUT SETS

CUT SET: A cut set is any set of failures of components and actions sufficient to cause system failure.

MINIMAL CUT SET: A minimal cut set is a set of failures <u>necessary</u> to cause system failure. A minimal cut set contains only a <u>single</u> cut set.

PUMPING SYSTEM EXAMPLE MINIMAL CUT SETS

Any Binary Combination of an Element of

$$\underbrace{\begin{bmatrix} \text{T1, Tank} \\ \text{P1, Pump} \\ \text{V1, Valve} \end{bmatrix}}_{\text{Train 1}} \text{ and of } \underbrace{\begin{bmatrix} \text{T2, Tank} \\ \text{P2, Pump} \\ \text{V2, Valve} \end{bmatrix}}_{\text{Train 2}}$$

$$\left.\begin{array}{ll} \text{C} & \text{Control System} \\ \text{E} & \text{Electric Power Source} \\ \text{CO} & \text{Cooling System} \end{array}\right\} \begin{array}{l} \text{Dependent Failure of} \\ \text{Pumping Train 1 and 2} \end{array}$$

Failure of Any Minimal Cut Set Will Result in System Failure

VENN DIAGRAM FOR FUEL SYSTEM SUPPLY FAILURE

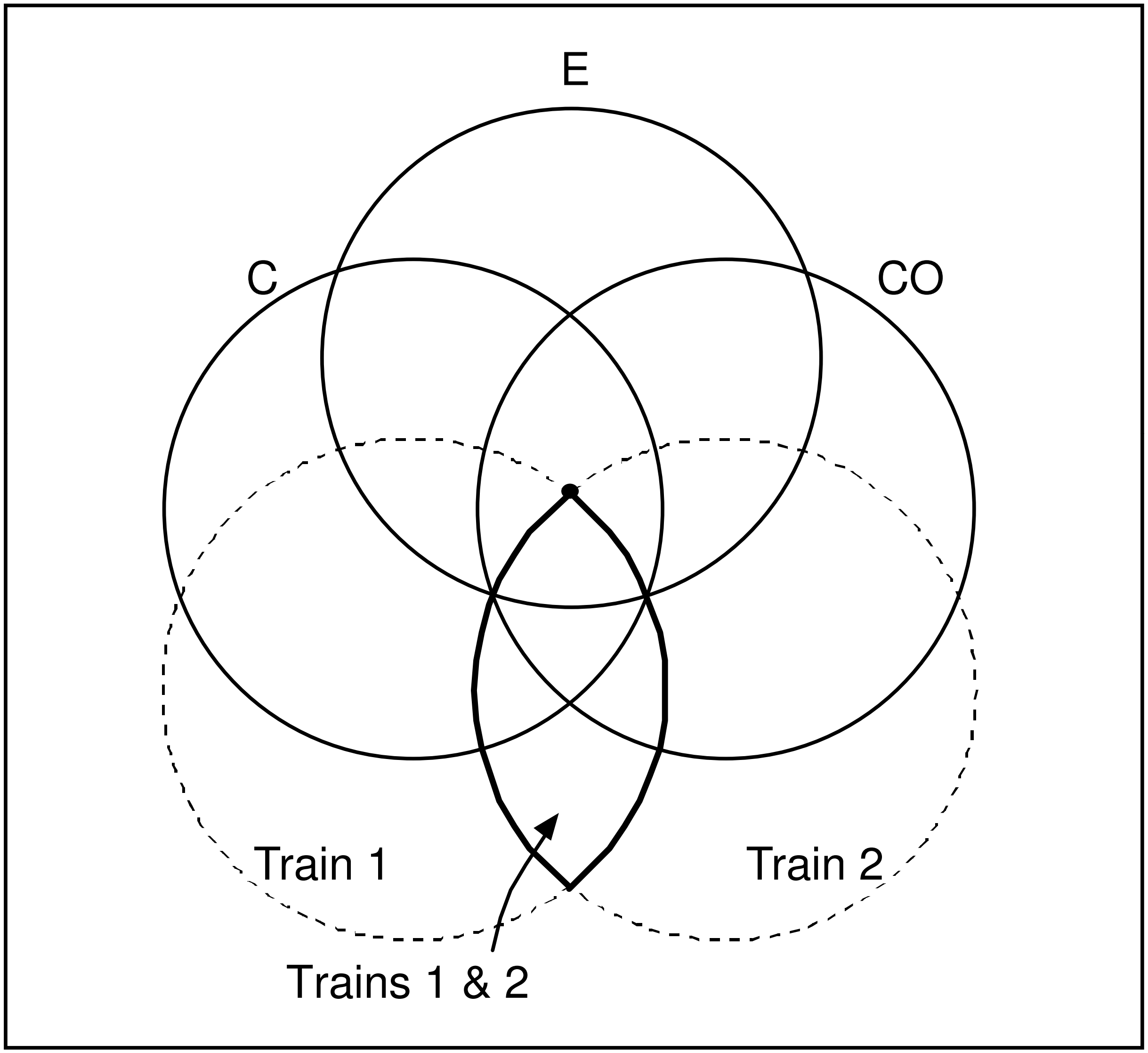


ILLUSTRATION OF DE-COMPOSITION OF TOP EVENT INTO A COMBINATION OF MINIMAL CUT SETS

$$T = E_1 \lozenge E_2 \quad (1)$$

$$E_1 = E_1 + C_1 + CO_1 + M_1 \quad (2)$$

$$E_2 = E_2 + C_2 + CO_2 + M_2 \quad (3)$$

$$M_1 = T_1 + P_1 + V_1 \quad (4)$$

$$M_2 = T_2 + P_2 + V_2 \quad (5)$$

$$E_1 = E_1 + C_1 + CO_1 + (T_1 + P_1 + V_1) \quad (6)$$

$$E_2 = E_2 + C_2 + CO_2 + (T_2 + P_2 + V_1) \quad (7)$$

NOTE: $E = E_1 = E_2$, $C = C_1 = C_2$, $CO = CO_1 = CO_2$

$$T = [(E + C + CO) + (T_1 + P_1 + V_1)] * [(E + C + CO) + (T_2 + P_2 + V_2)] \tag{8}$$

$$= \underbrace{(E_1 + C_1 + CO_1) * (E_2 + C_2 + CO_2)}_{(E + C + CO)} + (E_2 + C_2 + CO_2) * [(T_1 + P_1 + V_1) + (T_2 + P_2 + V_2)]$$

$$(E + C + CO)\ \cancelto{1}{\{1 + [(T_1 + P_1 + V_1) + (T_2 + P_2 + V_2)]\}}$$

$$+ \underbrace{(T_1 + P_1 + V_1) + (T_2 + P_2 + V_2)}_{\begin{bmatrix} T_1 \cdot T_2 + T_1 \cdot P_2 + T_1 \cdot V_2 \\ + P_1 \cdot T_2 + P_1 \cdot P_2 + P_1 \cdot V_2 \\ + V_1 \cdot T_2 + V_1 \cdot P_2 + V_1 \cdot V_2 \end{bmatrix}}$$

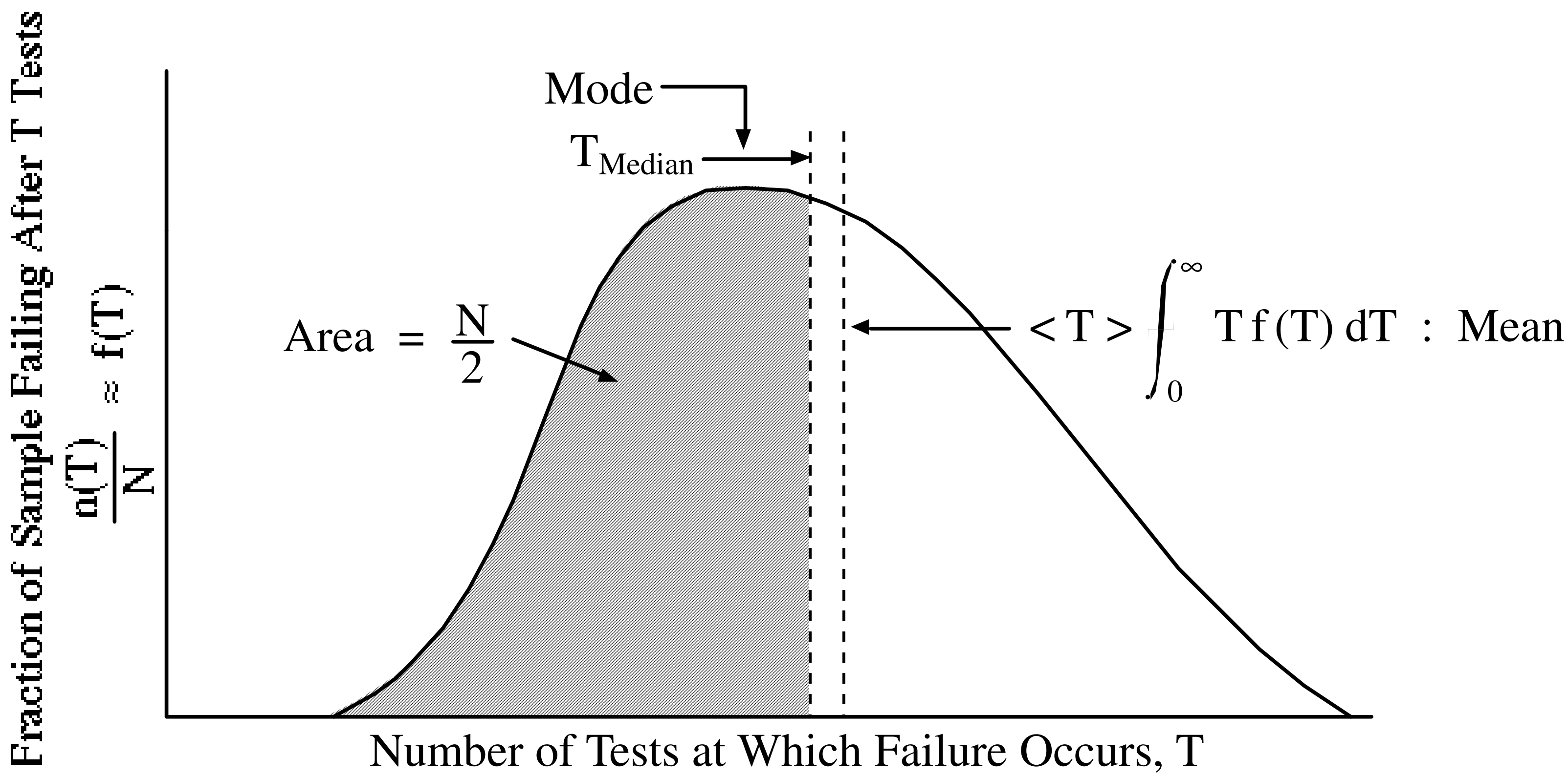
$$T = (E + C + CO) + \begin{bmatrix} T_1 \cdot T_2 + T_1 \cdot P_2 + T_1 \cdot V_2 \\ + P_1 \cdot T_2 + P_1 \cdot P_2 + P_1 \cdot V_2 \\ + V_1 \cdot T_2 + V_1 \cdot P_2 + V_1 \cdot V_2 \end{bmatrix} = \bigcup_{i=1}^{N} \left(MCS_i\right) \tag{9}$$

DATA SOURCES

- Generic Data Bases (those available are strongly safety-oriented; e.g., NPRDS/EPIX, NRC, GADS, . . .)
- Plant-Specific Data
- New Tests
- Subjective Judgment and Modeling

FAILURE PROBABILITY OF A COMPONENT

Consider a Set of N Identical Components, Which are Tested Repeatedly Until Failure



UNCERTAINTY

- FACTORS OF UNCERTAINTY
  - Randomness
  - Phenomenological Ignorance
  - Systematic Ignorance (complexity, Sensitivity)
  - Data Ignorance
- IMPORTANT UNCERTAIN PHENOMENA
  - Common Cause Failures
    - Internal
    - External
  - Rare Events (e.g., Reactor Core Melt Progression)
- TREATMENT OF UNCERTAINTY
  - Statistical (via Standard Deviation)
  - Sensitivity Analyses
  - Subjective Probability Elicitation
  - Research and Data Collection
  - Assignment of Bias

TYPES OF COMMON CAUSE FAILURES AND THEIR ASPECTS

	DEPENDENT	STRUCTURAL*	ENVIRONMENTAL	EXTERNAL*
Description of Failure Cause	Failure of an interfacing system, action or component	A common material or design flaw which simultaneously affects all components population	A change in the operational environment which affects all members of a component population simultaneously	An event originating outside the system which affects all members of a component population simultaneously
Hardware Examples	• Loss of electrical power • Loss of steam production in steam-driven feedwater system • A manufacturer provides defective replacement parts that are installed in all components of a given class	• Faulty materials • Aging • Fatigue • Improperly cured materials • Manufacturing flaw	• Dirty water in RCS with regard to pump seal • High pressure • High temperature • Vibration	• Weather: hurricanes, tornado, ice, heat, low cooling water flow • Earthquake (breaks pipe, disables cooling system, breaks containment) • Flooding→loss of electricity • Birds in engine of airplane
Human Examples	• Following a mistaken leader • An erroneous maintenance procedure is repeated for all components of a given class	• Incorrect training • Poor management • Poor motivation • Low pay	• Common cause psf's • New disease • Hunger • Fear • Noise • Radiation in control room	• Explosion • Toxic substance • Weather • Earthquake • Concern for families
Easy to Anticipate?:				
Component failure	High	Very Low	Medium	Medium
Human error	Medium	Very Low	Medium	Medium
Easy to Mitigate?:				
Component failure	High, if system designed for mitigation	Very Low, hard to design for mitigation	Low	Low
Human error	High, if feedback provided to identify the error promptly	Very Low, the factors making CCF likely also discourage being prepared for correction	Low	Low

* Usually there are no precursors

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10^{-4}/ry

LERF
10^{-5}/ry

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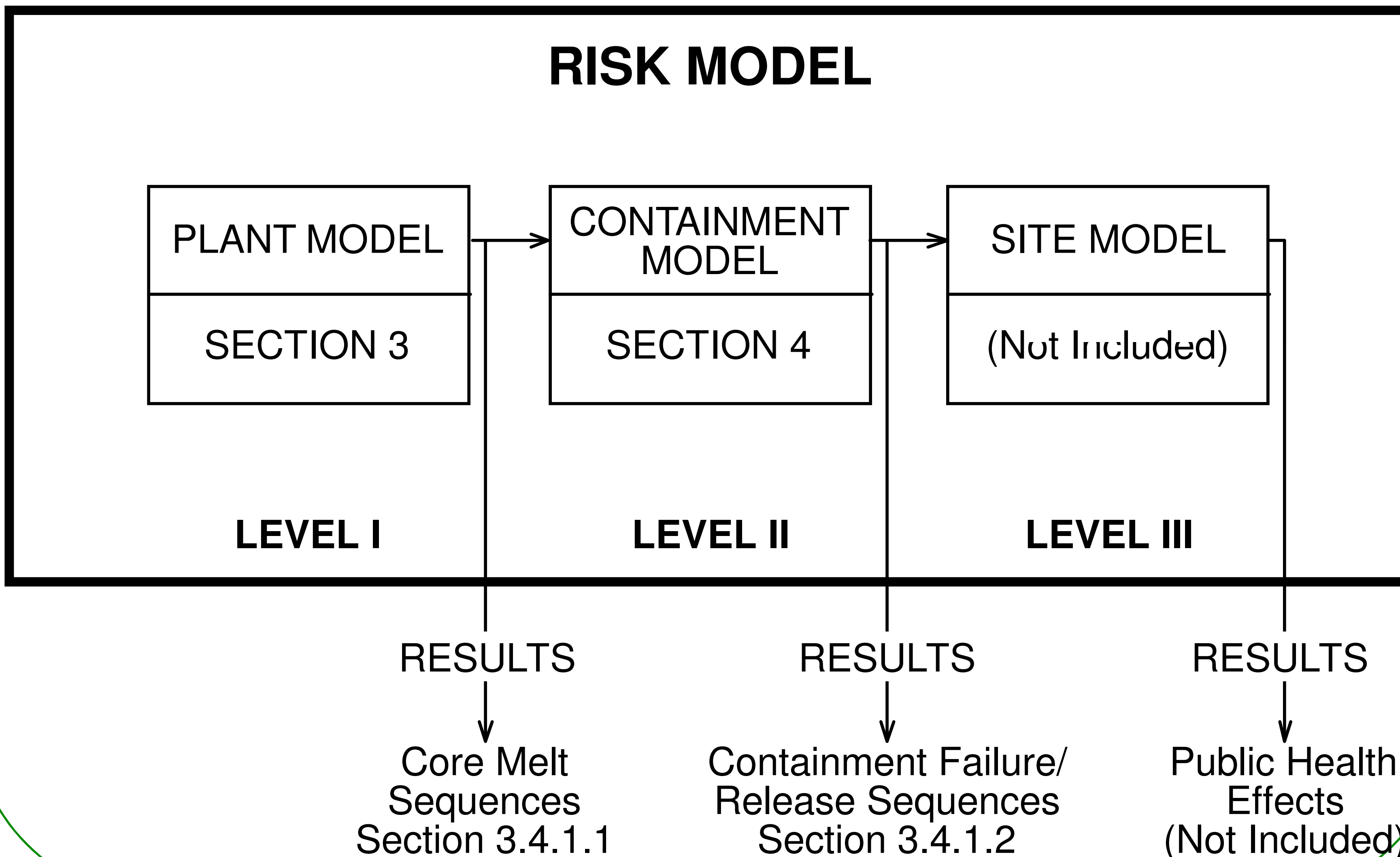
Shutdown / Transition Evolutions

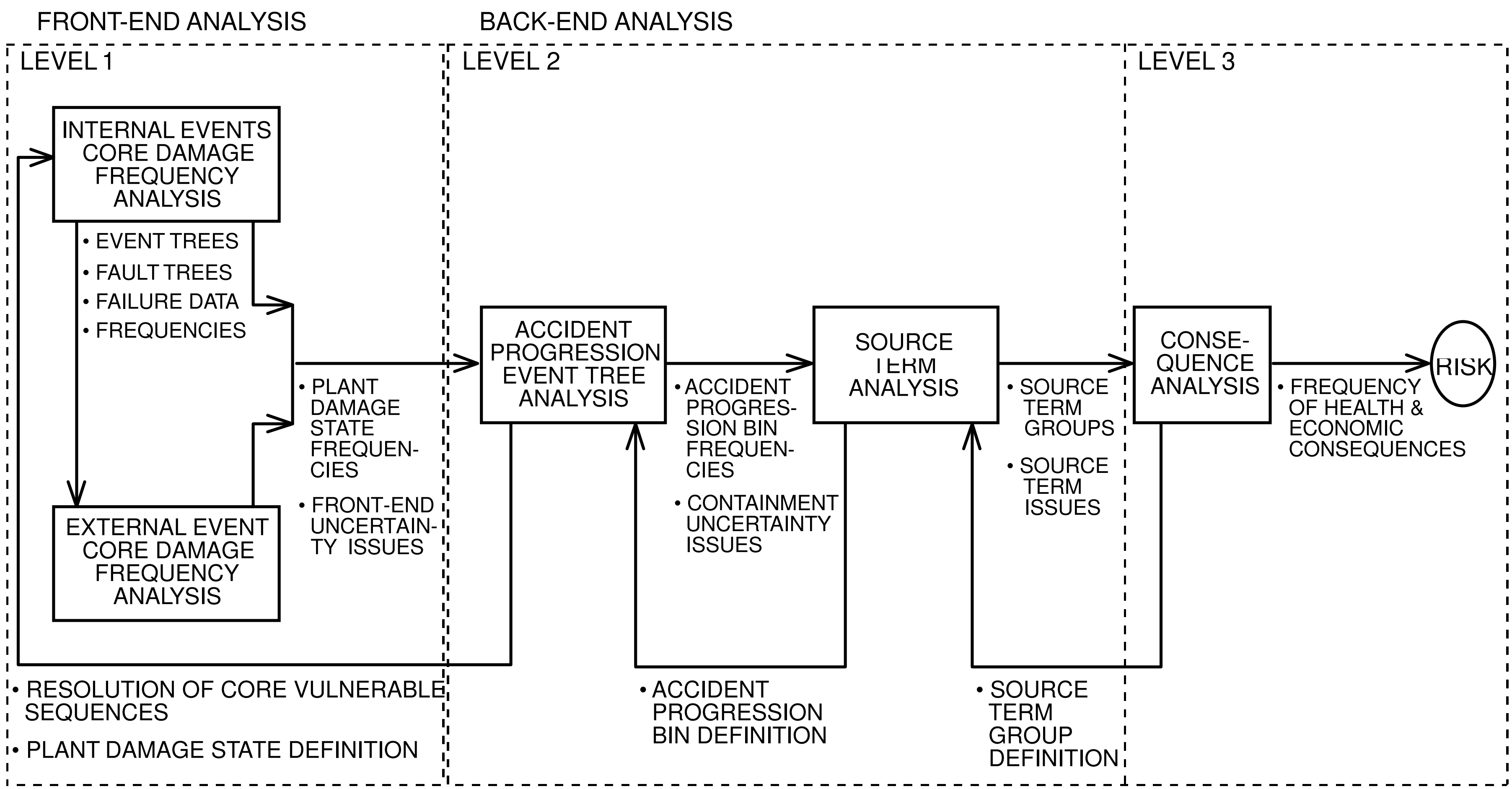
SCOPE
Internal Events
External Events

Uncertainties

RISK MODEL OVERVIEW

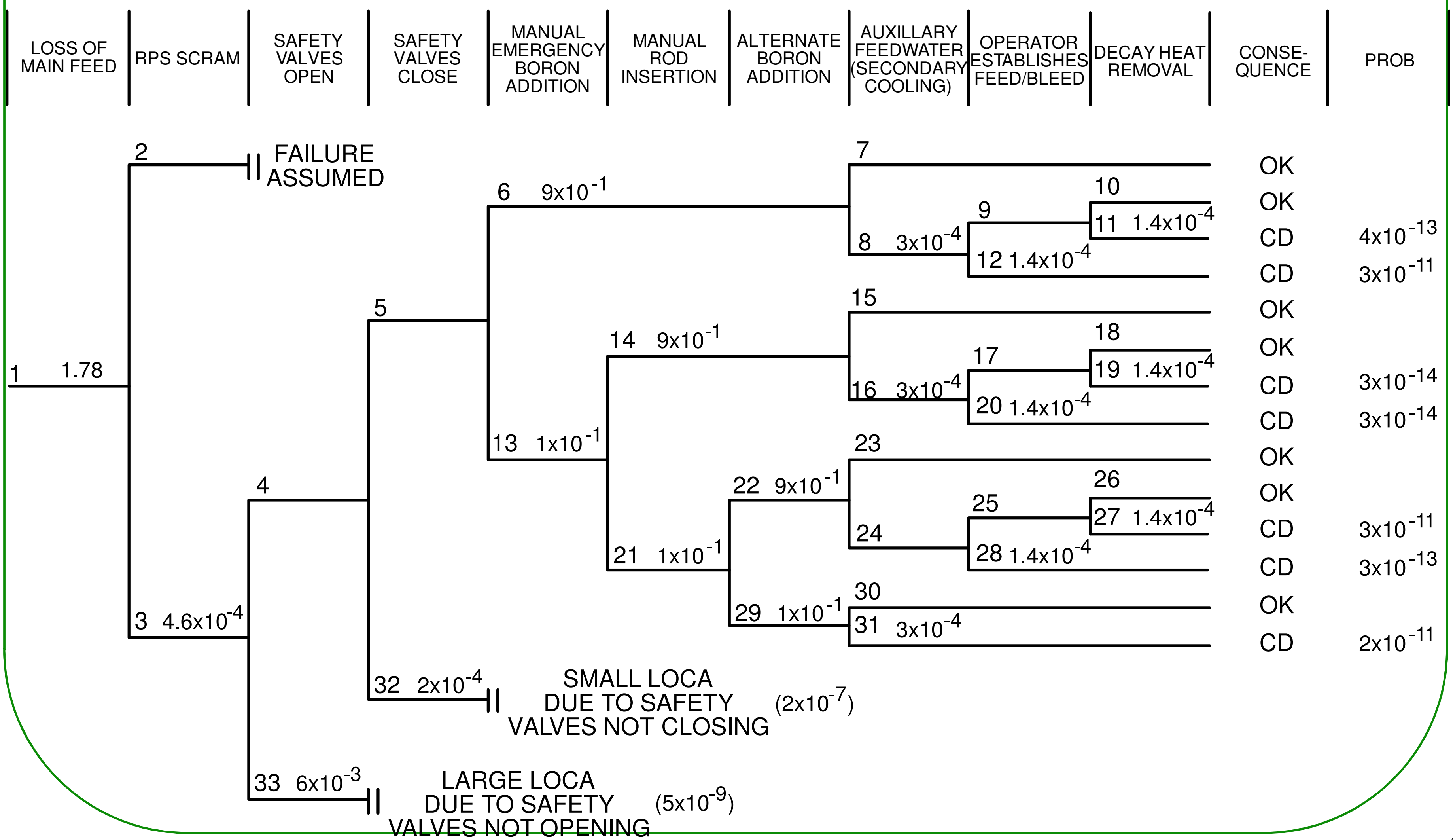


INTEGRATED LEVEL 3 PRA FRAMEWORK

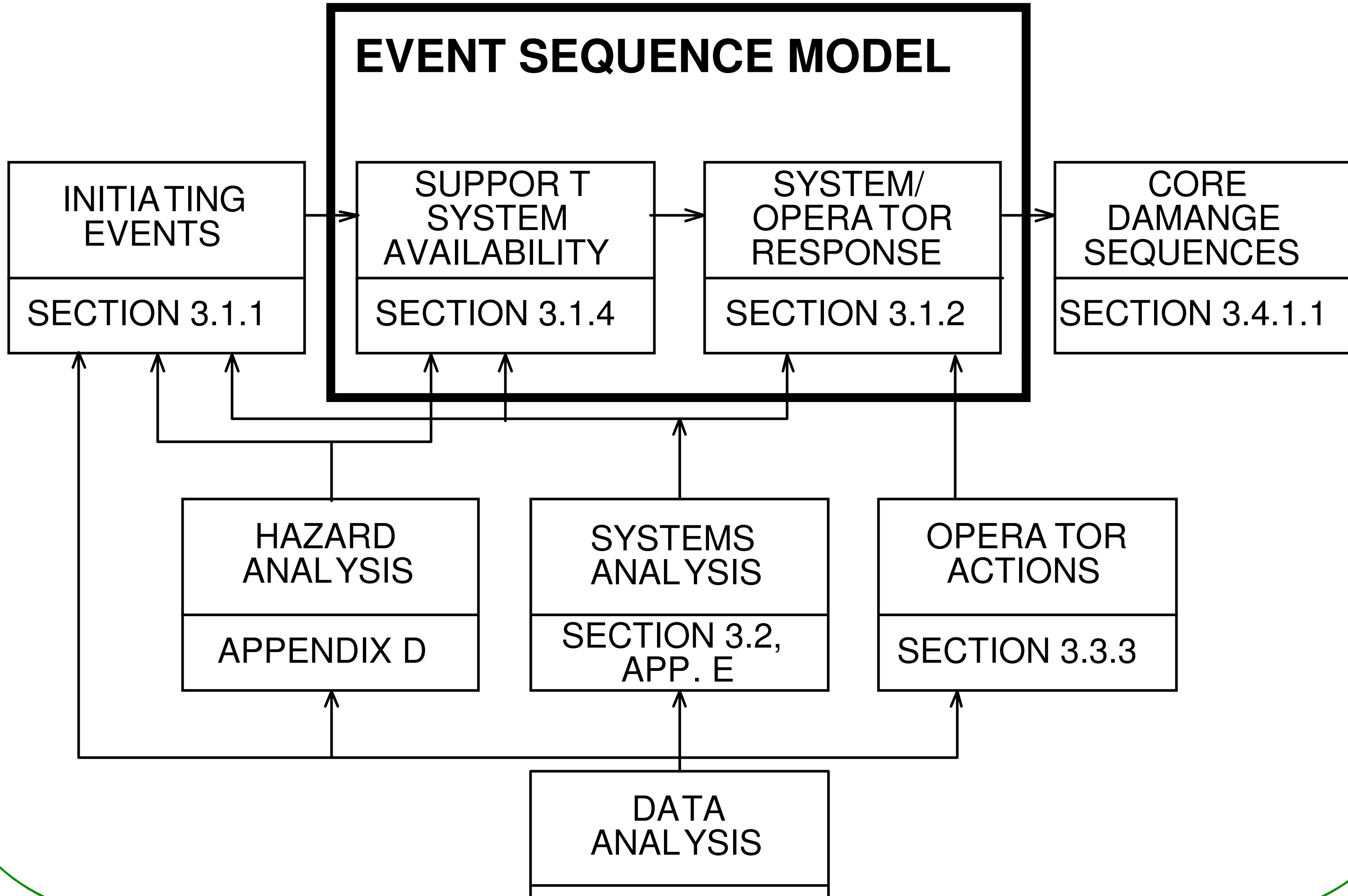


QUANTIFIED ATWS SEQUENCE EVENT TREE

ANTICIPATED TRANSIENT WITHOUT SCRAM

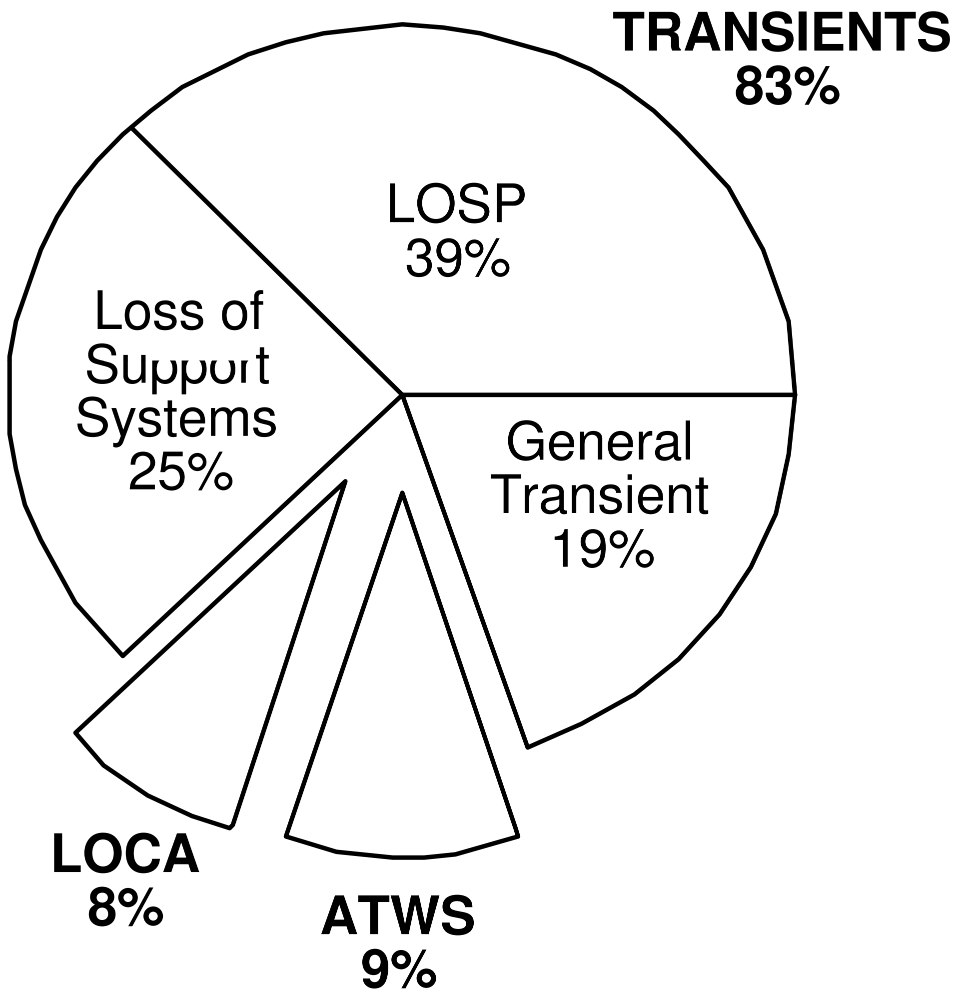


PLANT MODEL OVERVIEW (WITH IPE REPORT SECTION REFERENCES)

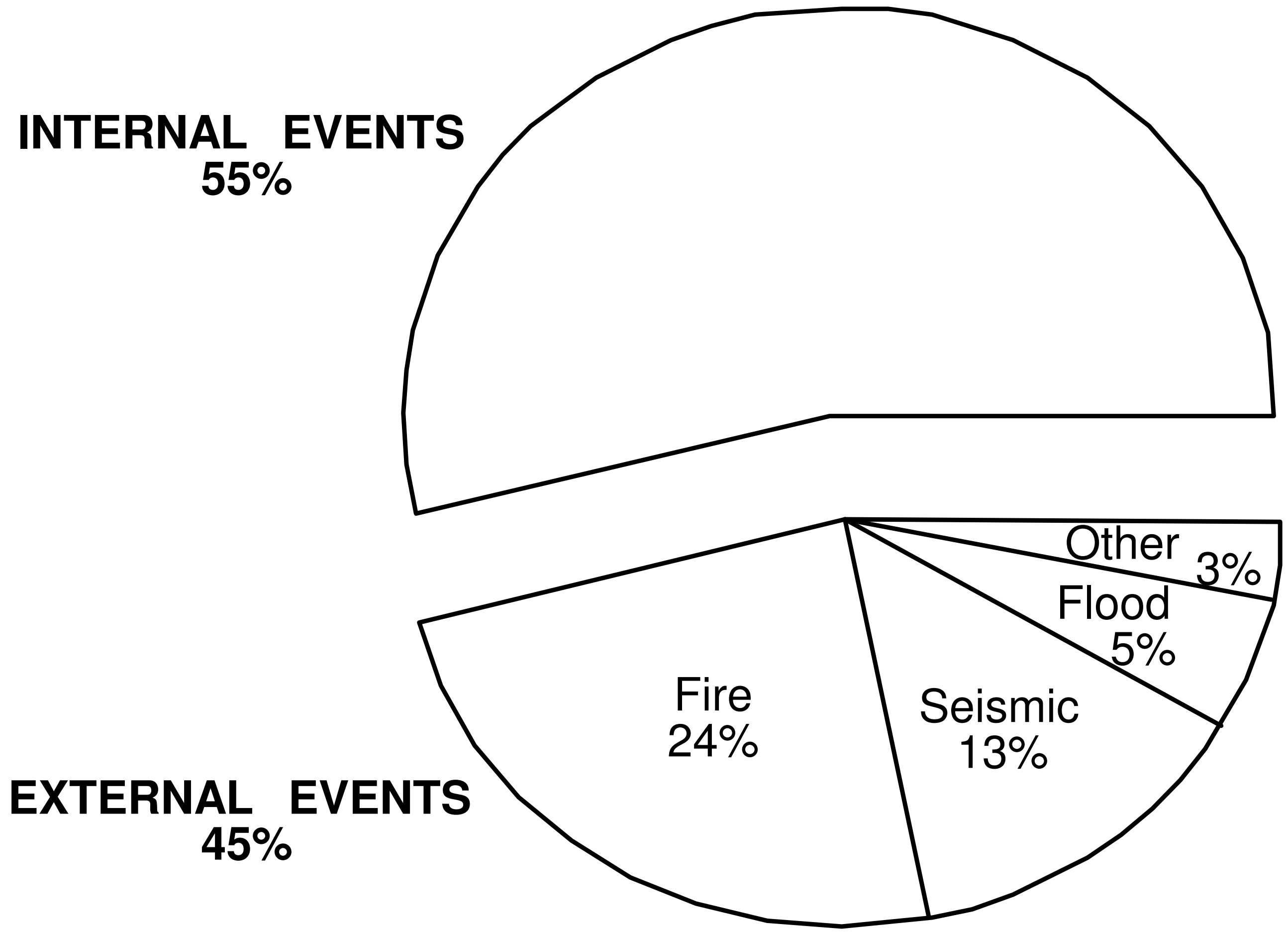


CONTRIBUTIONS TO CORE DAMAGE FREQUENCY

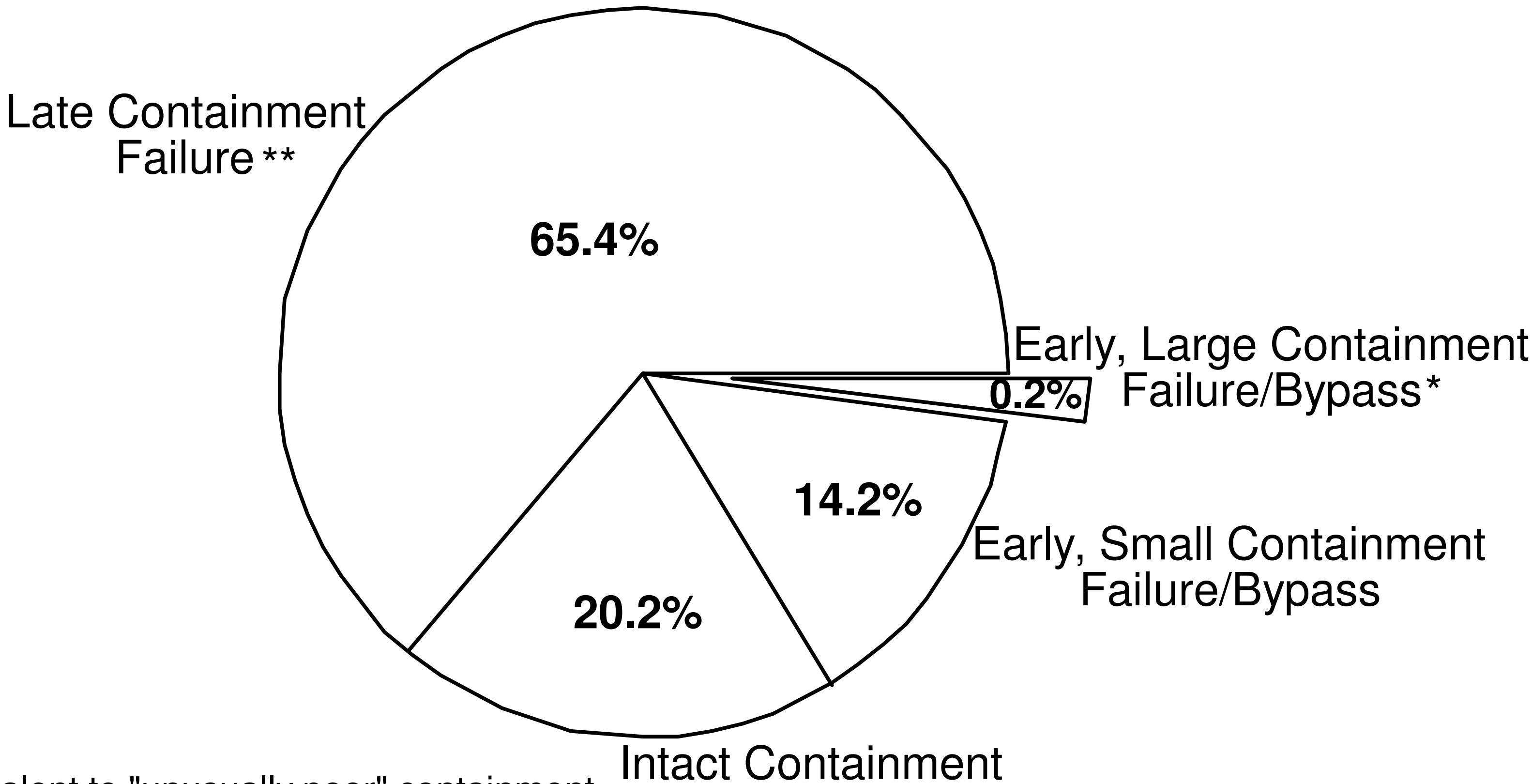
Accidents Grouped by Initiating Event



CONTRIBUTIONS TO CORE DAMAGE FREQUENCY
Accidents Grouped by Internal and External Initiating Event



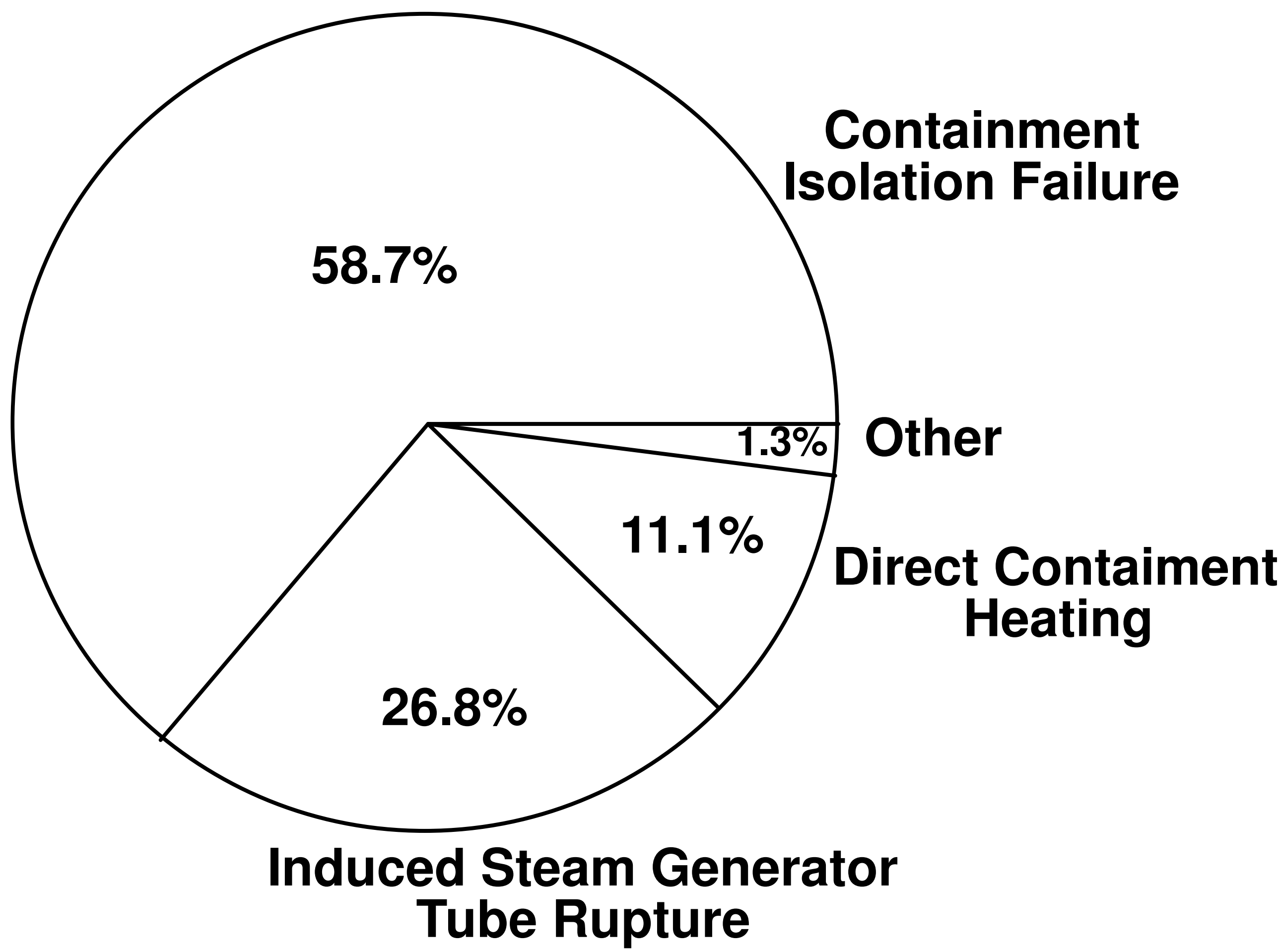
CONTAINMENT PERFORMANCE RESULTS
(Conditional Failure Probability Given Core Damage)



* Equivalent to "unusually poor" containment performance, as defined in GL 88-20

**The containment failure probability of late containment failure is believed to be overestimated relative to containment intact. No credit has been taken for post-core melt recovery actions.

CONTAINMENT FAILURE MODE CONTRIBUTIONS TO EARLY, LARGE CONTAINMENT FAILURES/BYPASS
(“Unusually Poor” Containment Performance)



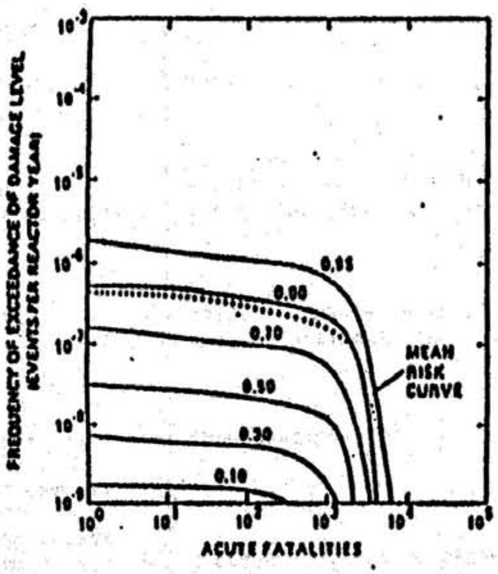


FIGURE 1-1a. RISK OF EARLY FATALITIES

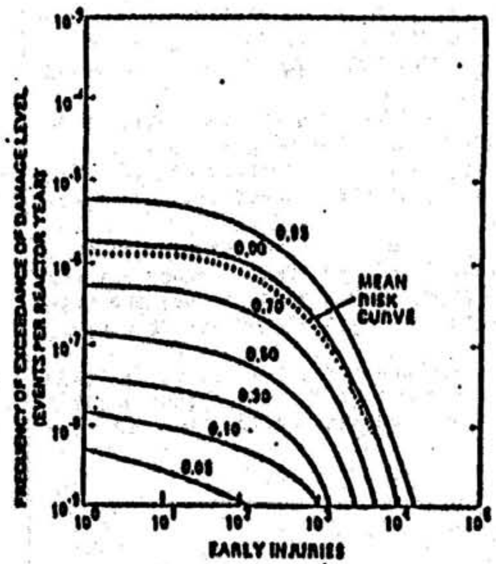


FIGURE 1-1b. RISK OF INJURIES

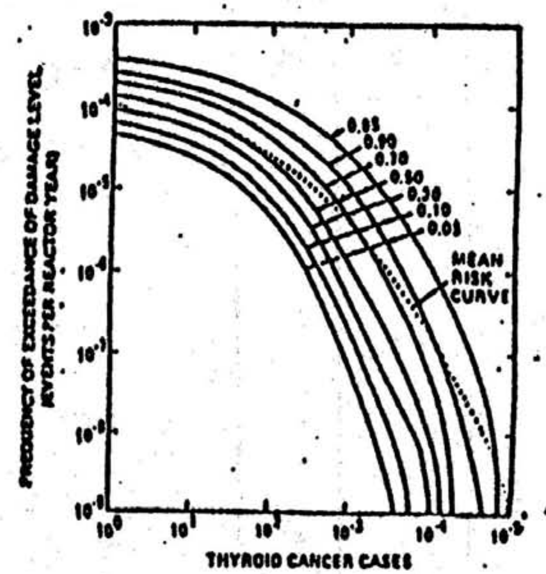


FIGURE 1-1c. RISK OF THYROID CANCER CASES

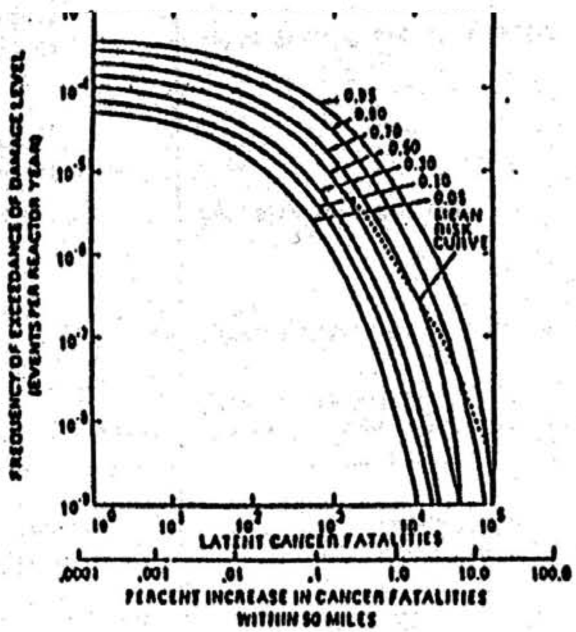


FIGURE 1-1d. RISK OF LATENT CANCER FATALITIES (OTHER THAN FATAL THYROID CANCERS)

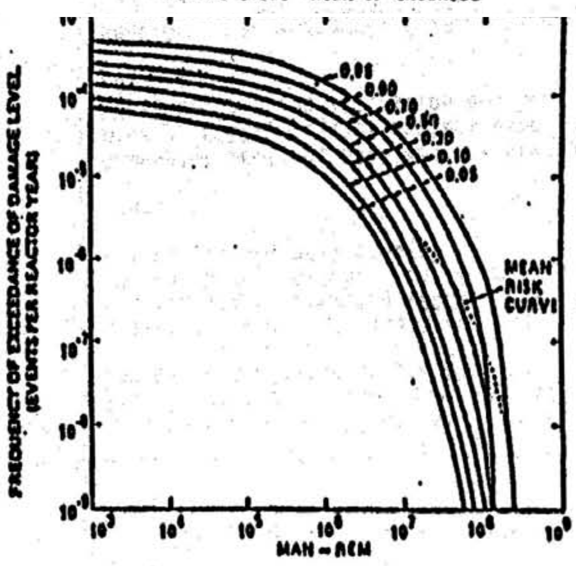


FIGURE 1-1e. RISK OF MAN-REM

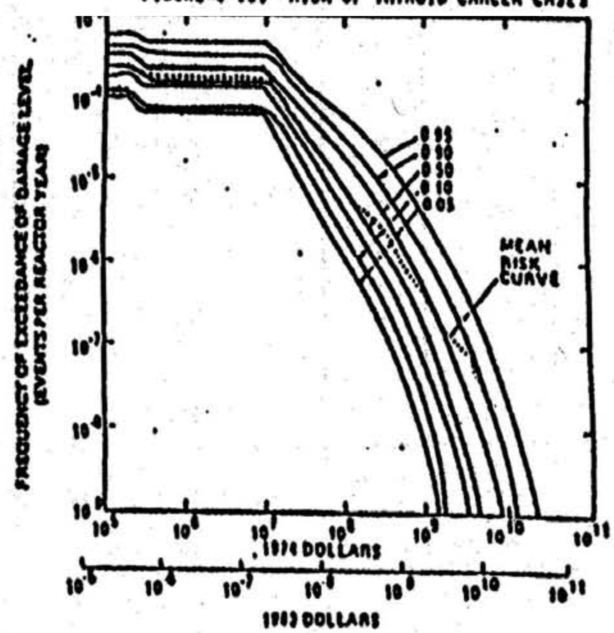


FIGURE 1-1f. RISK OF PROPERTY DAMAGE AND EVACUATION COSTS

QUANTITATIVE SAFETY GOALS OF THE US NUCLEAR REGULATORY COMMISSION (August, 1986)

Early and latent cancer mortality risks to an individual living near the plant should not exceed 0.1 percent of the background accident or cancer mortality risk, approximately
5×10^{-7}/year for early death and
2×10^{-6}/year for death from cancer.

- The prompt fatality goal applies to an average individual living in the region between the site boundary and 1 mile beyond this boundary.
- The latent cancer fatality goal applies to an average individual living in the region between the site boundary and 10 miles beyond this boundary.

SOCIETAL RISKS

- Annual Individual Occupational Risks

All industries	7x10-5
Coal Mining:	24x10-5
Fire Fighting:	40x10-5
Police:	32x10-5
US President:	1,900x10–5 (!)

- Annual Public Risks

Total:	870x10-5
Heart Disease:	271x10-5
All cancers:	200x10-5
Motor vehicles:	15x10-5

From: Wilson & Crouch, *Risk/Benefit Analysis,* Harvard University Press, 2001.

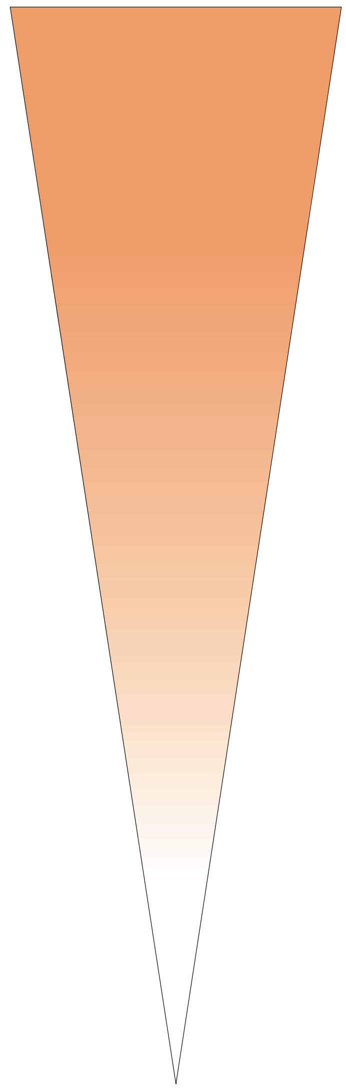
SUBSIDIARY GOALS

- The average core damage frequency (CDF) should be less than 10-4/ry (once every 10,000 reactor years)

- The large early release frequency (LERF) should be less than 10-5/ry (once every 100,000 reactor years)

"ACCEPTABLE" VS. "TOLERABLE" RISKS (UKHSE)

Increasing individual risks and societal concerns



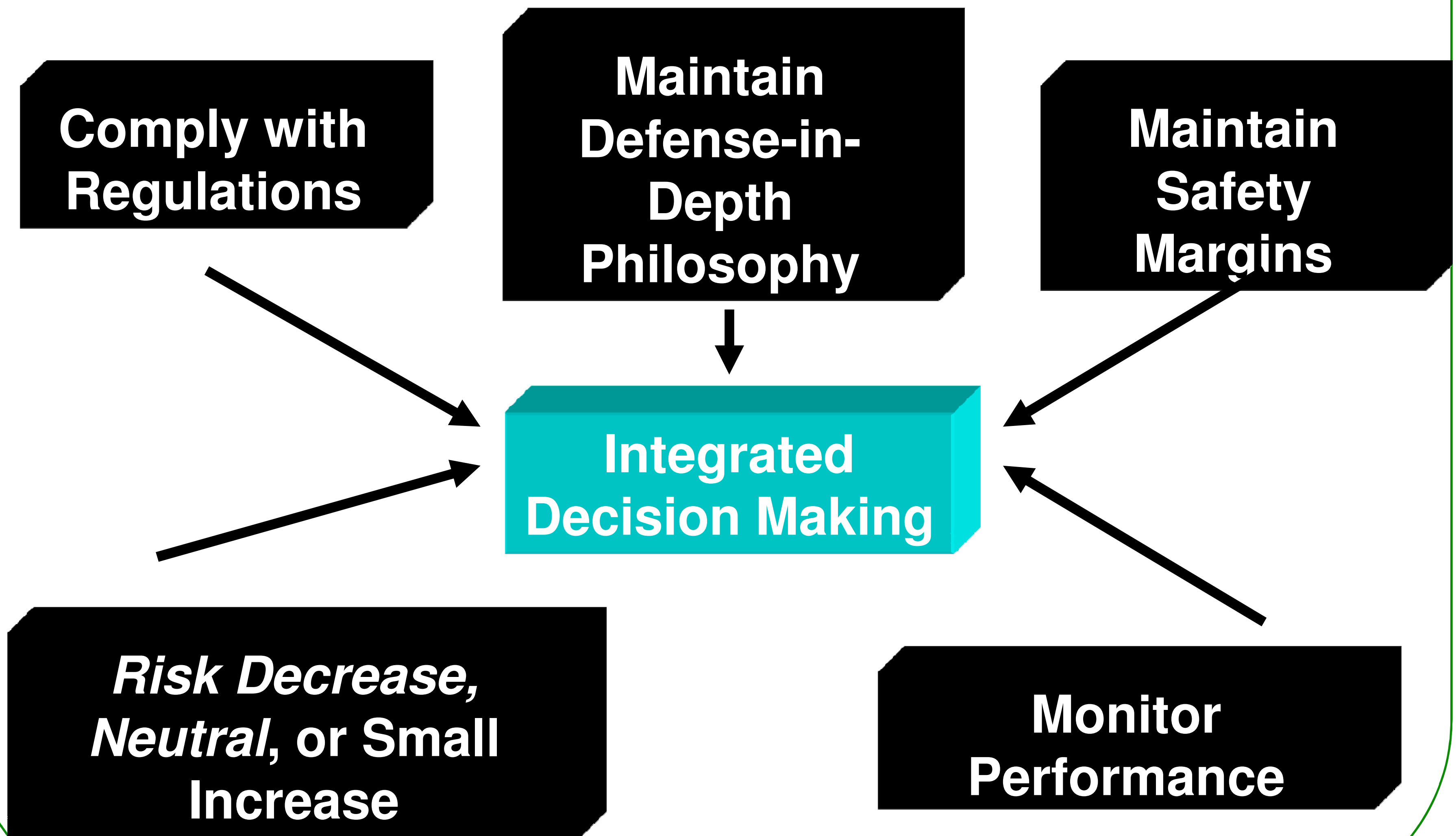
UNACCEPTABLE REGION	Risk cannot be justified save in extraordinary circumstances
TOLERABLE REGION	Control measures must be introduced for risk in this region to drive residual risk towards the broadly acceptable region
BROADLY ACCEPTABLE REGION	Level of residual risk regarded as insignificant -- further effort to reduce risk not likely to be required

Adapted from "The tolerability of risk from nuclear power stations", Health Safety Executive.

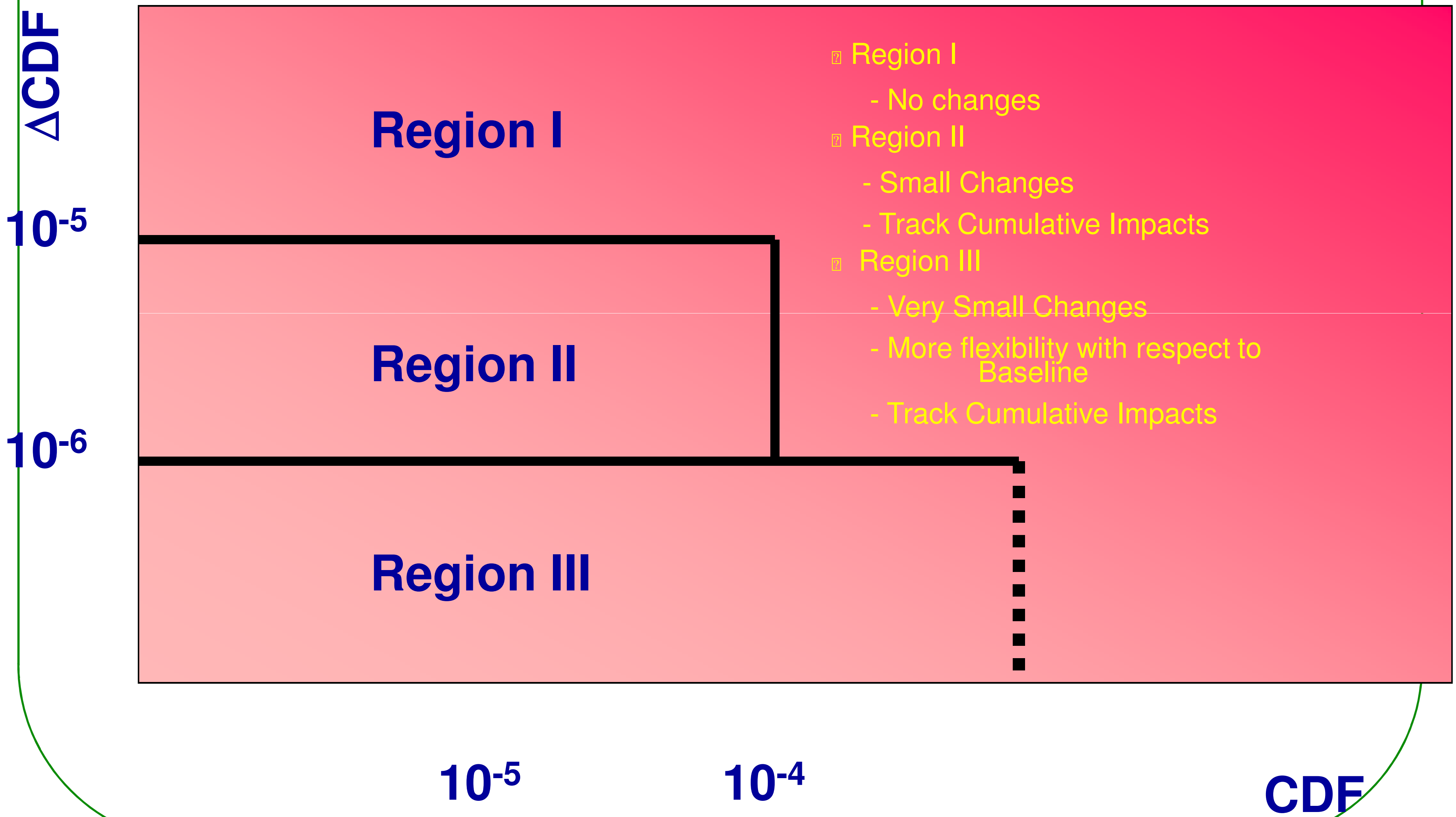
PRA POLICY STATEMENT (1995)

- The use of PRA should be increased to the extent supported by the state of the art and data and in a manner that complements the defense-in-depth philosophy.
- PRA should be used to reduce unnecessary conservatisms associated with current regulatory requirements.

RISK-INFORMED DECISION MAKING FOR LICENSING BASIS CHANGES (RG 1.174, 1998)



ACCEPTANCE GUIDELINES FOR CORE DAMAGE FREQUENCY



RISK-INFORMED FRAMEWORK

Traditional "Deterministic" Approaches

- Unquantified Probabilities
- Design-Basis Accidents
- Structuralist Defense in Depth
- Can impose heavy regulatory burden
- Incomplete

Risk-Informed Approach

- Combination of traditional and risk-based approaches

Risk-Based Approach

- Quantified Probabilities
- Scenario Based
- Realistic
- Rationalist Defense in Depth
- Incomplete
- Quality is an issue

RISK IMPORTANCE MEASURES

$$\text{Risk} = R(q_1, q_2, \ldots, q_n),$$

where

r_i = reliability of the i^{th} plant component, action, or cut set

q_i = unreliability of the i^{th} component = $1 - r_i$

$I_{\text{Fussell-Vesely}_i}$ = the fraction of total risk involving failure of element, i

$$I_{\text{Fussell}-\text{Vesely}_i} = \frac{R(q_i)}{R_{\text{Nom}}} = \frac{R\left(mcs_{i_1} + mcs_{i_2} + \quad \cdots \quad + mcs_{i_m}\right)}{R\left(mcs_1 + \quad \cdots \quad + mcs_n\right)}$$

where

$R(q_i)$ = risk arising from event sequences involving failure of component, action or cut set, i

R_{Nom} = nominal plant risk

m = number of minimal cut sets involving element (basic event) i

n = total number of minimal cut sets

RISK IMPORTANCE MEASURES

Risk Achievement Worth (RAW_i) Maximum relative possible increase in total risk due to failure of element, i; the element is assumed always to fail.

$$RAW_i - \frac{R(q_i = 1)}{R_{Nom}}$$

where

RAW_i = the risk achievement worth of the i^{th} component, action or cut set

COMPONENT RISK IMPORTANCE

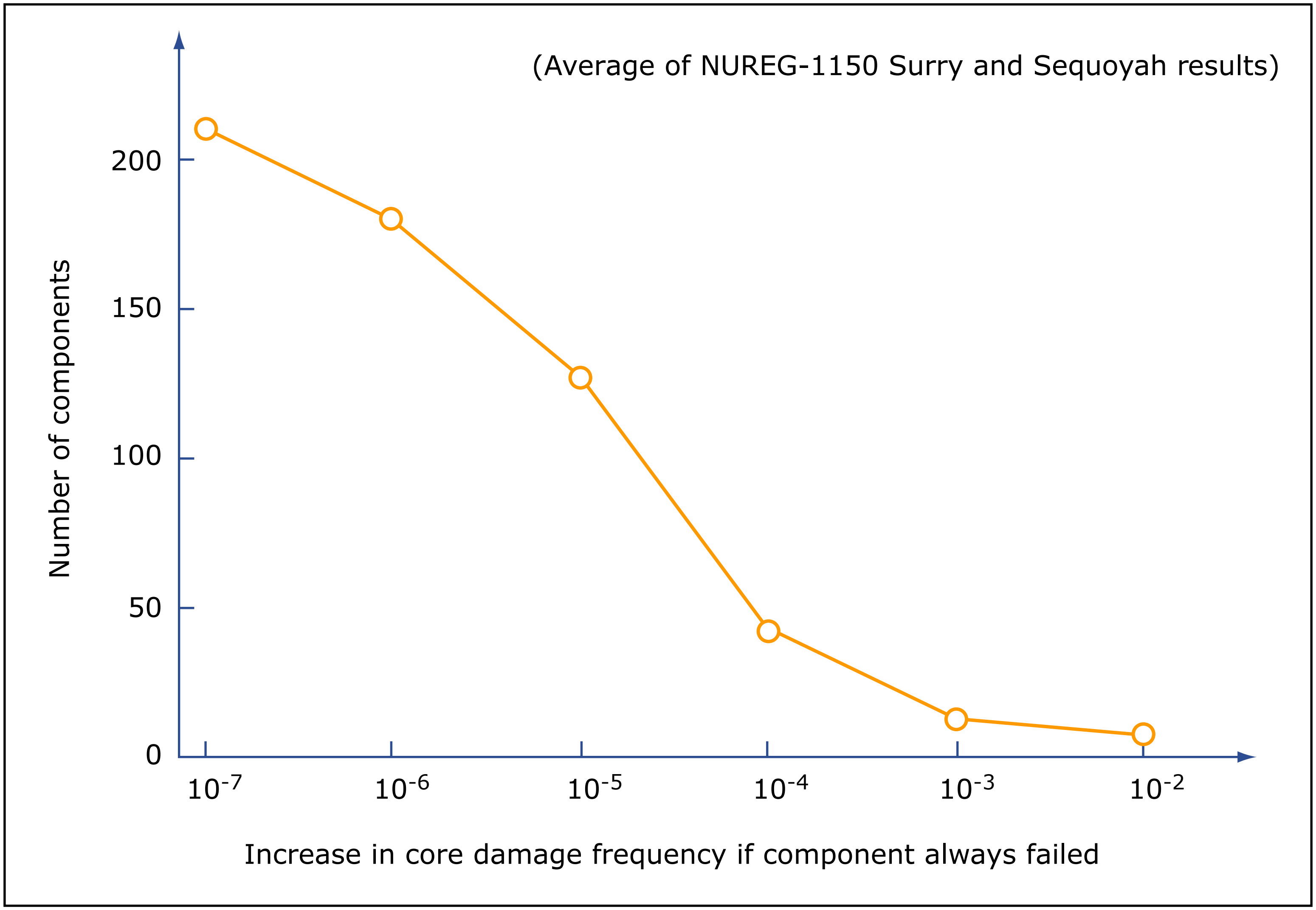


Image by MIT OpenCourseWare. Adapted from F. Gillespie, MIT Reactor Safety Course, 1993.

RISK IMPORTANCE MEASURES

Risk Reduction Worth (RRW_i) = Maximum possible relative reduction in risk due to perfection of event i reliability; the component is assumed always to succeed every time.

$$RRW_i = \frac{R_{Nom}}{R(q_i = 0)},$$

where

RRW_i = the relative risk decrease importance of the i^{th} component, action or cut set

CORE DAMAGE FREQUENCY PERCENT INCREASE PER SYSTEM1

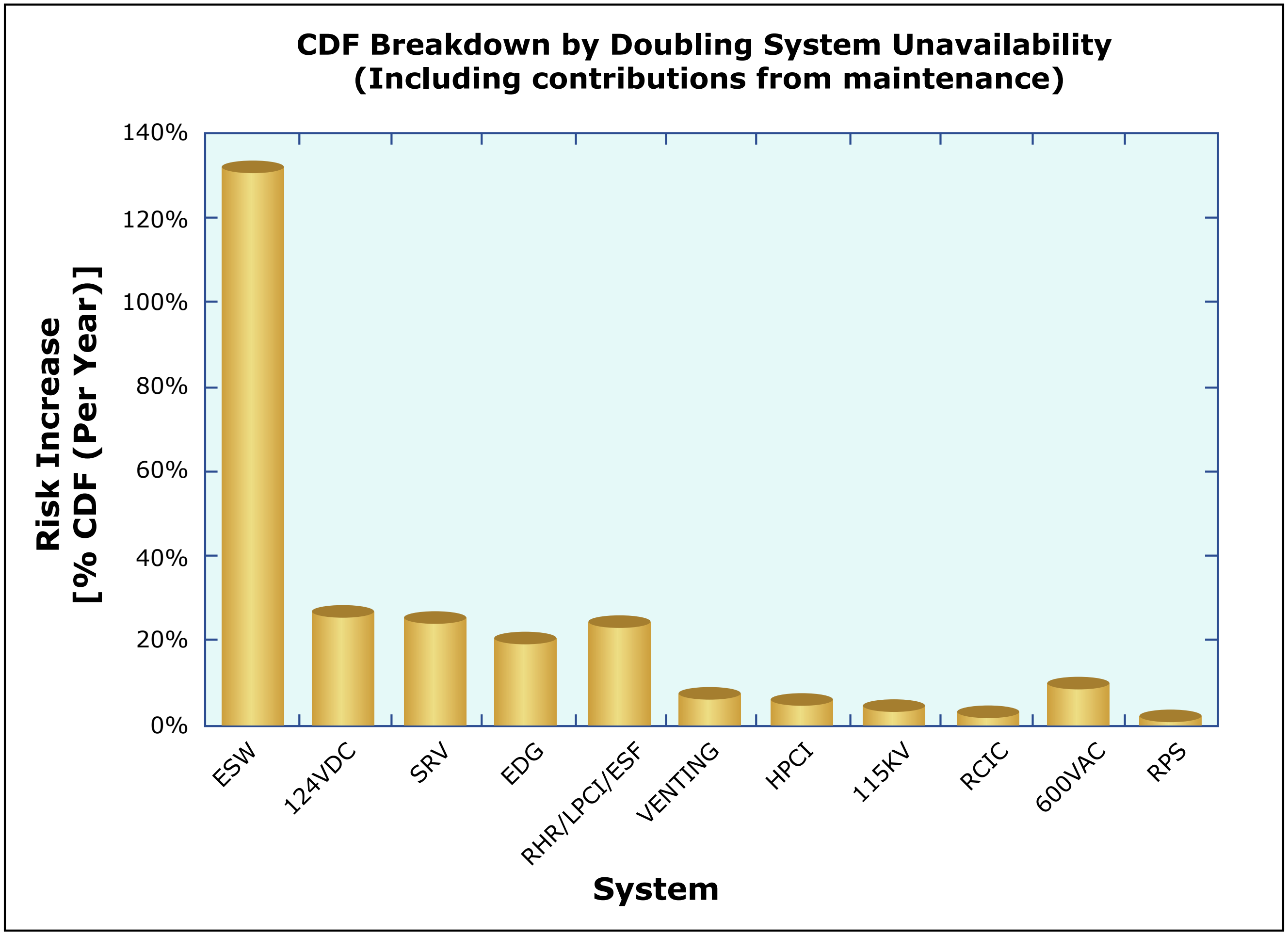


Image by MIT OpenCourseWare.

USES OF RISK IMPORTANCE MEASURES

- Fussell-Vesely
  - Measure a Component’s or System’s Participation in Risks
  - Can Be Used to Identify Which Components or Systems Contribute to Current Risks
- Risk Achievement Worth
  - Identifies Which Components or Systems Must Be Kept Reliable
- Risk Reduction Worth
  - Identifies Which Components or Systems Are Most Valuable for Improvement
  - Note

$$I_{Fussell-Vesely_i} = 1 - \frac{1}{RRW_i}$$

SYSTEM COMPONENT COST AND RELIABILITY DATA

Component	Component Failure Probability
Tank, T-1 or T-2	3.00E-5
Valve, V-1 or V-2	1.20E-4
Pump, P-1 or P-2	9.00E-5
Electric Power, E	1.50E-4
Control System, C	3.00E-4
Cooling System, CO	1.00E-4

SUMMARY OF IMPORTANCE RANKINGS

Component / or System — Importance Measures	Control System, C	Electric Power System, E	Valve, V-1
Fussell-Vesely	0.54	0.27	$5x10^{-5}$
Risk Reduction Worth	2.18	1.37	1.00005
Risk Achievement Worth	1819	1819	1.44

TIMELINE FOR NUCLEAR WASTE DISPOSAL

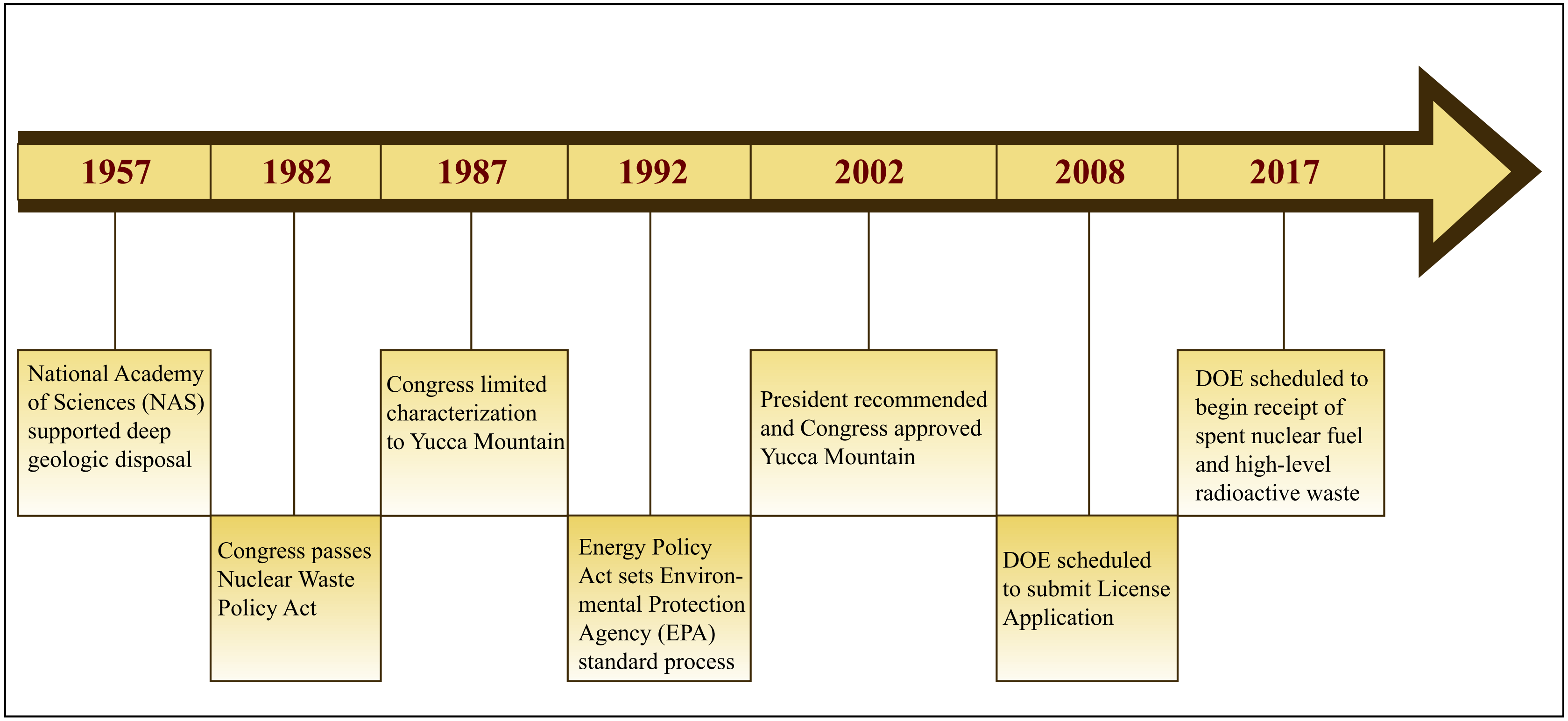
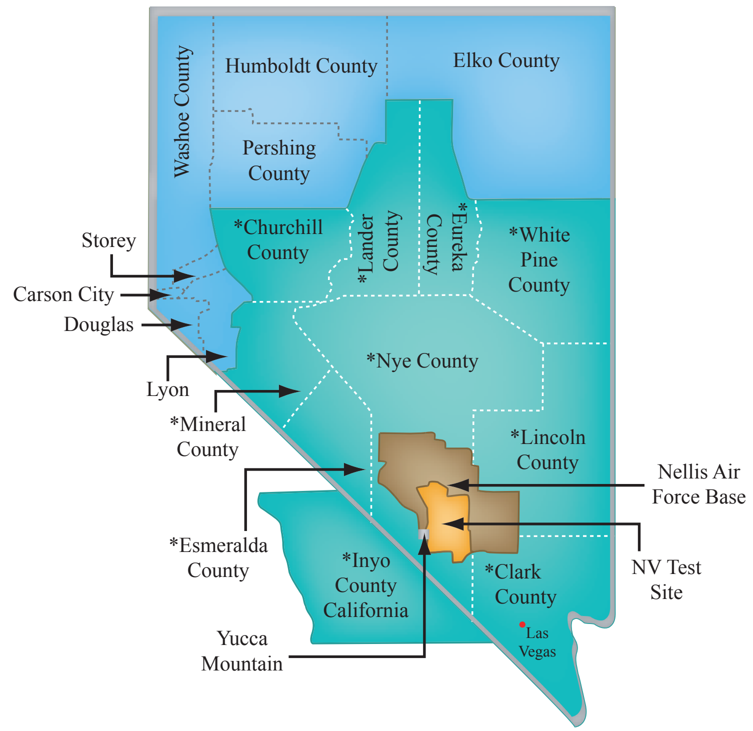


Image by MIT OpenCourseWare.

YUCCA MOUNTAIN, NEVADA



* Counties designated as affected units of local government

- 100 miles northwest of Las Vegas in Nye County
- Located on Western boundary of the Nevada Test Site, a U.S. Department of Energy (DOE) facility

Image by MIT OpenCourseWare.

YUCCA MOUNTAIN SUBSURFACE OVERVIEW

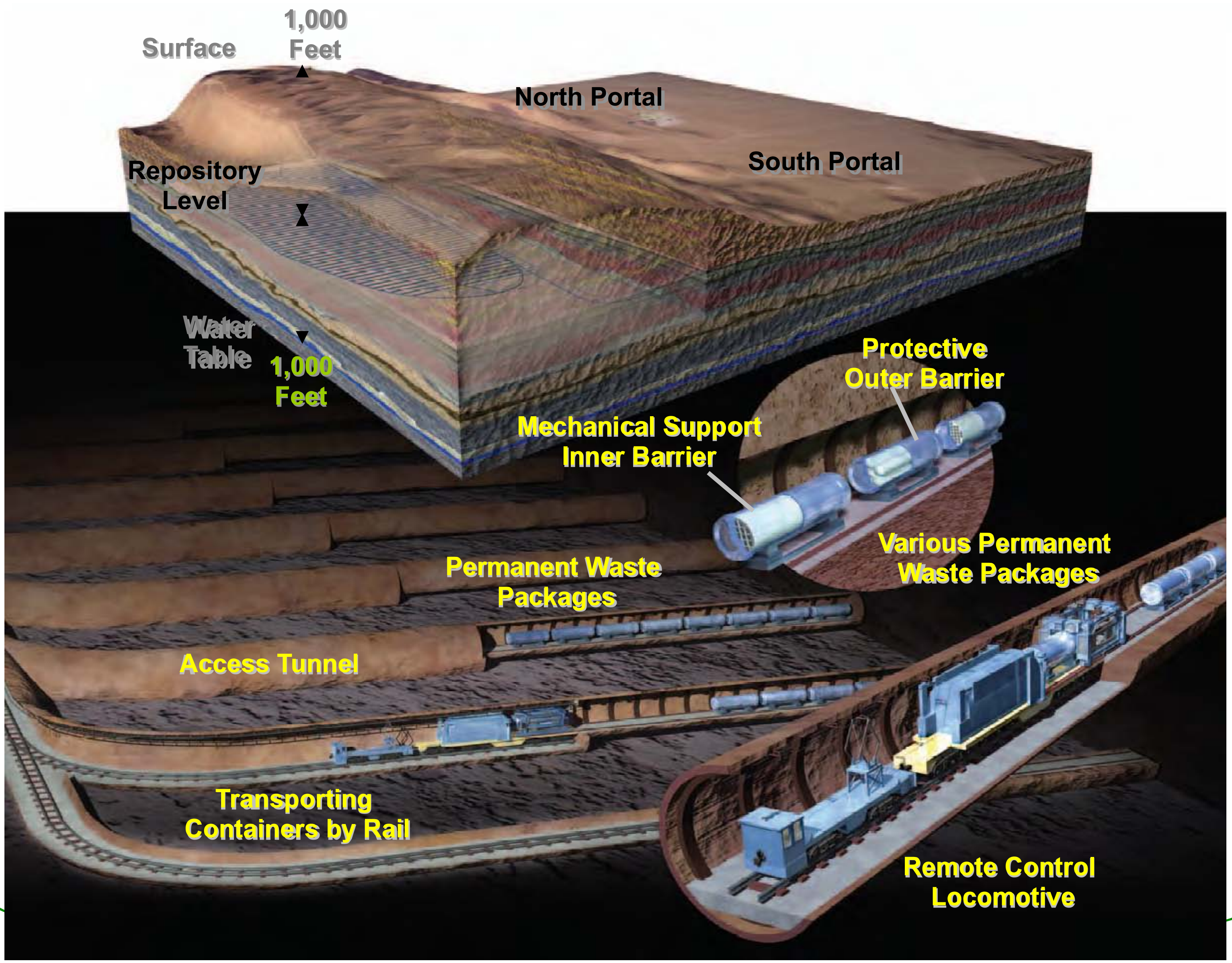
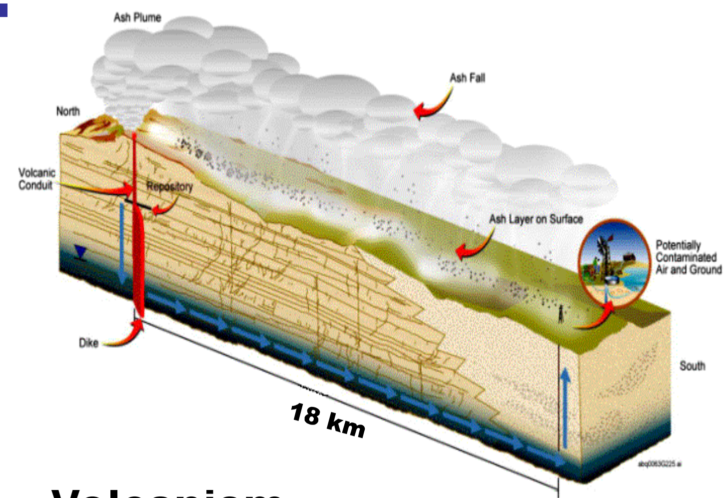
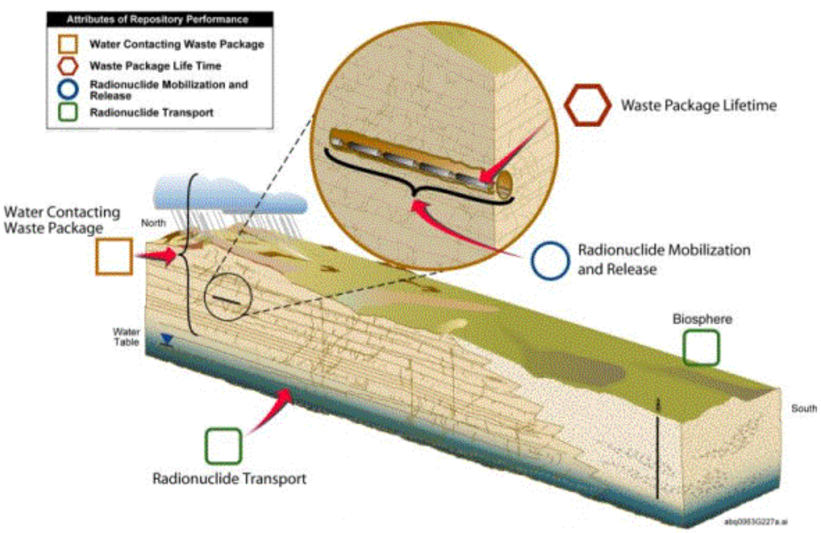


Image by U.S. Office of Civilian Radioactive Waste Management.

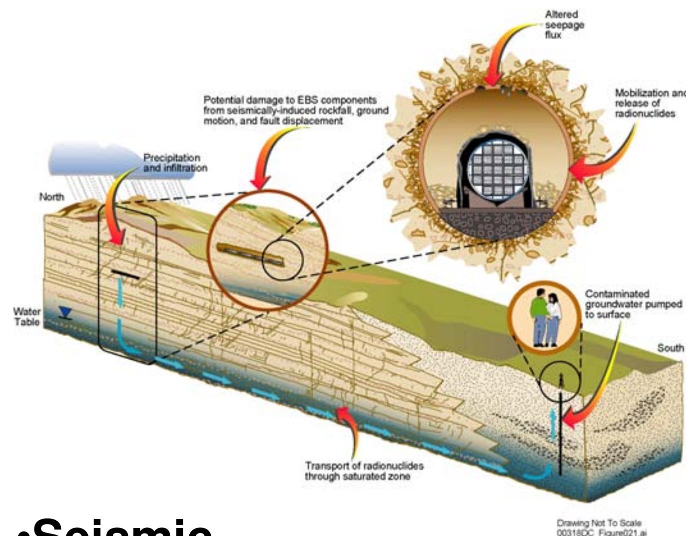
HYPOTHETICAL SCENARIOS



- **Volcanism**



- **Nominal**
- **Early defects**



- **Seismic**

Source: U.S. Department of Energy.

YUCCA MOUNTAIN: PREDICTED AVERAGE ANNUAL DOSE FOR 10,000 YEARS

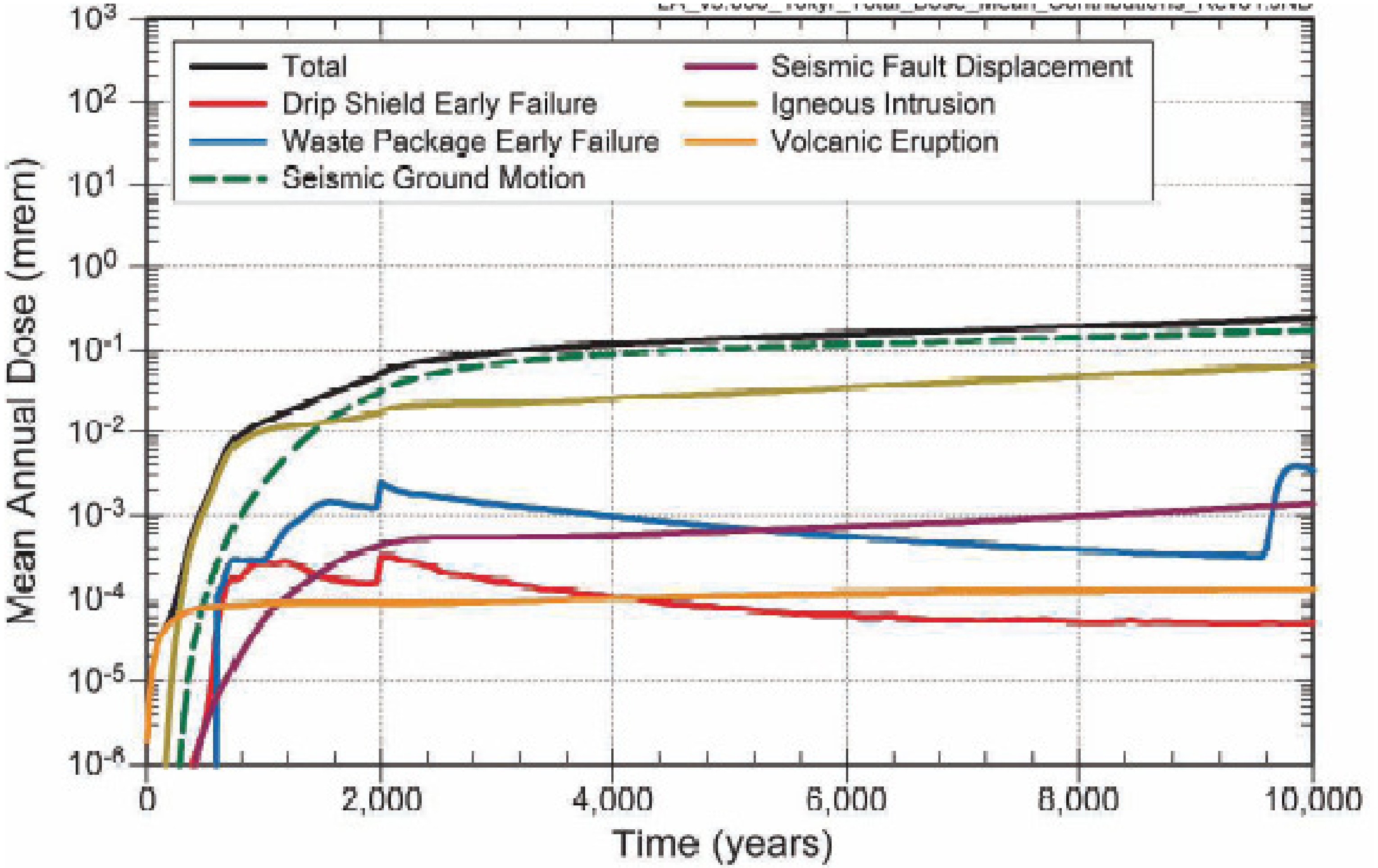


Fig. F-17 in *Draft Supplemental Environmental Impact Statement for a Geologic Repository at Yucca Mountain*. U.S. Department of Energy, October 2007, DOE/EIS-0250F-S1D.

YUCCA MOUNTAIN: PREDICTED MEDIAN ANNUAL DOSE FOR 1,000,000 YEARS

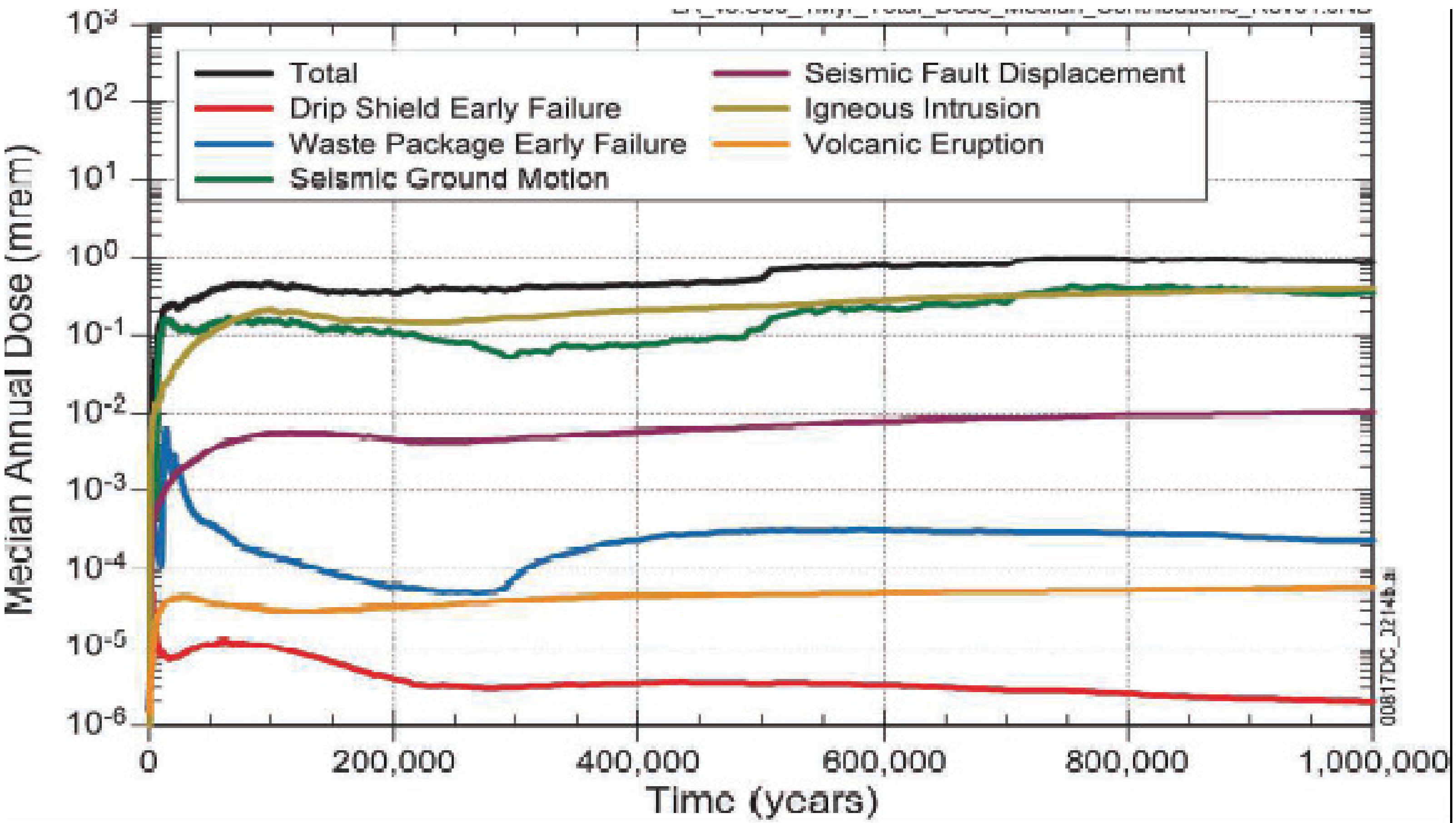


Fig. F-17 in *Draft Supplemental Environmental Impact Statement for a Geologic Repository at Yucca Mountain*. U.S. Department of Energy, October 2007, DOE/EIS-0250F-S1D.

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