Operational Reactor Safety

22.091/22.903

Professor Andrew C. Kadak
Professor of the Practice

Probabilistic Safety Analysis
Lecture 11

Topics to be Covered

- Probabilistic Basics
- Event Trees
- Fault Trees
- Applications
- Examples
- Safety Goals
- Uses

Deterministic Safety Analysis

- Chapter 15 Analyses and Regulations Require
  - Design Basis Accident Analysis
  - Establishes strict criteria for assumptions at most conservative conditions
  - Assumes single failure criteria (worst)
  - Assumes other systems function normally
  - Most restrictive is Appendix K - LOCA criteria
  - Defines safety grade components that must work

Probabilistic Safety Analysis

- Models entire plant and all systems using best estimate analysis
- Nothing is assumed to work - Probabilities of failure of components assigned
- Includes human error
- Detailed analysis of consequences of failure required to determine the conditional consequences of failure of other components

PWR Engineered Safety Systems

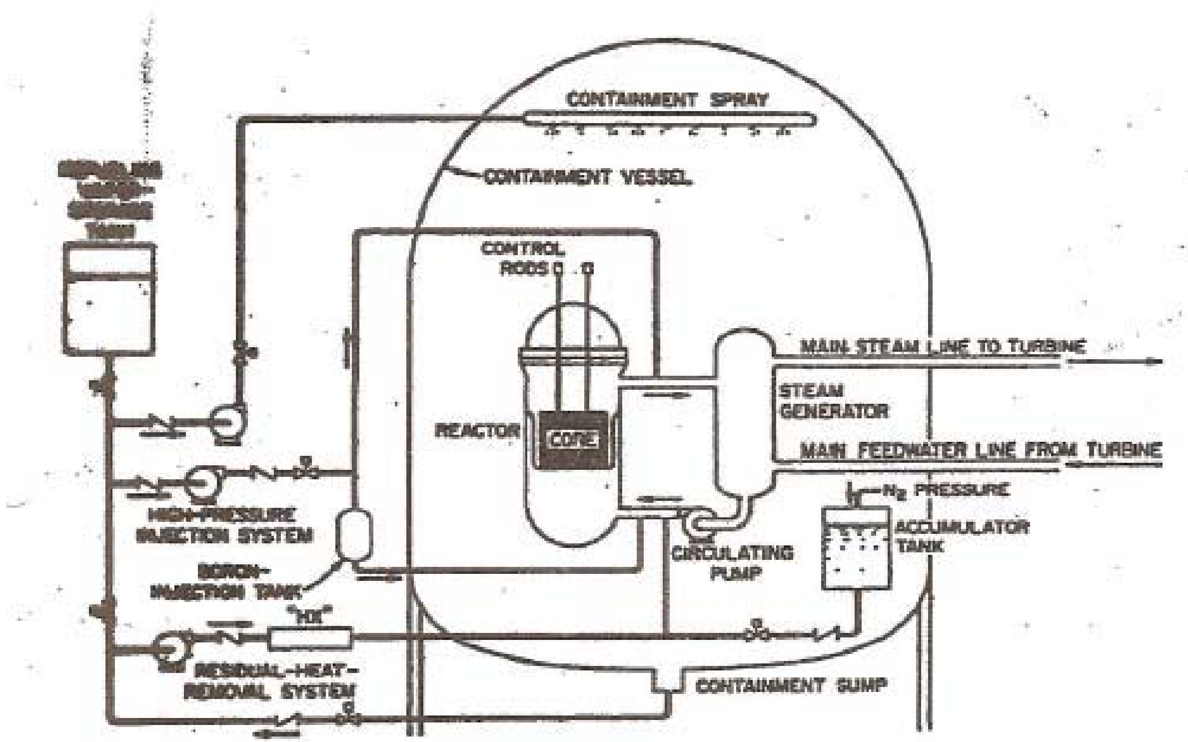


FIGURE 14-2
Engineered safety systems for a PWR. (From W. B. Cottrell, "The ECCS Rule-Making Hearing," *Nuclear Safety*, vol. 15, no. 1, Jan.–Feb. 1974.)

BWR Early Engineered Safety Systems

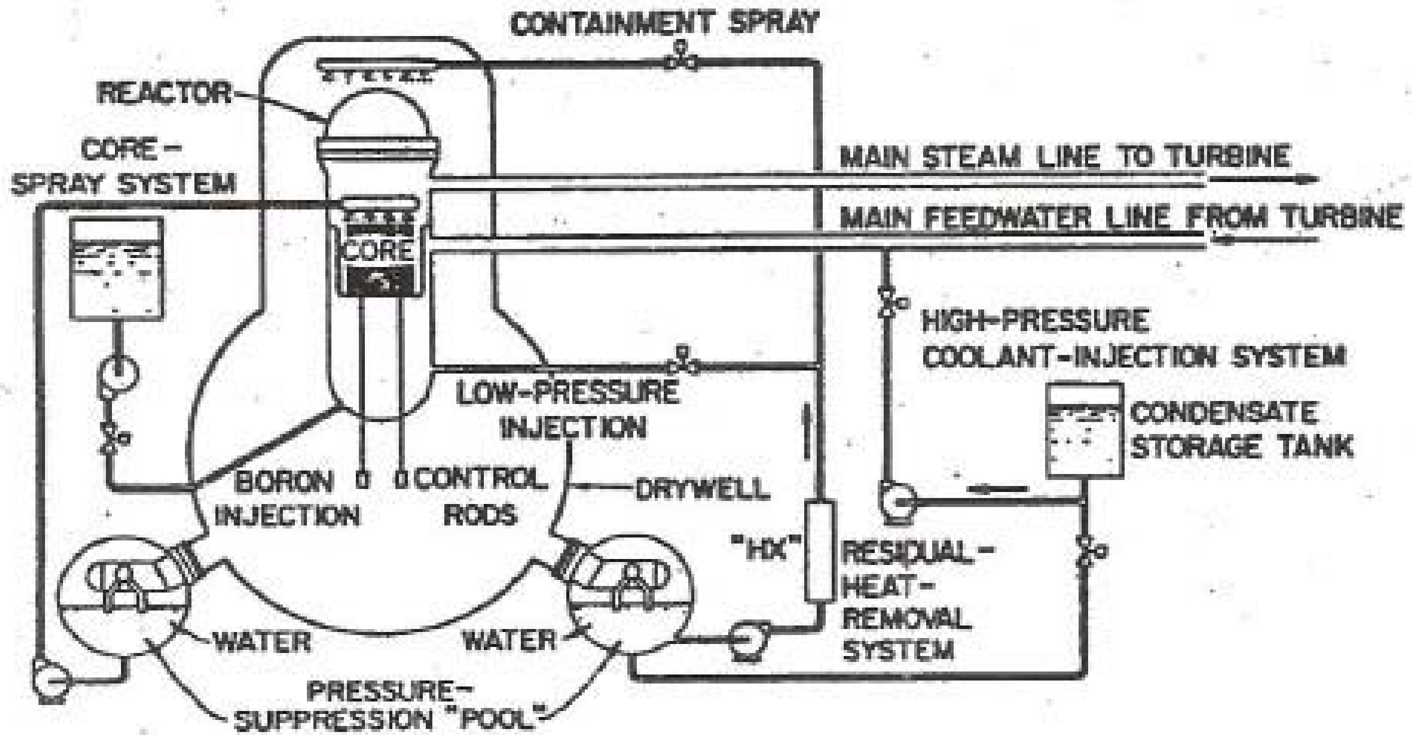


FIGURE 14-6
Engineered safety systems for an early BWR. (From W. B. Cottrell, "The ECCS Rule-Making Hearing," *Nuclear Safety*, vol. 15, no. 1, Jan.–Feb. 1974.)

PSA Applications

- Risk (Based) or Informed regulation
  - an informed combination of deterministic and probabilistic analysis with judgement
- Safety Goals - How safe is safe enough
- Individual licensing decisions to assess marginal impact of plant changes.
- Performance based regulation.

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.

Potential Offsite Doses

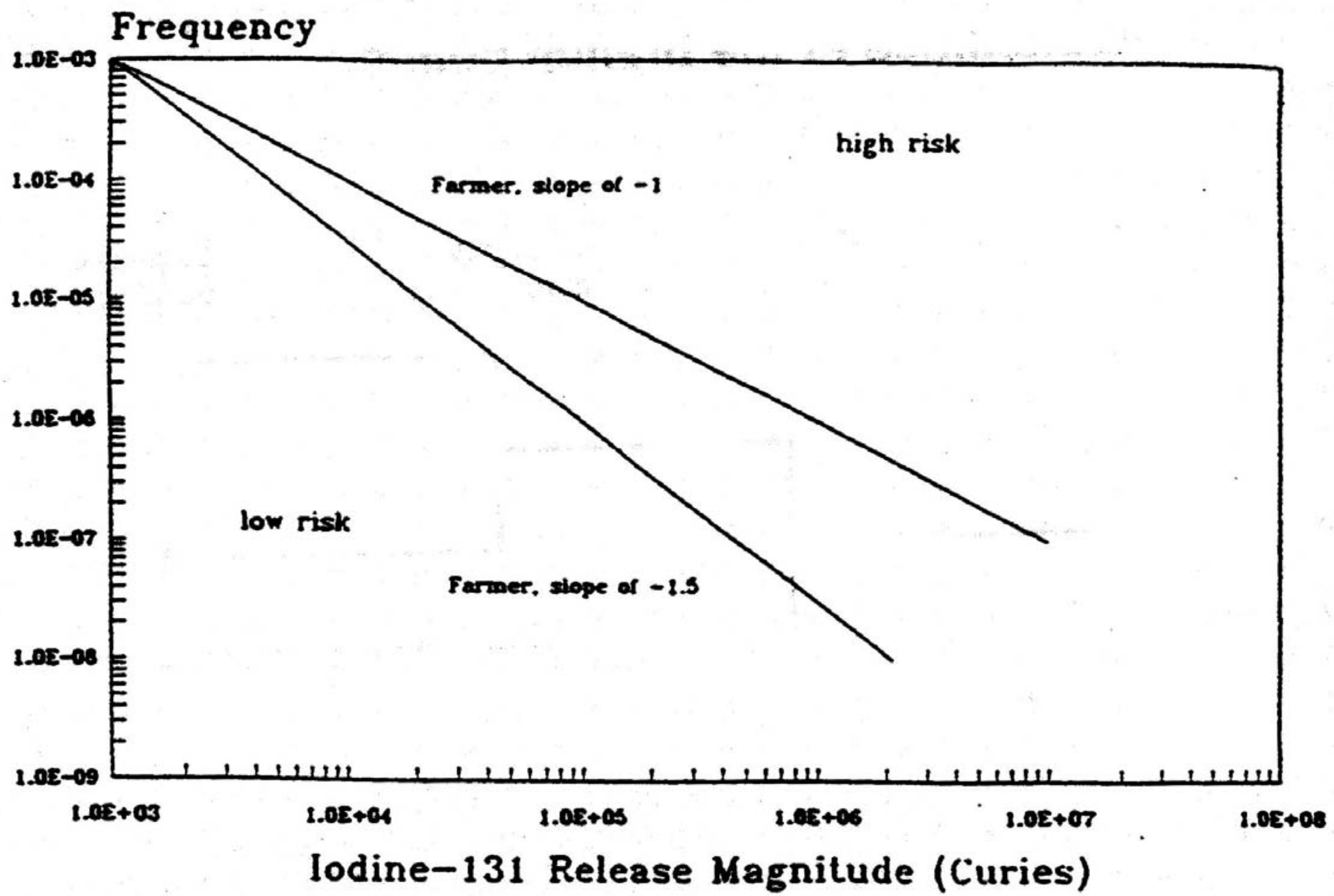
POTENTIAL OFFSITE DOSES DUE TO DESIGN-BASIS ACCIDENTS
(CONSERVATIVE CASE)

Accident	Two Hour Exclusion Boundary (3200 feet or 975 meters)		Duration of Accident Low Population Zone (4 miles or 6.4 km)	
	Thyroid (Rem)	Whole Body (Rem)	Thyroid (Rem)	Whole Body (Rem)
Loss of Coolant	155	3	81	3
Control Rod Ejection	<1	<1	<1	<1
Fuel Handling	2	2	<1	<1
Steam Line Break	16	1	3	1
10 CFR 100 Dose Guideline	300	25	300	25

Farmer's Paper (1967)

- Iodine-131 is a major threat to health in a nuclear plant accident.
- Attempting to differentiate between credible (DBAs) and incredible accidents (Class 9; multiple protective system failures) is not logical.
- If one considers a fault, such as a loss-of-coolant accident (LOCA), one can determine various outcomes, from safe shutdown and cooldown, to consideration of delays and partial failures of shutdown or shutdown cooling with potential consequences of radioactivity release.

THE FARMER LINE



Historical Risk Studies

1. Farmer's Paper (1967) call for new approach to reactor safety
2. Reactor Safety Study (1975) Wash-1400
3. German Risk Study (1979)
4. Risk Assessment Review Group Report (1979)
5. Zion and Indian Point PRAs (1981)
   ↳ near Chicago ↳ near NYC
6. NUREG - 1150 (1989) gov't study on severe accidents
7. Individual Plant Examinations

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?

Reactor Safety Study (WASH-1400; 1975)

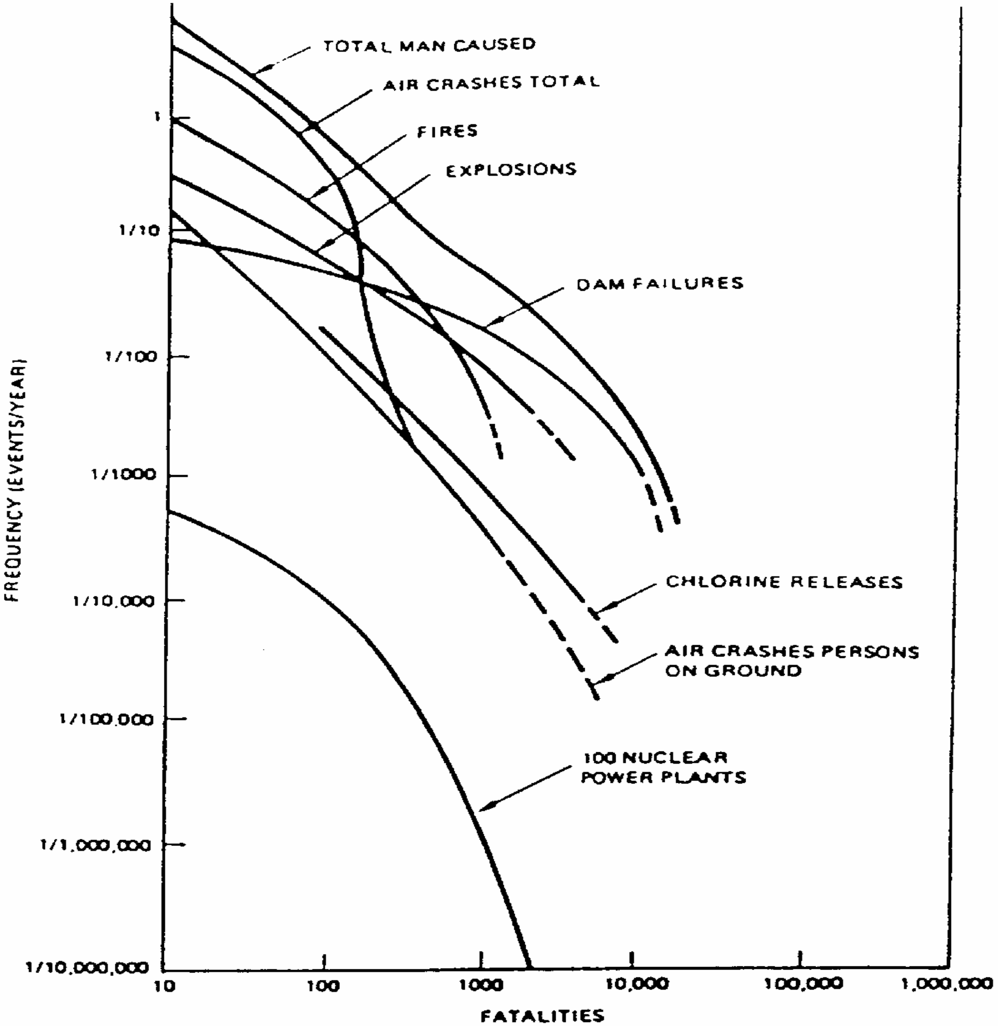
Prior Beliefs:

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)

CRITICAL SAFETY FUNCTIONS

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

The Single-Failure Criterion

- "Fluid and electric systems are considered to be designed against an assumed single failure if neither (1) a single failure of any active component (assuming passive components function properly) nor (2) a single failure of a passive component (assuming active components function properly), results in a loss of the capability of the system to perform its safety functions."
- The intent is to achieve high reliability (probability of success) without quantifying it.
- Looking for the worst possible single failure leads to better system understanding.

Defense in Depth

"Defense-in-Depth is an element of the Nuclear Regulatory Commission'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, USNRC, 1999]

PRA Model Overview

Level I	Level II	Level III
PLANT MODEL	**CONTAINMENT MODEL**	**SITE/CONSEQUENCE MODEL**
Results Accident sequences leading to plant damage states	Results Containment failure/release sequences	Results Public health effects

PLANT MODE
At-power Operation
Shutdown / Transition Evolutions

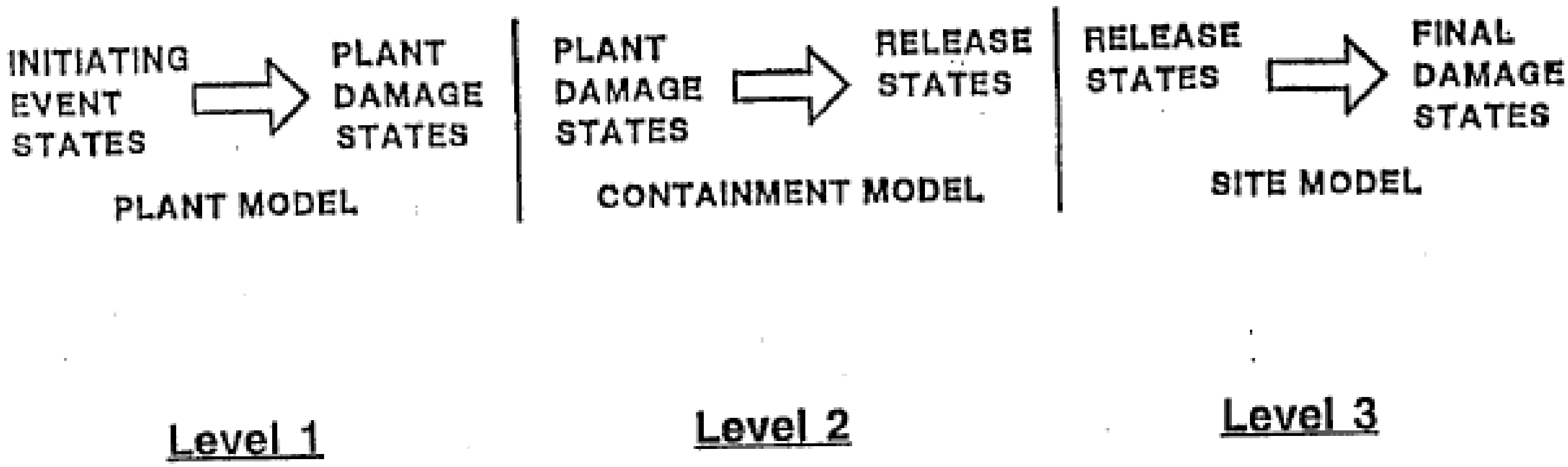
SCOPE
Internal Events
External Events

Uncertainties

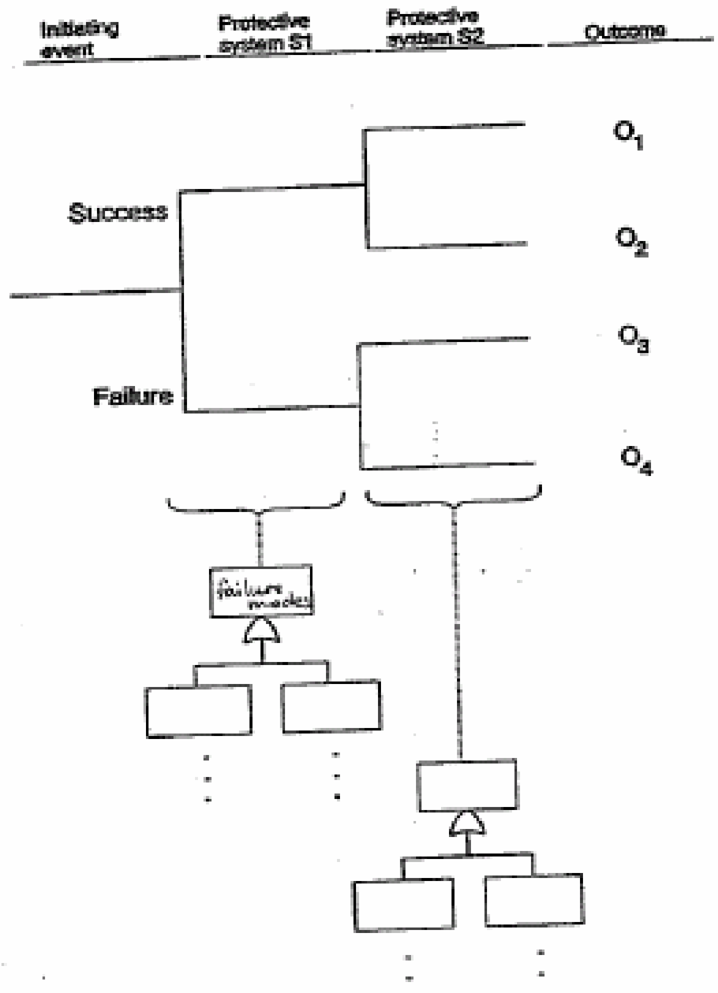
Basic Elements of PSA

- Probability
- Combinatorial Events and Expectations
- Event Trees
- Fault Trees
- Risk
- Data
- Uncertainties
- Nuclear Power Plant PRA Structure
- Typical Results

Transition of a Risk Assessment



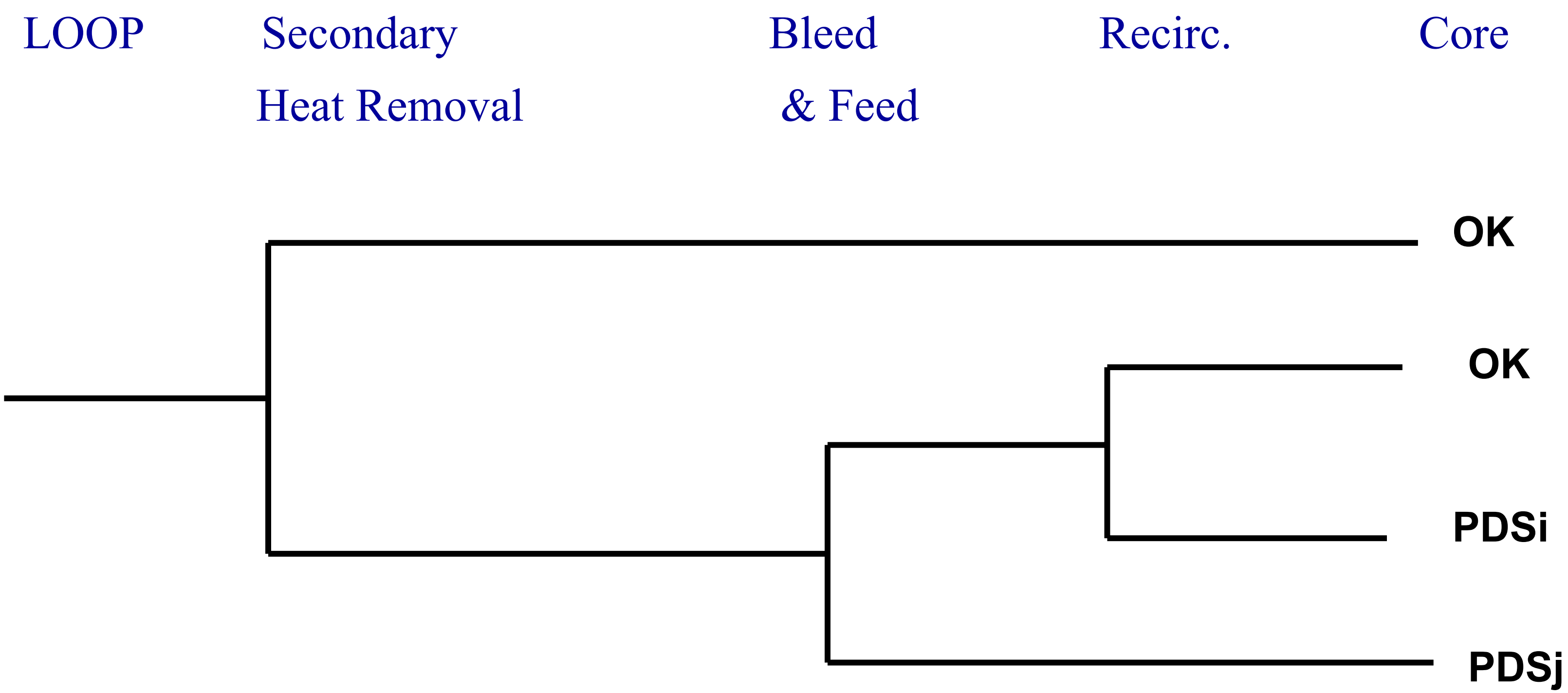
Event and Fault Tree Structure



Example of event tree analysis with fault trees

Massachusett
Department of Nucle
adak, 2008

Loss-of-offsite-power event tree



CDF and LERF Definitions

- Core damage frequency is defined as the sum of the frequencies of those accidents that result in uncovery and heatup of the reactor core to the point at which prolonged oxidation and severe fuel damage involving a large fraction of the core (i.e., sufficient, if released from containment, to have the potential for causing offsite health effects) is anticipated.

- Large early release frequency is defined as the frequency of those accidents leading to significant, unmitigated releases from containment in a time frame prior to effective evacuation of the close-in population such that there is the potential for early health effects. Such accidents generally include unscrubbed releases associated with early containment failure shortly after vessel breach, containment bypass events, and loss of containment isolation.

Draft Regulatory Guide 1.200 Rev. 1, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities"

At Power Level I Results

CDF = 4.5×10^{-5} / yr (Modes 1, 2, 3)

Initiator Contribution to CDF Total:

- Internal Events……………………..56%

- External Events …………………….44%

Seismic Events	24%
Fires	18%
Other	2%

From: K. Kiper, MIT Lecture, 2006

Level I Results

- Functional Sequences

Contribution	CDF
– Transients - Station Blackout/Seal LOCA	45%
– Transients - Loss of Support Systems/Seal LOCA	29%
– Transients - Loss of Feedwater/Feed & Bleed	12%
– LOCA - Injection/Recirculation Failure	7%
– ATWS - No Long Term Reactivity Control	6%
– ATWS - Reactor Vessel Overpressurization	2%

From: K. Kiper, MIT Lecture, 2006

At Power Level II Results

Release Categories	Conditional Probability
Large-Early	0.002
Small-Early	0.090
Large-Late	0.249
Intact	0.659

Large-Early Release Freq (LERF) = 7×10^{-8} / yr

Large-Early Failure Mode	Percent Contribution
Containment Bypass	82%
Containment Isolation Failure	18%
Gross Containment Failure	0.1%

From: K. Kiper, MIT Lecture, 2006

SHUTDOWN

Shutdown, Full Scope, Level 3 PSA (1988)

Results: Mean $CDF_{shutdown}$ ~ Mean CDF_{power}

- Dominant CD sequence:
  Loss of RHR at reduced inventory.
- Risk dominated by operator actions - causing <u>and</u> mitigating events.
- Significant risk reductions with low-cost modifications and controls.

  Midloop level monitor, alarm

  Procedures, training

  Administrative controls on outage planning

From: K. Kiper, MIT Lecture, 2006

Shutdown PRA Issues

- Risk is dominated by operator actions - importance of HRA.
- Generic studies give useful insights, but risk-controlling factors are plant-specific.
- Shutdown risk is dynamic - average risk is generally low (relative to full power risk), but is subject to risk "spikes."
- Shutdown risk is more amenable to "management." At-power risk is designed in.

From: K. Kiper, MIT Lecture, 2006

Integrated Risk (All Modes) – 2002 Update

Mode	Description	CDF	Percent of Total
Mode 1	Full-power (>70% pwr)	4.28 E-5	63%
Mode 2	Low-power (<70% pwr)	0.15 E-5	2%
Mode 3	Hot Standby	0.08 E-5	1%
Mode 4	Hot Shutdown	0.05 E-5	1%
Mode 5	Cold Shutdown	0.91 E-5	13%
Mode 6	Refueling	1.38 E-5	20%
Total Core Damage Frequency		6.86E-5	100%

From: K. Kiper, MIT Lecture, 2006

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."

Zion and Indian Point PRAs (1981)

- First PRAs sponsored by the industry.
- Comprehensive analysis of uncertainties (Bayesian methods).
- Detailed containment analysis (not all accidents lead to containment failure).
- "External" events (earthquakes, fires) may be significant contributors to risk.

SEABROOK STATION RISK RESULTS

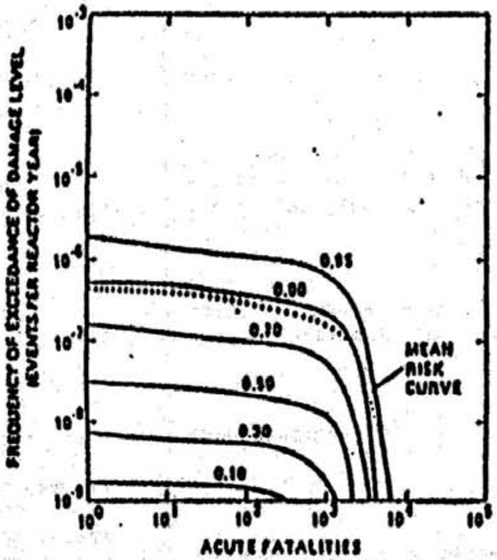


FIGURE 1-1a. RISK OF EARLY FATALITIES

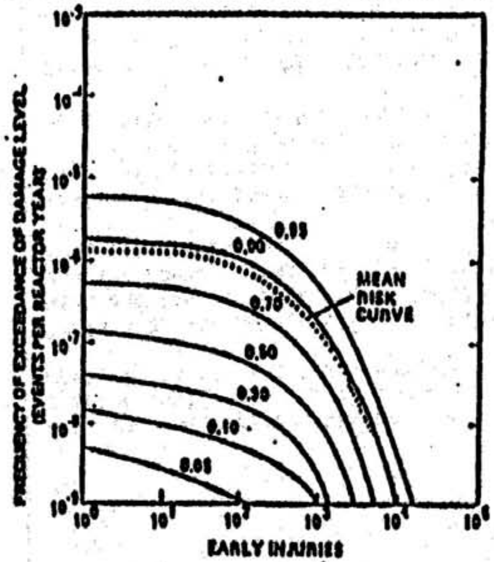


FIGURE 1-1b. RISK OF INJURIES

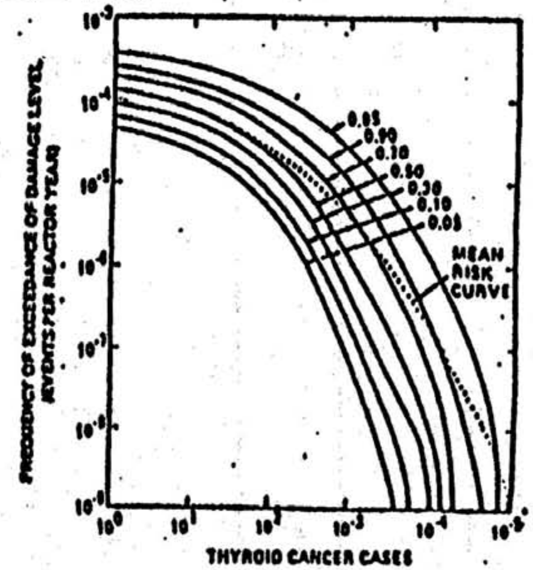


FIGURE 1-1c. RISK OF THYROID CANCER CASES

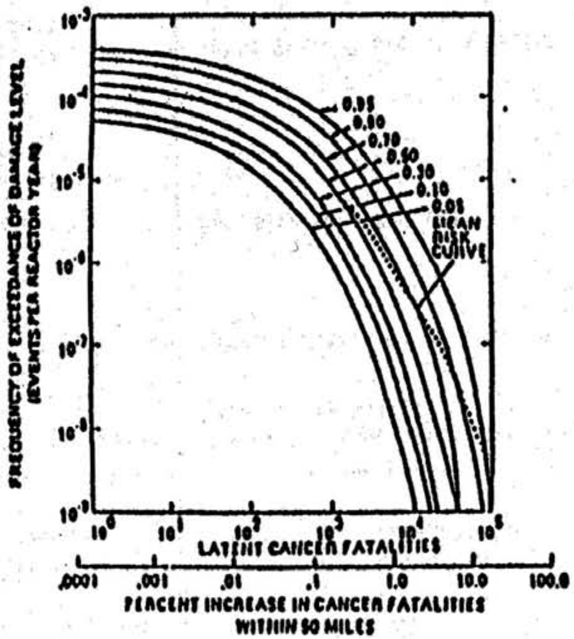


FIGURE 1-1d. RISK OF LATENT CANCER FATALITIES (OTHER THAN FATAL THYROID CANCERS)

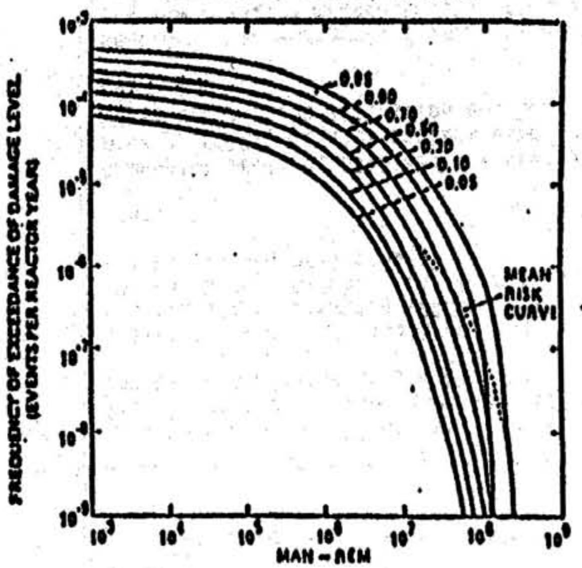


FIGURE 1-1e. RISK OF MAN-REM

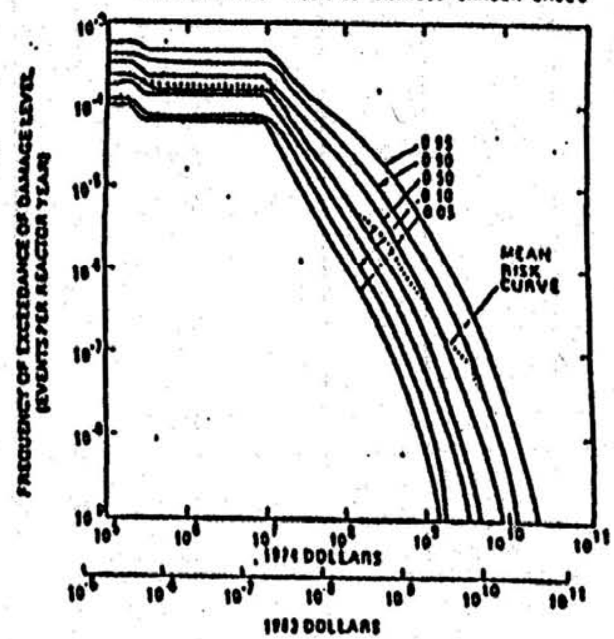


FIGURE 1-1f. RISK OF PROPERTY DAMAGE AND EVACUATION COSTS

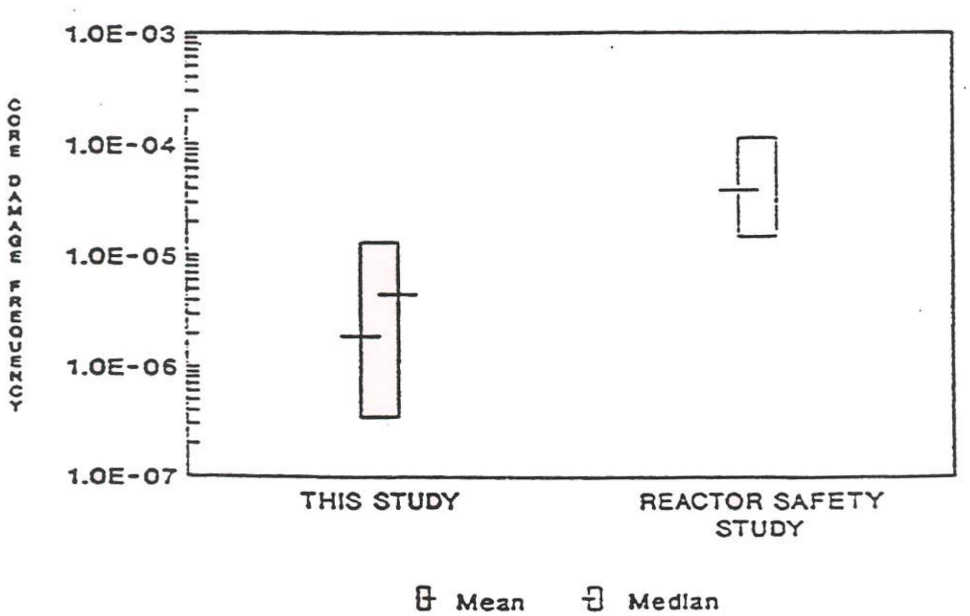
SUMMARY OF ACCIDENT SEQUENCES WITH SIGNIFICANT RISK AND CORE MELT FREQUENCY CONTRIBUTIONS

Initiating Event	Additional System Failures/ Human Actions	Resulting Dependent Failures	Sequence Frequency (per reactor year)	Sequence Ranking		
				Core Melt	Latent Health Risk	Early Health Risk
Loss of Offsite Power	Onsite AC Power, No Recovery of AC Power Before Core Damage	Component cooling, high pressure makeup (ECCS), reactor coolant pump seal LOCA, containment filtration and heat removal.	3.3-5	1	1	*
Loss of Offsite Power	Service Water, No Recovery of Offsite Power	Onsite AC power, component cooling, high and low pressure makeup (ECCS), reactor coolant pump seal LOCA, containment filtration and heat removal.	9.2-6	2	2	*
Small LOCA	Residual Heat Removal	None.	8.9-6	3	*	*
Control Room Fire	None	Component cooling, high and low pressure makeup (ECCS), reactor coolant pump seal LOCA, containment filtration and heat removal.	8.7-6	4	3	*
Loss of Main Feedwater	Solid State Protection System	Reactor trip, emergency feedwater, high and low pressure makeup (ECCS), containment filtration and heat removal.	8.3-6	5	4	*
Steam Line Break Inside Containment Heat Removal	Operator Failure to Establish Long Term		5.6-6	6	*	*
Reactor trip	Component Cooling	High and low pressure makeup (ECCS), reactor coolant pump seal LOCA, containment filtration and heat removal.	4.6-6	7	5	*
Loss of Offsite Power	Train A Onsite Power, Train B Service Water, No Recovery of AC Power Before Core Damage	Train B onsite power, component cooling, high and low pressure makeup (ECCS), reactor coolant pump seal LOCA, containment filtration and heat removal.	4.4-6	8	6	*
Loss of Offsite Power	Train B Onsite Power, Train A Service Water, No Recovery of AC Power Before Core Damage	Train A onsite power, component cooling, high and low pressure makeup (ECCS), reactor coolant pump seal LOCA, containment filtration and heat removal.	4.4-6	9	7	*
PCC Area Fire	None	Component cooling, high and low pressure makeup (ECCS), reactor coolant pump seal LOCA, containment filtration, and heat removal.	4.1-6	10	8	*

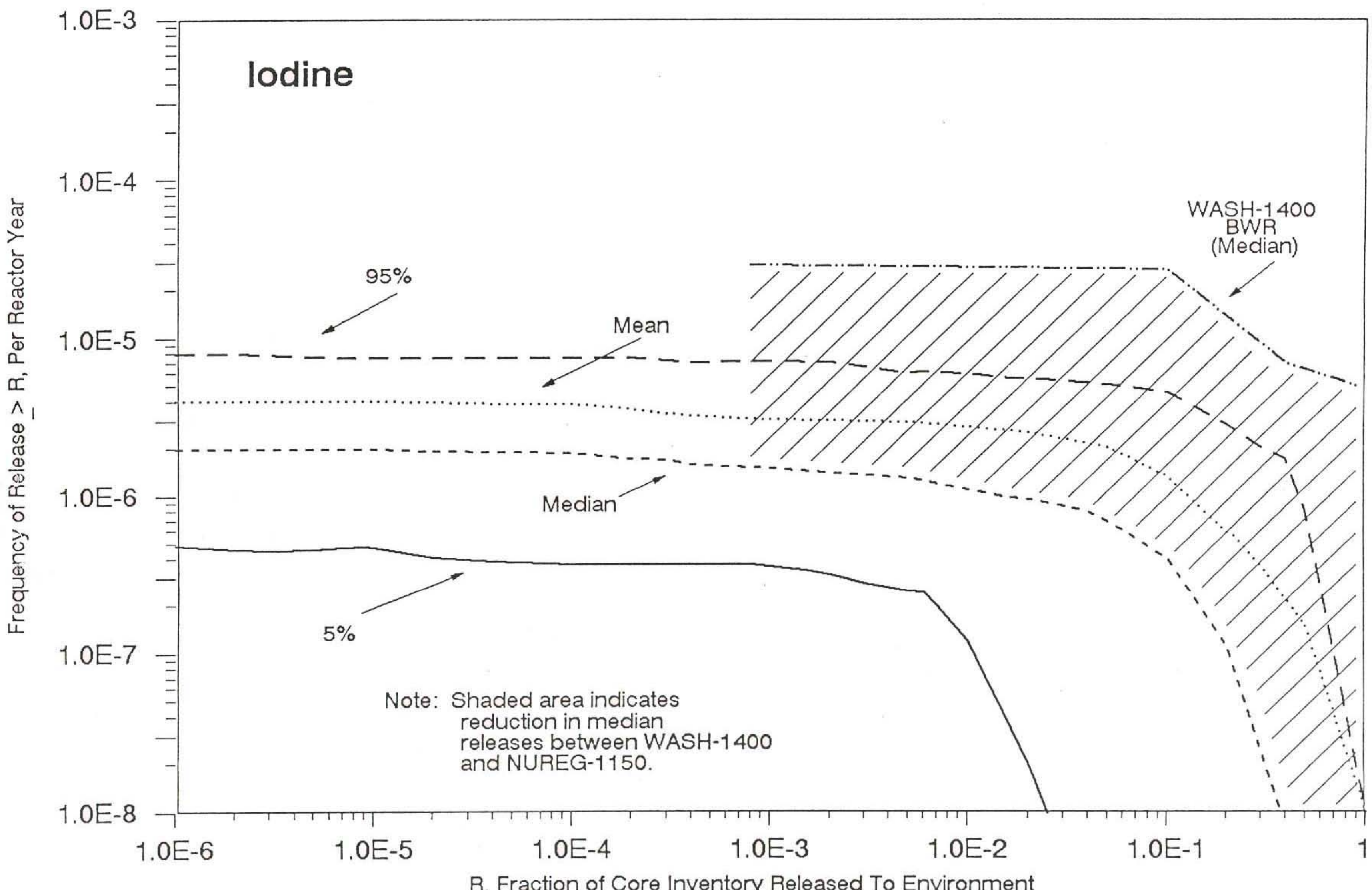
*Negligible contribution to risk.

NOTE: Exp tial notation is indicated in abbreviated form; i.e., 3.3-5 = 3.3 0^{-5}.

NUREG-1150 and RSS CDF for Peach Bottom



Comparison of Iodine Releases (Peach Bottom)



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 7×10^{-5}
  - Coal Mining: 24×10^{-5}
  - Fire Fighting: 40×10^{-5}
  - Police: 32×10^{-5}
  - US President $1{,}900 \times 10^{-5}$ (!)
- *Annual Public Risks*

  - Total 870×10^{-5}
  - Heart Disease 271×10^{-5}
  - All cancers 200×10^{-5}
  - Motor vehicles: 15×10^{-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)

Large Early Release Frequency

- LERF is being used as a surrogate for the early fatality QHO.

- It is defined as the frequency of those accidents leading to significant, unmitigated releases from containment in a time frame prior to effective evacuation of the close-in population such that there is a potential for early health effects.

- Such accidents generally include unscrubbed releases associated with early containment failure at or shortly after vessel breach, containment bypass events, and loss of containment isolation.

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

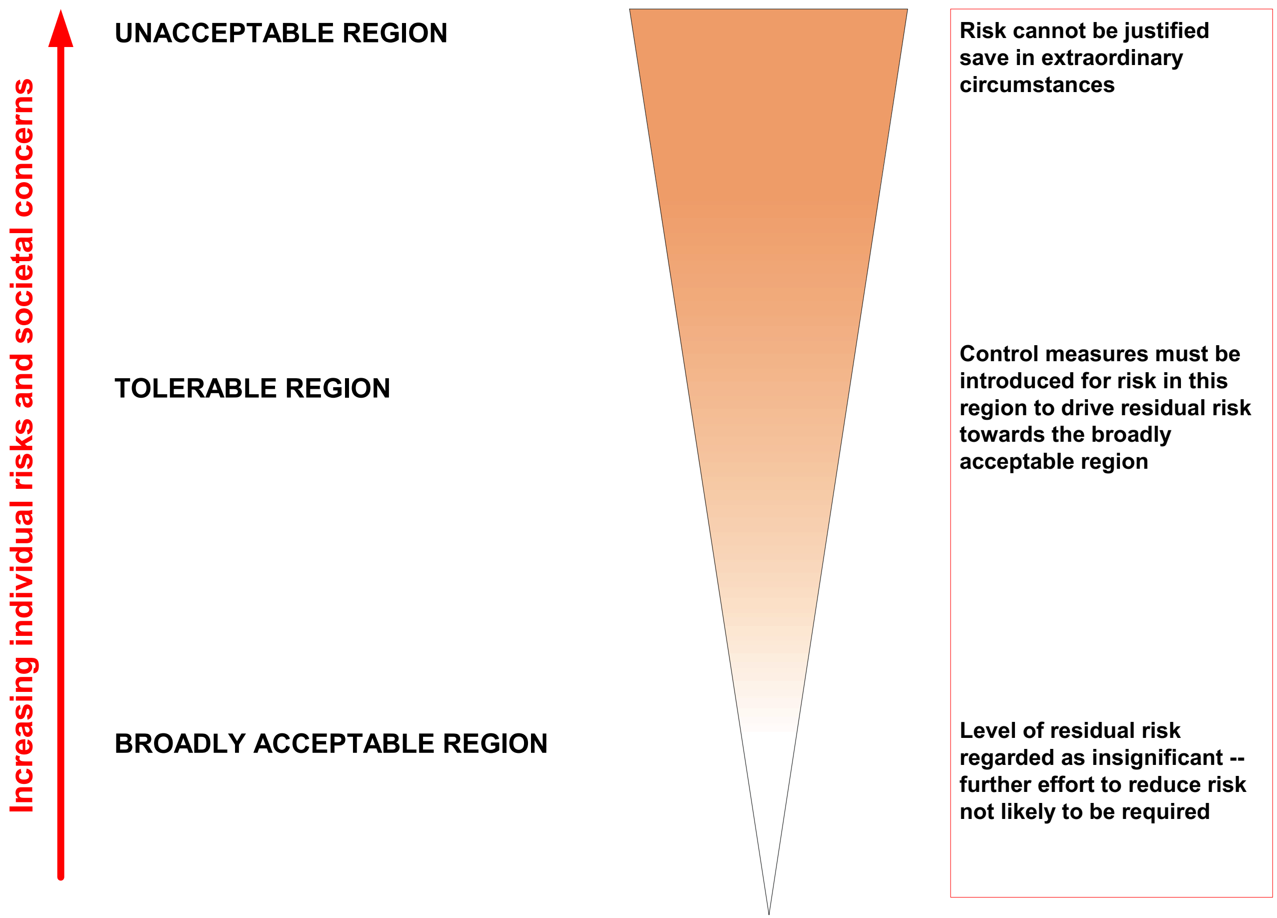
Public health effects

PLANT MODE
At-power Operation
Shutdown / Transition Evolutions

SCOPE
Internal Events
External Events

Uncertainties

“Acceptable” vs. “Tolerable” Risks (UKHSE)

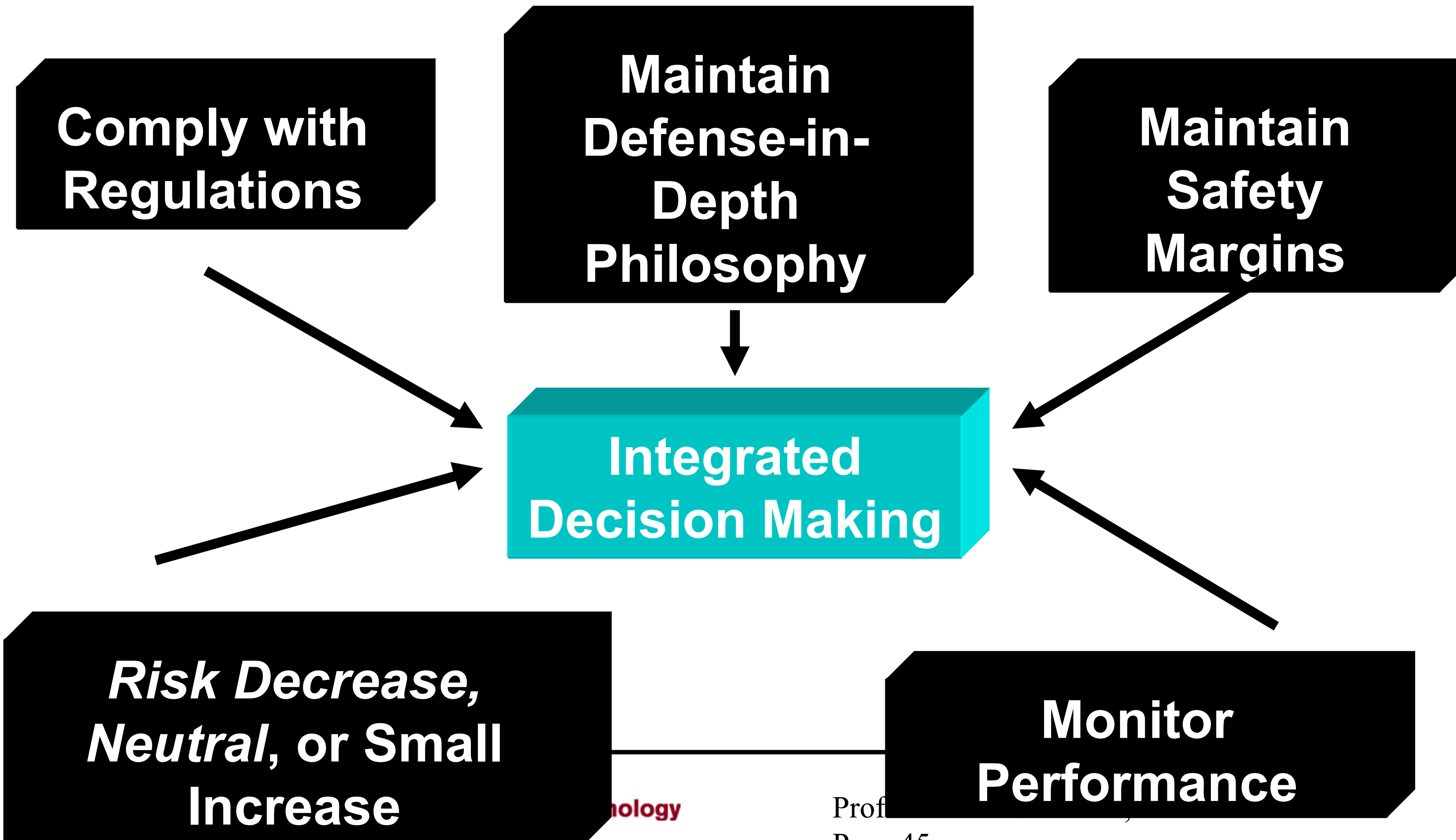


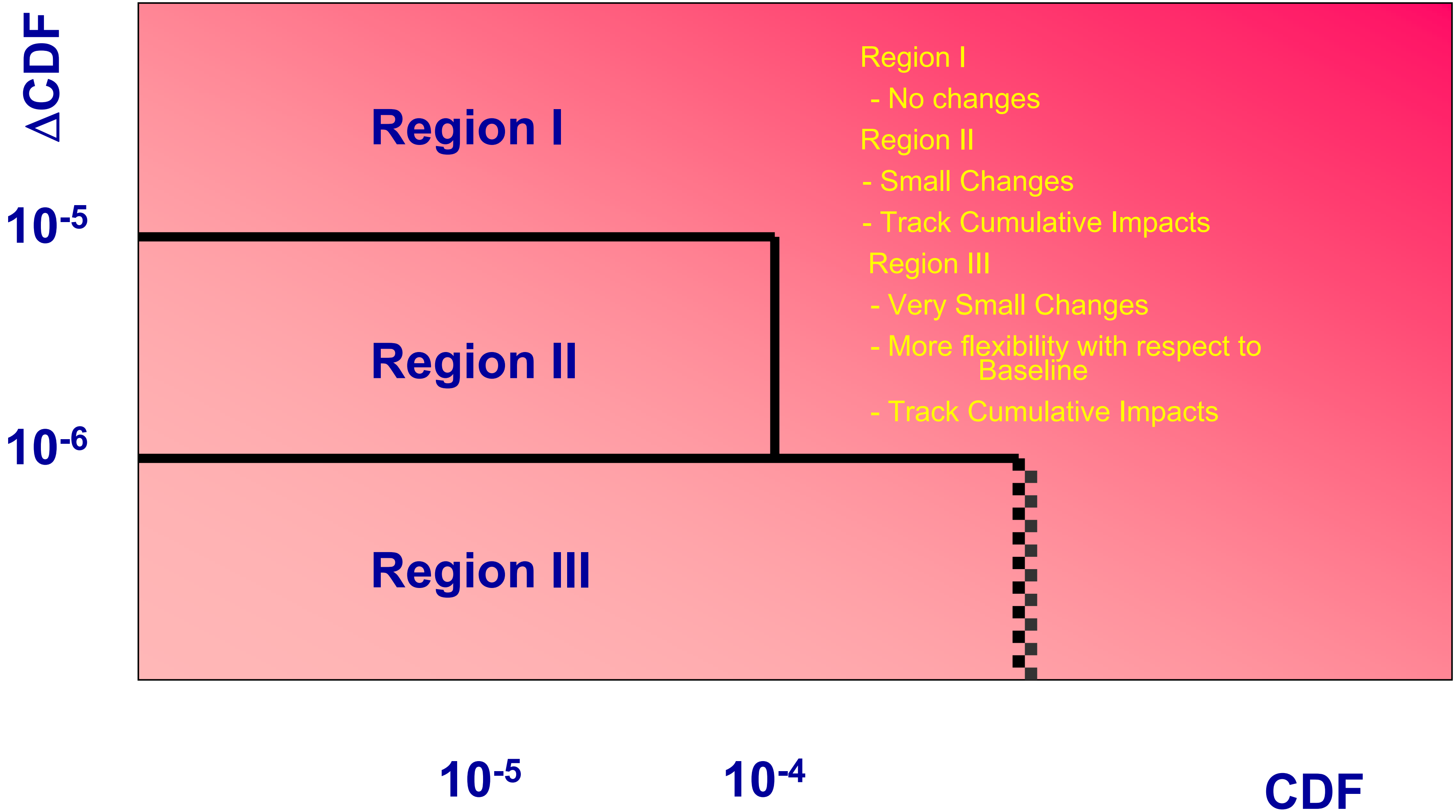
PRA Policy Statement (1995)

- 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

Homework

- Knief
  - Problems: 14.16, 19, 24, 28

22.091 Nuclear Reactor Safety
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