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# Risk-Informed Operations

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# The Pre-PRA Era

- **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.**

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# 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.**

# Reactor Safety Study (WASH-1400; 1975)

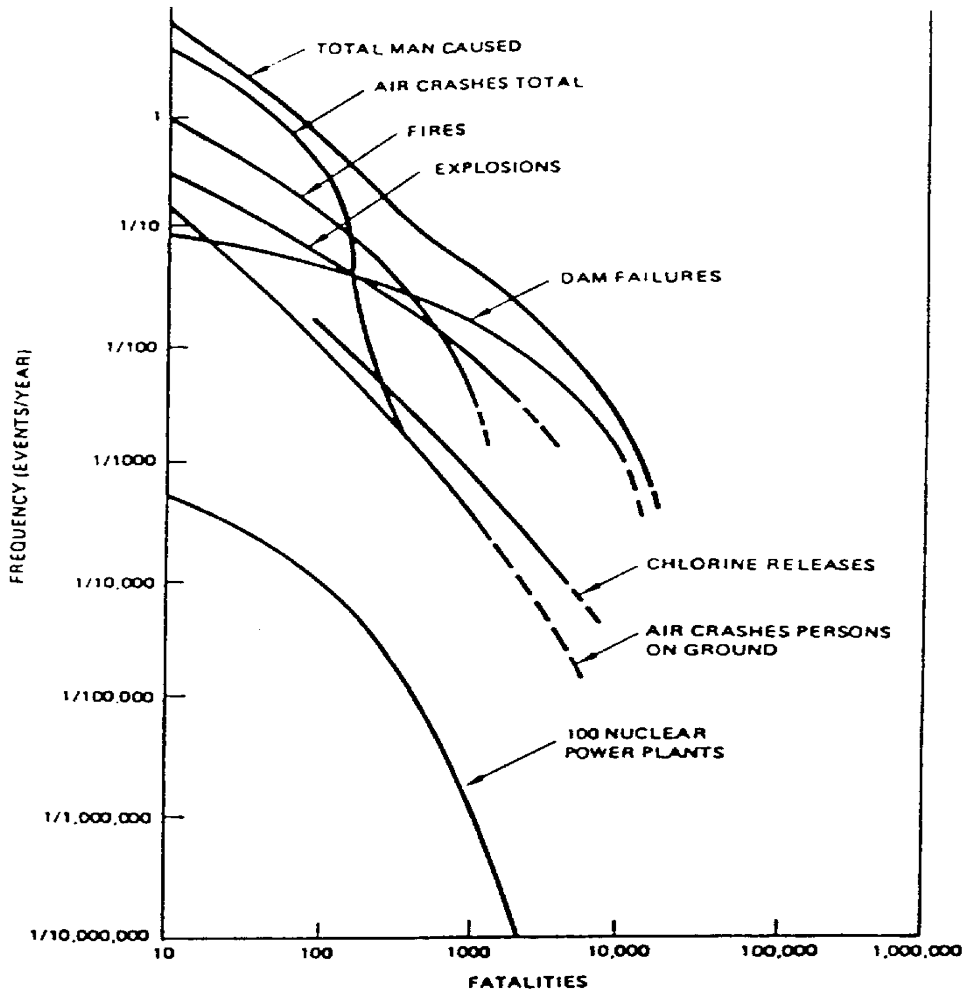
**Prior Beliefs:**

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)**

# SEABROOK STATION RISK RESULTS

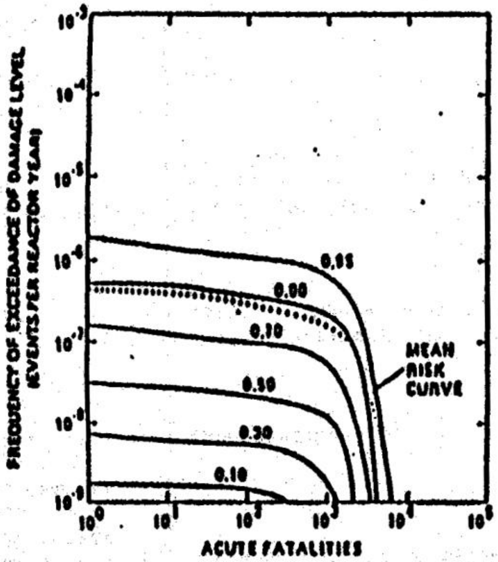



FIGURE 1-1a. RISK OF EARLY FATALITIES

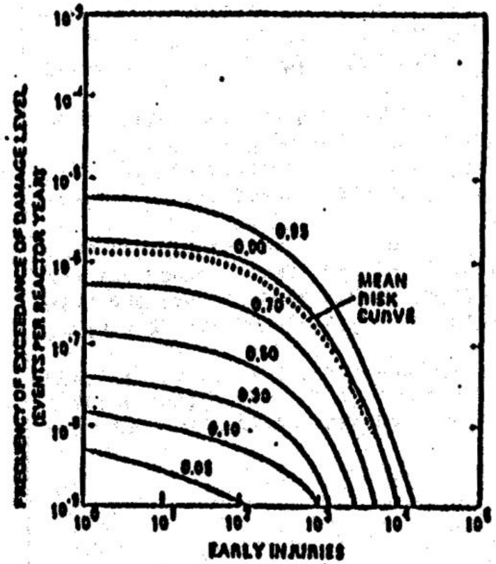



FIGURE 1-1b. RISK OF INJURIES

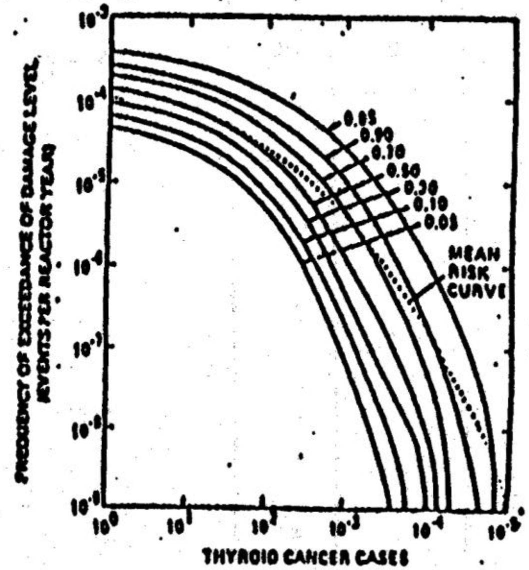



FIGURE 1-1c. RISK OF THYROID CANCER CASES

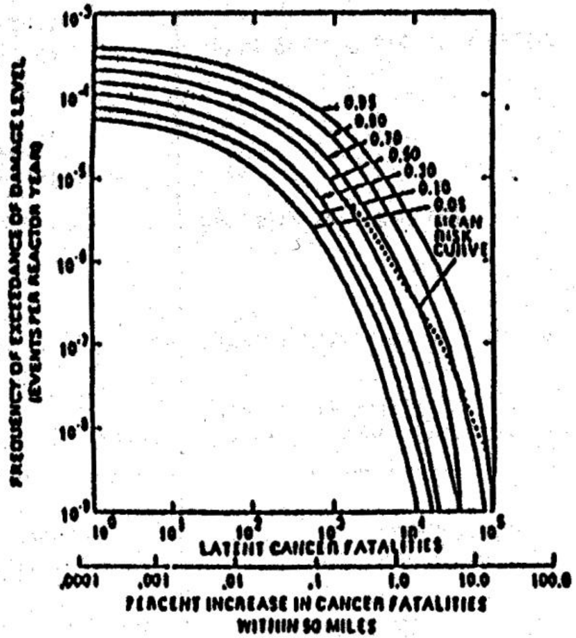



FIGURE 1-1d. RISK OF LATENT CANCER FATALITIES
(OTHER THAN FATAL THYROID CANCERS)

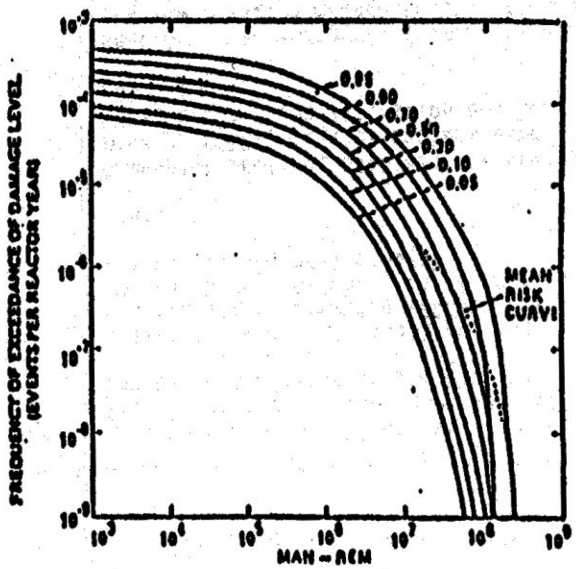



FIGURE 1-1e. RISK OF MAN-REM

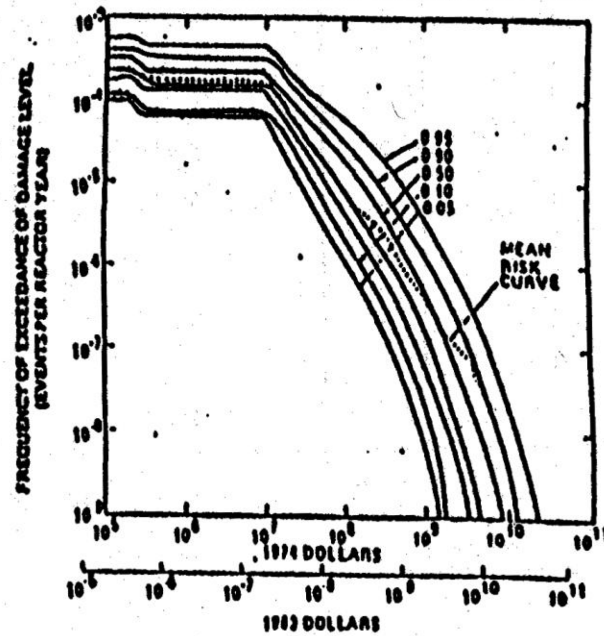



FIGURE 1-1f. RISK OF PROPERTY DAMAGE
AND EVACUATION COSTS

# At Power Level I Results

**CDF = $4.5\text{x}10^{-5}$ / yr**

**Initiator Contribution to CDF Total:**

- ✓ **Internal Events…………………..56%**
- ✓ **External Events …………………44%**
  - ➢ ***Seismic Events*** ***24%***
  - ➢ ***Fires*** ***18%***
  - ➢ ***Other*** ***2%***

K. Kiper, MIT Presentation, 2008

# 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% |

K. Kiper, MIT Presentation, 2008

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# 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**
**$5 \times 10^{-7}$/year for early death and**
**$2 \times 10^{-6}$/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.**

# 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

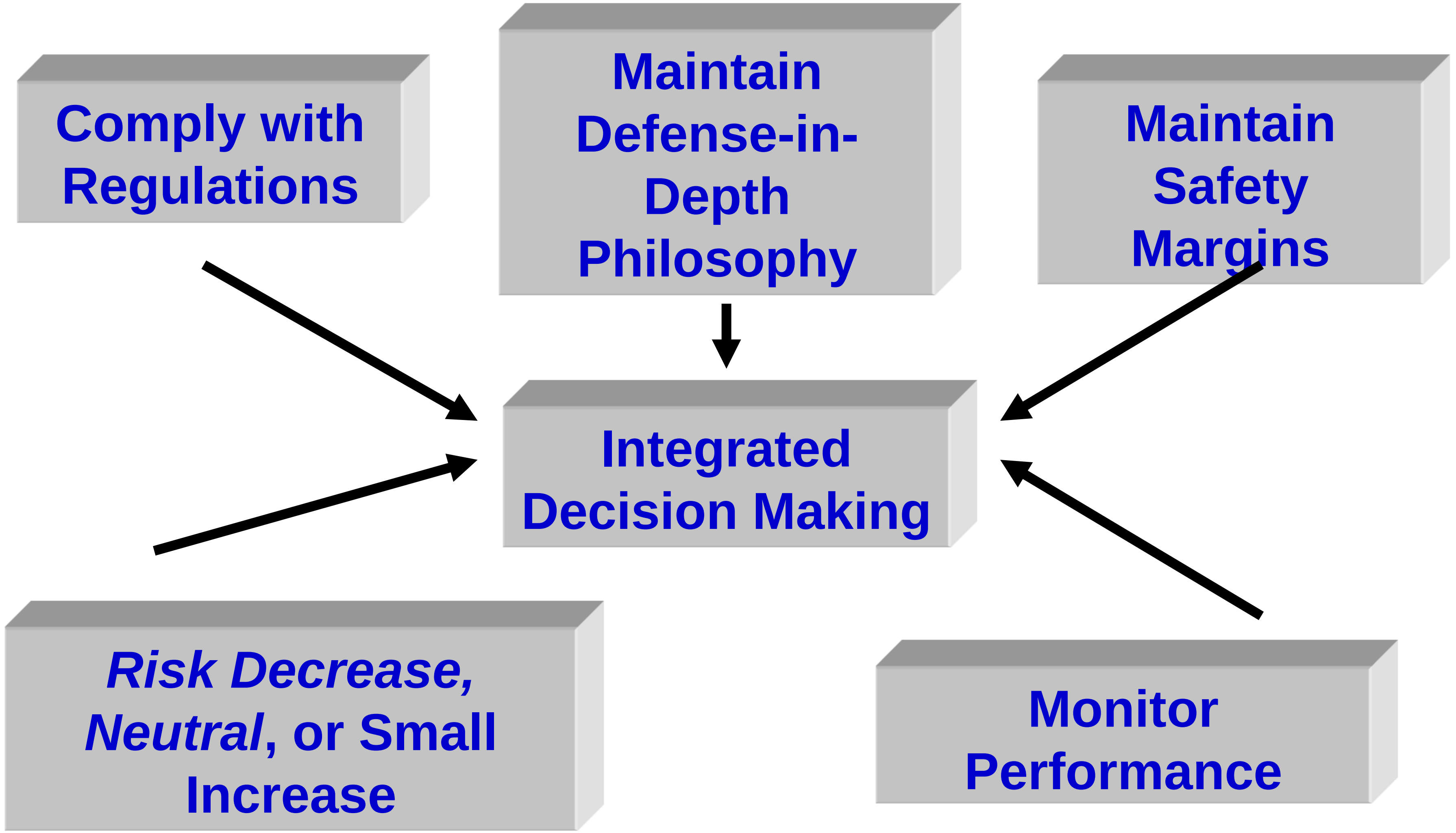
SCOPE
Internal Events
External Events

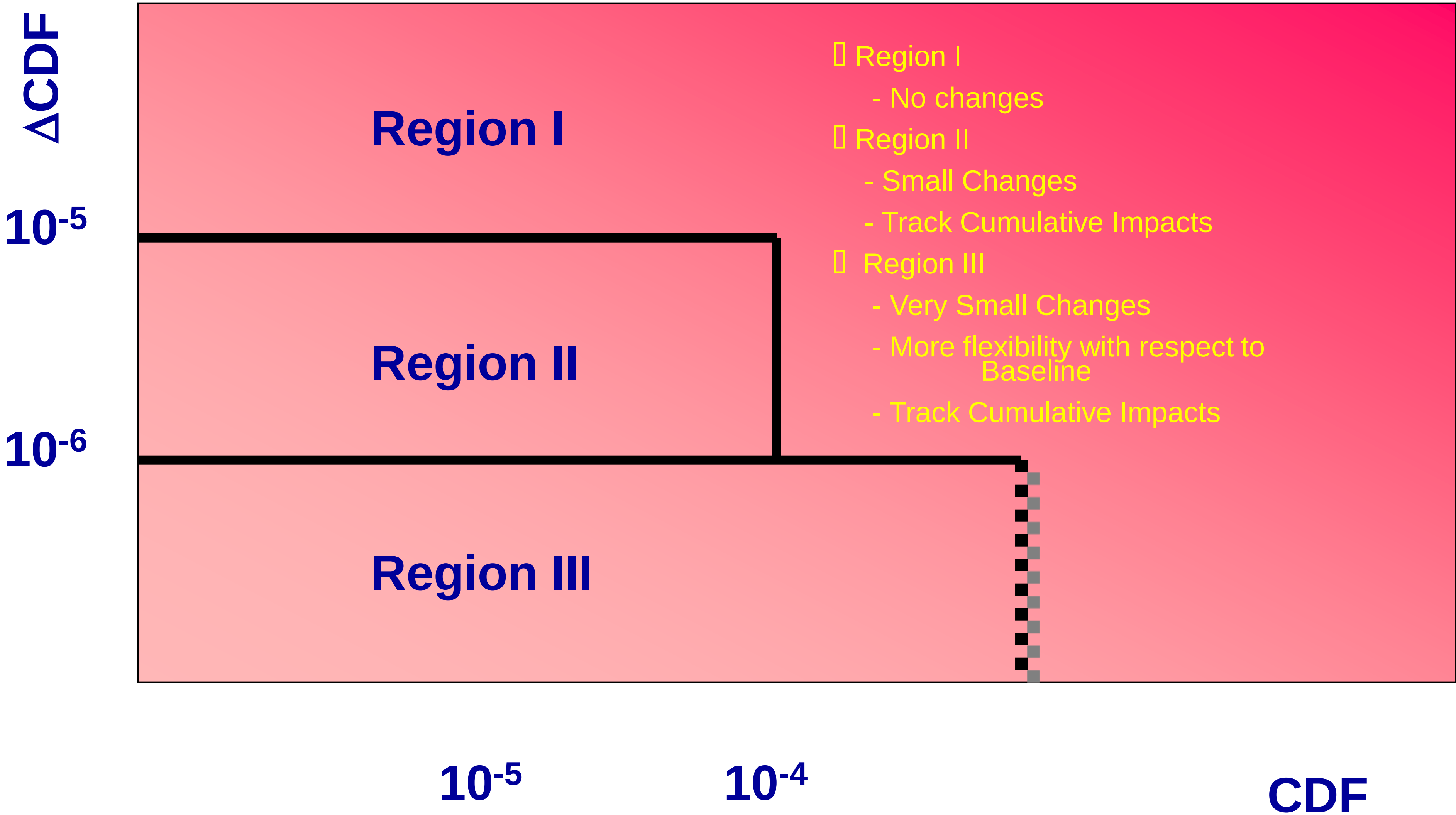
**Uncertainties**

# 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 Changes to the Licensing Basis (RG 1.174; 1998)



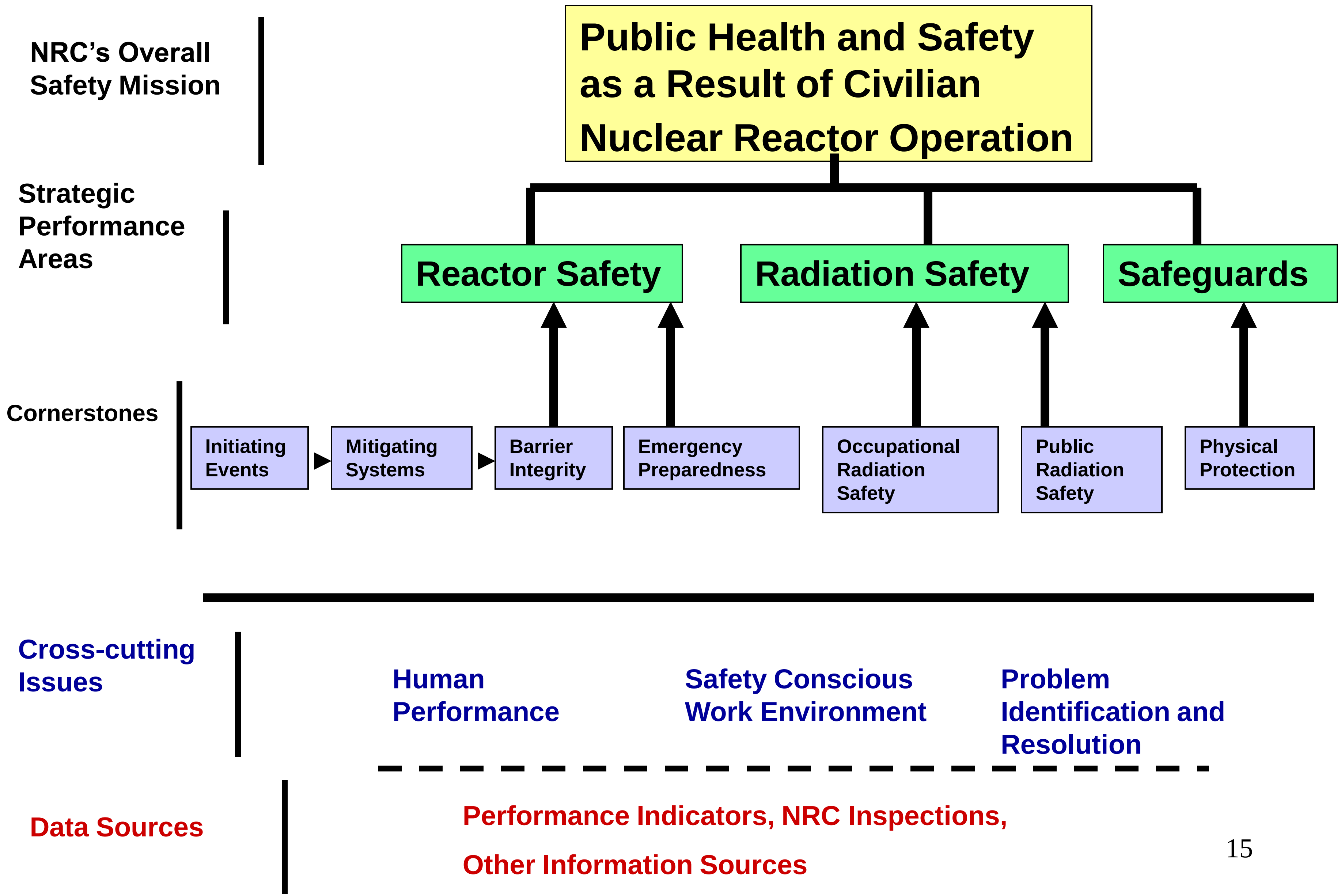



**Acceptance Guidelines for Core Damage Frequency**

# South Texas Project Experience with Allowed Outage Times

- **AOTs extended from 3 days to 14 days for emergency AC power and 7 days for Essential Cooling Water and Essential Chilled Water systems.**
- **Actual experience: Less than 5 days.**

# Reactor Oversight Process



# Levels of Significance Associated with Performance Indicators and Inspection Findings

| | |
|---|---|
| Very low risk significance (for PIs: Within peer performance) | $\Delta CDF < 1E\text{-}6$ |
| Low to moderate risk significance | $1E\text{-}6 < \Delta CDF < 1E\text{-}5$ |
| Substantive risk significance | $1E\text{-}5 < \Delta CDF < 1E\text{-}4$ |
| High risk significance | $\Delta CDF > 1E\text{-}4$ |

| | | Licensee Response Column | Regulatory Response Column | Degraded Cornerstone Column | Multiple Repetitive Degraded Cornerstone Column | Unacceptable Performance Column |
|---|---|---|---|---|---|---|
| Results | | All assessment inputs (performance Indicators (PI) and inspection findings) Green; cornerstone objectives fully met | One or two White inputs (in different cornerstones) in a strategic performance area; Cornerstone objectives fully met | One degraded cornerstone (2 White inputs or 1 Yellow input) or any 3 White inputs in a strategic performance area; cornerstone objectives met with minimal reduction in safety margin | Repetitive degraded cornerstone, multiple degraded cornerstones, multiple Yellow inputs, or 1 Red input[1]; cornerstone objectives met with longstanding issues or significant reduction in safety margin | Overall unacceptable performance; plants not permitted to operate within this band, unacceptable margin to safety |
| Response | Regulatory Conference | Routine Senior Resident Inspector (SRI) interaction | Branch Chief (BC) or Division Director (DD) meet with Licensee | DD or Regional Administrator (RA) meet with Licensee | EDO (or Commission) meet with Senior Licensee Management | Commission meeting with Senior Licensee Management |
| | Licensee Action | Licensee Corrective Action | Licensee corrective action with NRC oversight | Licensee self assessment with NRC oversight | Licensee performance improvement plan with NRC oversight | |
| | NRC Inspection | Risk-informed baseline inspection program | Baseline and supplemental inspection **95001** | Baseline and supplemental inspection **95002** | Baseline and supplemental inspection **95003** | |
| | Regulatory Actions | None | Document response to degrading area in assessment letter | Document response to degrading condition in assessment letter | 10 CFR 2.204 DFI<br>10 CFR 50.54(f) letter<br>CAL/Order | Order to modify, suspend, or revoke licensed activities |
| Communications | Assessment Report | BC or DD review / sign assessment report (w/ inspection plan) | DD review / sign assessment report (w/ inspection plan) | RA review / sign assessment report (w/ inspection plan) | RA review / sign assessment report (w/ inspection plan)<br><br>Commission informed | |
| | Public Assessment Meeting | SRI or BC meet with Licensee | BC or DD meet with Licensee | RA discuss performance with Licensee | EDO (or Commission) discuss performance with Senior Licensee Management | Commission meeting with Senior Licensee Management |
| | Increasing Safety Significance → | | | | | |

[1] It is expected in a few limited situations that an inspection finding of this significance will be identified that is not indicative of overall licensee performance. The staff will consider treating these inspection findings as exceptions for the purpose of determining appropriate actions

# ASME Boiler and Pressure Vessel Code (BPVC) Section XI

- **Class 1** components include piping and components whose failure would prevent orderly reactor shutdown and cause a loss of coolant in excess of normal makeup capability.

- **Class 2** components include safety system components of the following: residual heat removal system, reactor containment heat removal systems, emergency core cooling system including injection and recirculation portions, air cleanup systems used to reduce radioactivity within the reactor containment, containment hydrogen control system and portions of the steam and feedwater systems.

- **Class 3** components include portions of the reactor auxiliary systems that provide boric acid, emergency feedwater system, portions of components and process cooling systems (electrical and/or compressed air) that cool other safety systems including the spent pool cooling system, on-site emergency power supply and auxiliary systems.

# ASME BPVC Section XI Requirements

- **<u>Class 1 piping systems</u>: 25% welds examined every 10-year interval**
- **<u>Class 2 piping systems</u>: 7.5% welds examined every 10-year interval**
- **<u>Class 3 piping systems</u>: Only pressure test for leakage every 10-year interval**

- Failures are not occurring at the design-based locations.

- Failures are occurring at locations where unanticipated and unusual operating conditions have developed, such as, thermal stratification in sloping pipe systems (e.g., the pressurizer surge line), flow-assisted corrosion, and stress corrosion cracking.

# Risk Evaluation Matrix

**CONSEQUENCE CATEGORY (Safety Significance)**

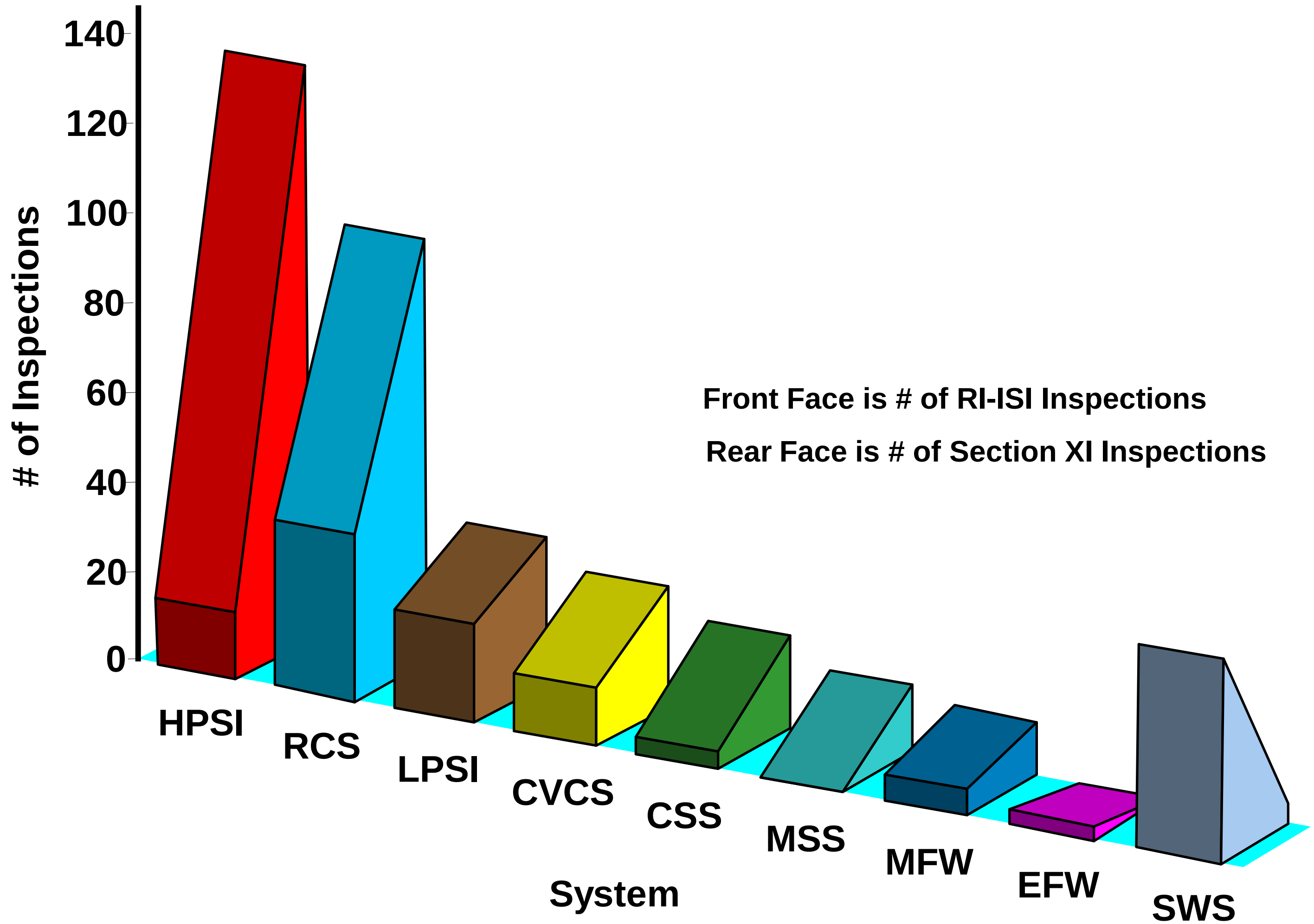
| DEGRADATION CATEGORY (Pipe Rupture Potential) | NONE | LOW | MEDIUM | HIGH |
|---|---|---|---|---|
| HIGH | LOW (Cat. 7) | MEDIUM (Cat. 5) | HIGH (Cat. 3) | HIGH (Cat. 1) |
| MEDIUM | LOW (Cat. 7) | LOW (Cat. 6) | MEDIUM (Cat. 5) | HIGH (Cat. 2) |
| LOW | LOW (Cat. 7) | LOW (Cat. 7) | LOW (Cat. 6) | MEDIUM (Cat. 4) |

# Numerical Criteria for Consequence Evaluation

| Consequence Category | Corresponding CCDP Range |
|---|---|
| High | CCDP > 1E-4 |
| Medium | 1E-6 < CCDP ≤ 1E-4 |
| Low | CCDP ≤ 1E-6 |

**NOTE: Additional numerical criteria may be used involving containment system performance, number of safety-system trains lost, importance measures, and others.**

# Plant X: Number of Inspections Before and After



# Cost and Man-Rem Savings