Risk-Informed Regulation at the U.S. Nuclear Regulatory Commission

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Robinson Fire – 3/28/2010

- On March 28, 2010, an automatic reactor trip occurred at H.B. Robinson Nuclear Generating Station due to a trip of reactor coolant pump 2. The trip was caused by electrical faults and subsequent fires on two non-safety related 4KV buses.

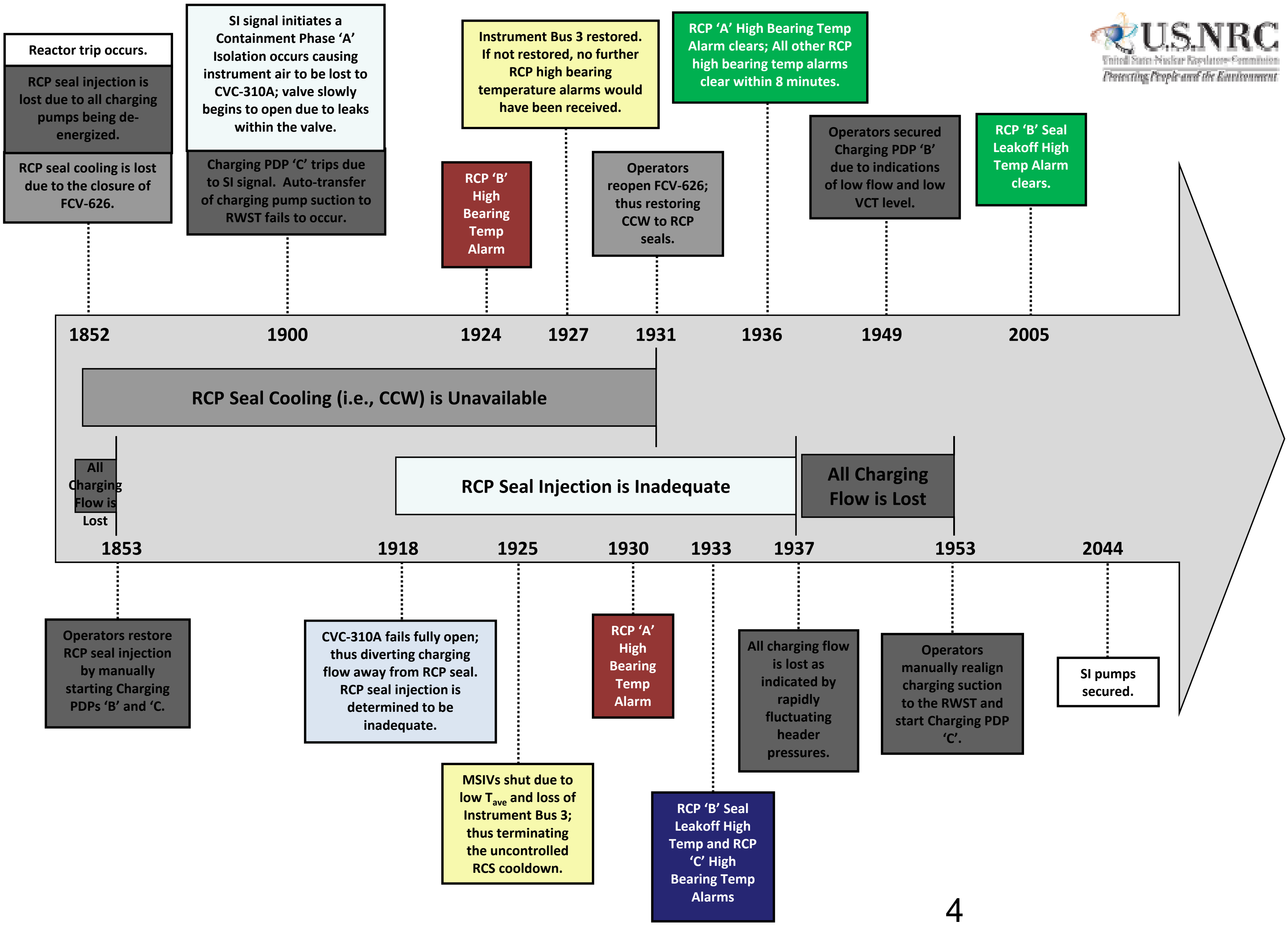
Robinson Fire – 3/28/2010

- Subsequent to the reactor trip, an automatic safety injection (SI) occurred due to an excessive RCS cooldown
- Plant response was complicated by equipment malfunctions and failure of the operating crew to understand plant symptoms and properly control the plant.

What was the significance of this event to the public?

How "near" of a miss was the event?

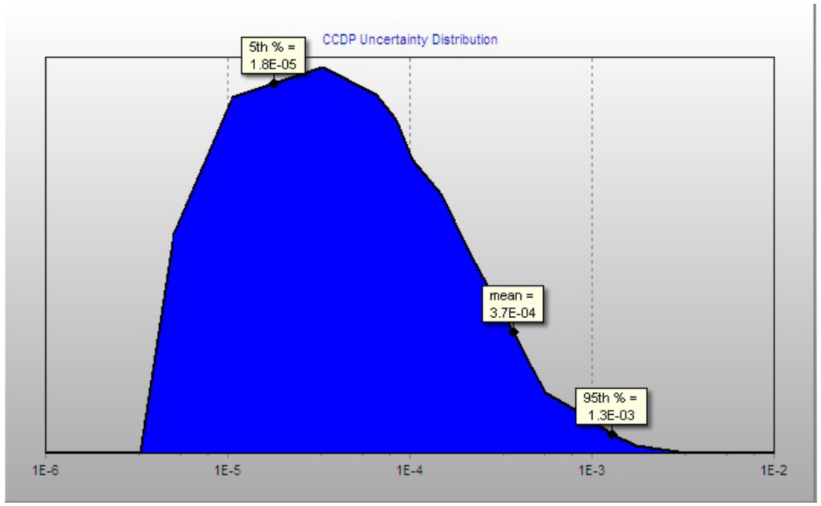
What could have made it better or worse?


Reactor trip occurs.
RCP seal injection is lost due to all charging pumps being de-energized.
RCP seal cooling is lost due to the closure of FCV-626.
1852
SI signal initiates a Containment Phase ‘A’ Isolation occurs causing instrument air to be lost to CVC-310A; valve slowly begins to open due to leaks within the valve.
Charging PDP ‘C’ trips due to SI signal. Auto-transfer of charging pump suction to RWST fails to occur.
1900
RCP ‘B’ High Bearing Temp Alarm
1924
Instrument Bus 3 restored. If not restored, no further RCP high bearing temperature alarms would have been received.
1927
Operators reopen FCV-626; thus restoring CCW to RCP seals.
1931
RCP ‘A’ High Bearing Temp Alarm clears; All other RCP high bearing temp alarms clear within 8 minutes.
1936
Operators secured Charging PDP ‘B’ due to indications of low flow and low VCT level.
1949
RCP ‘B’ Seal Leakoff High Temp Alarm clears.
2005
RCP Seal Cooling (i.e., CCW) is Unavailable
All Charging Flow is Lost
RCP Seal Injection is Inadequate
All Charging Flow is Lost
1853
Operators restore RCP seal injection by manually starting Charging PDPs ‘B’ and ‘C.
1918
CVC-310A fails fully open; thus diverting charging flow away from RCP seal. RCP seal injection is determined to be inadequate.
1925
MSIVs shut due to low T_{ave} and loss of Instrument Bus 3; thus terminating the uncontrolled RCS cooldown.
1930
RCP ‘A’ High Bearing Temp Alarm
1933
RCP ‘B’ Seal Leakoff High Temp and RCP ‘C’ High Bearing Temp Alarms
1937
All charging flow is lost as indicated by rapidly fluctuating header pressures.
1953
Operators manually realign charging suction to the RWST and start Charging PDP ‘C’.
2044
SI pumps secured.

Robinson Fire – 3/28/2010

What insights does PRA provide for this event?

- How close to core damage?
  – Core damage probability (CCDP) for this event is ~3.7E-4 (~1 in 2700)
- Dominant failures that could have led to core damage
  − RCP seal loss-of-coolant accident (LOCA)
  − Common cause failure of RHR or HPI pumps
  − Common cause failure of HPI injection MOVs
- Key operator actions
  − Trip the RCPs on loss of seal cooling
  − Initiate shutdown cooling mode RHR
- How well do we know the answer?



http://pbadupws.nrc.gov/docs/ML1124/ML112411359.pdf

Outline

- Origins and Mission of the NRC
- Historical NRC PRA Studies and Foundations
- Policy
- Risk-Informed Regulation
  - Regulatory Analysis & Backfitting
  - Licensing
  - Oversight
- Research Activities

Outline

- **Origins and Mission of the NRC**
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- Research Activities

Origins of the NRC

- Atomic Energy Act of 1946 – created the Atomic Energy Commission (AEC):
  - Priority was to maintain strict control over atomic technology and to investigate its military applications.
  - Acknowledged the potential peaceful benefits of atomic power, but did not allow for private, commercial application of atomic energy
- Atomic Energy Act of 1954 – assigned the AEC three main functions:
  - Continue weapons program,
  - Promote the commercial uses of nuclear power, and
  - Protect against the hazards of peaceful applications of atomic energy.

Origins of the NRC

- Atomic Energy Act of 1954 outlined a two-step procedure for granting licenses.
  - Construction permit issued if there was “reasonable assurance” that the prospective plant could be constructed and operated at the proposed site “without undue risk to the health and safety of the public.”
  - Operating license issued once the AEC determined the plant fully met safety requirements
  - An Advisory Committee on Reactor Safeguards (ACRS) (comprised part-time consultants who were recognized authorities on various aspects of reactor technology) also conducted independent reviews

Origins of the NRC

- Early licensing action reviewed on a case-by-case basis
- Difficulty in defining a “maximum credible accident”
  - In general terms, would be caused by one single equipment failure or operational error that would result in the most hazardous release of fission products.
- Deterministic acceptance criteria evolved
  - Spectrum of design basis events
  - Most limiting single failure

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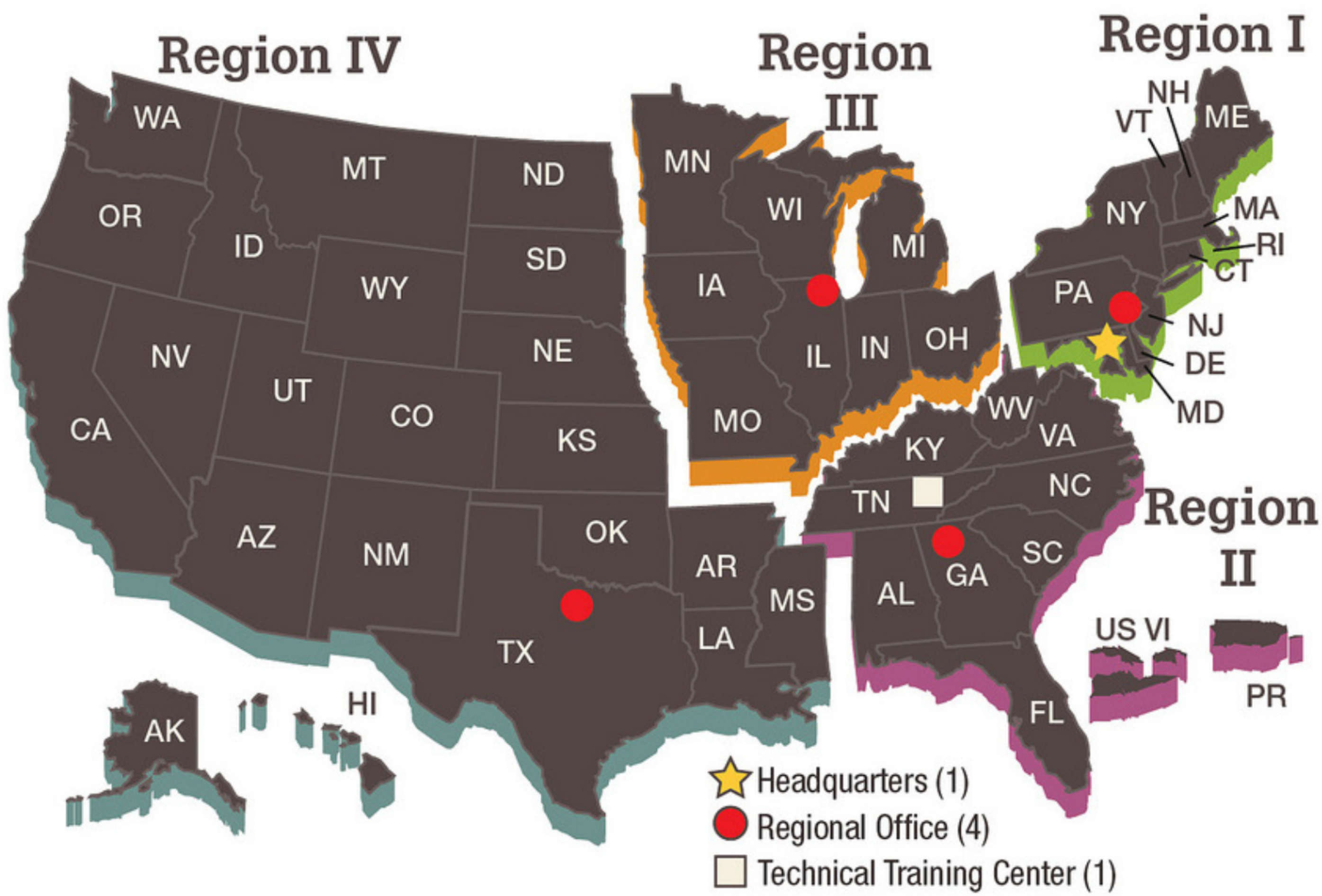
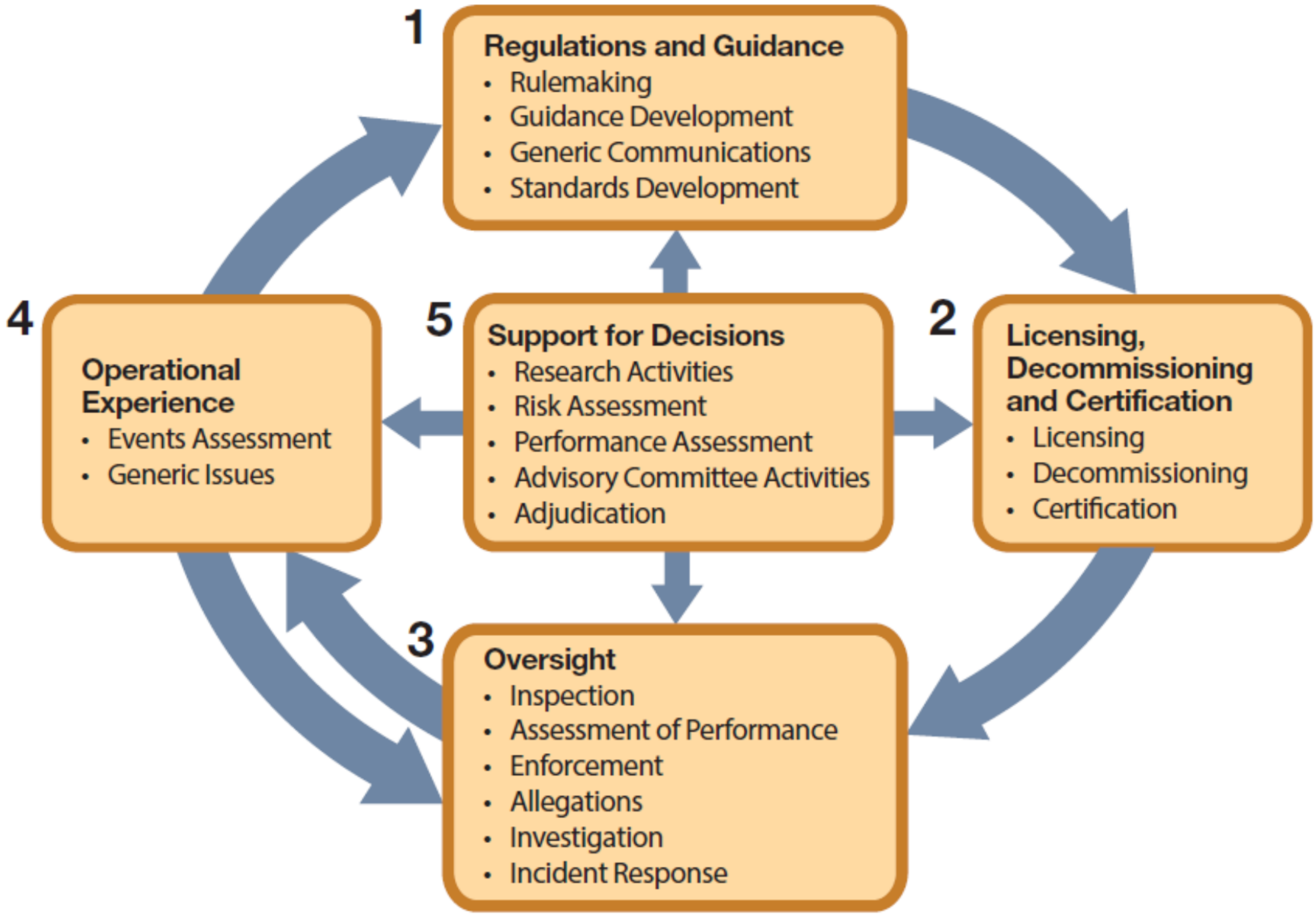
Origins of the NRC

- Energy Reorganization Act of 1974
  - Split the AEC into the U.S. Energy Research and Development Administration (ERDA) and the U.S. Nuclear Regulatory Commission
  - NRC Mission - License and regulate the Nation’s civilian use of radioactive materials to protect public health and safety, promote the common defense and security, and protect the environment.

NRC - Organization



NRC – Regulatory Process



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Historical PRA Context

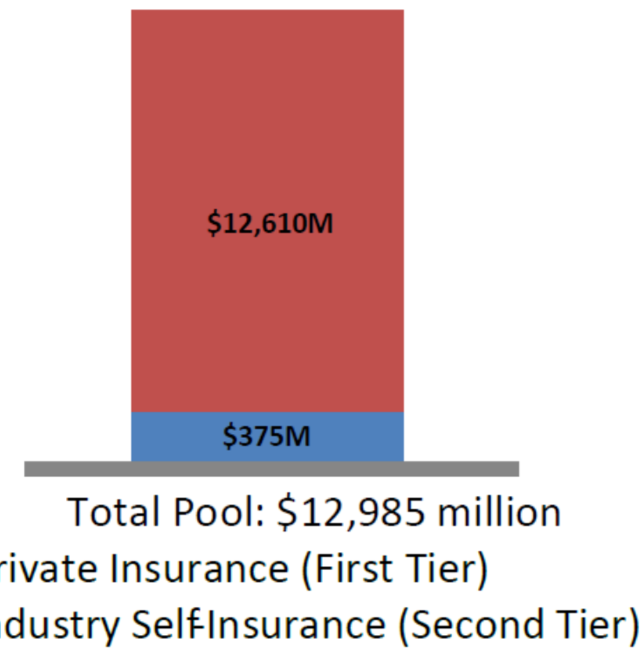
- Price-Anderson Nuclear Industries Indemnity Act (Price-Anderson Act) of 1957
  - Recognition that the chances of a severe reactor accident could not be reduced to zero.
  - Private insurers could not provide sufficient coverage to pay claims for deaths, injuries, and property damage following an accident
  - Government needed to understand the consequences and likelihood of severe accidents

Historical PRA Context

Price Anderson

- Covers liability claims for personal injury and property damage caused by a nuclear accident involving a commercial nuclear power plant.
- Helped encourage private investment in commercial nuclear power by placing a cap on the total liability for each licensee.
- The insurance pools paid approximately $71 million in claims and litigation costs associated with the Three Mile Island accident.

Nuclear Insurance Under The Price-Anderson Act



Owners of nuclear power plants pay for $375 million in private insurance. If a nuclear accident surpasses this amount, each plant pays up to $121.255 million into a second tier insurance pool.

http://www.nrc.gov/reading-rm/doc-collections/fact-sheets/nuclear-insurance.pdf

Historical PRA Context

- WASH-740, “Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants," (1957)
  - Performed to gauge insurance and liability costs for nuclear power plants
- Considered three accident scenarios:
  - All the fission products were released from the reactor core, but none escaped from the containment building.
  - All of the noble gases (Xe, Kr, Br) and iodines plus 1% of the Sr were released to the atmosphere.
  - 50% of all fission products were discharged to the atmosphere.

Historical PRA Context

- WASH-740 – PRA Perspective
  - Pessimistic consequence analysis
  - The probability of a catastrophic reactor accident was stated to be "exceedingly low"
  - Authors stated that no one knew, or probably would ever know, exactly how low the probability was
  - Some experts did provide probabilistic perspectives (but study stated the numbers had “no demonstrable basis in fact’’)
    - CD/no release: 1E-02/yr to 1E-4/yr
    - CD/release contained: 1E-3/yr to 1E-4/yr
    - Major release: 1E-5/yr to 1E-9/yr

Historical PRA Context

- WASH-1400, “Reactor Safety Study” (1975)
  - Commissioned in 1972 and led by Norman Rasmussen (MIT)
  - Purpose of the study was "to provide a basis for submitting recommendations to the Congress regarding the extension or modification of the Price-Anderson Act."
  - Utilized modern PRA techniques to determine likelihood of serious accidents
  - Analyzed two plants: Surry (PWR) and Peach Bottom (BWR)
  - Concluded that risks from nuclear power were very small in comparison to other risks

Historical PRA Context

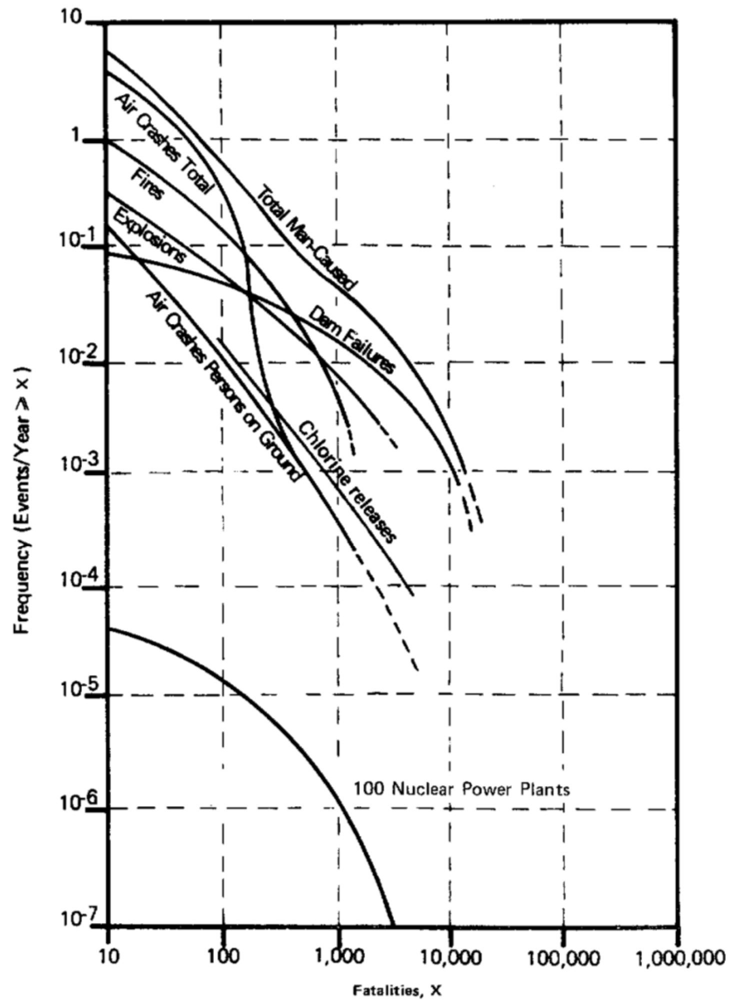


FIGURE 6-1 Frequency of Man-Caused Events Involving Fatalities.

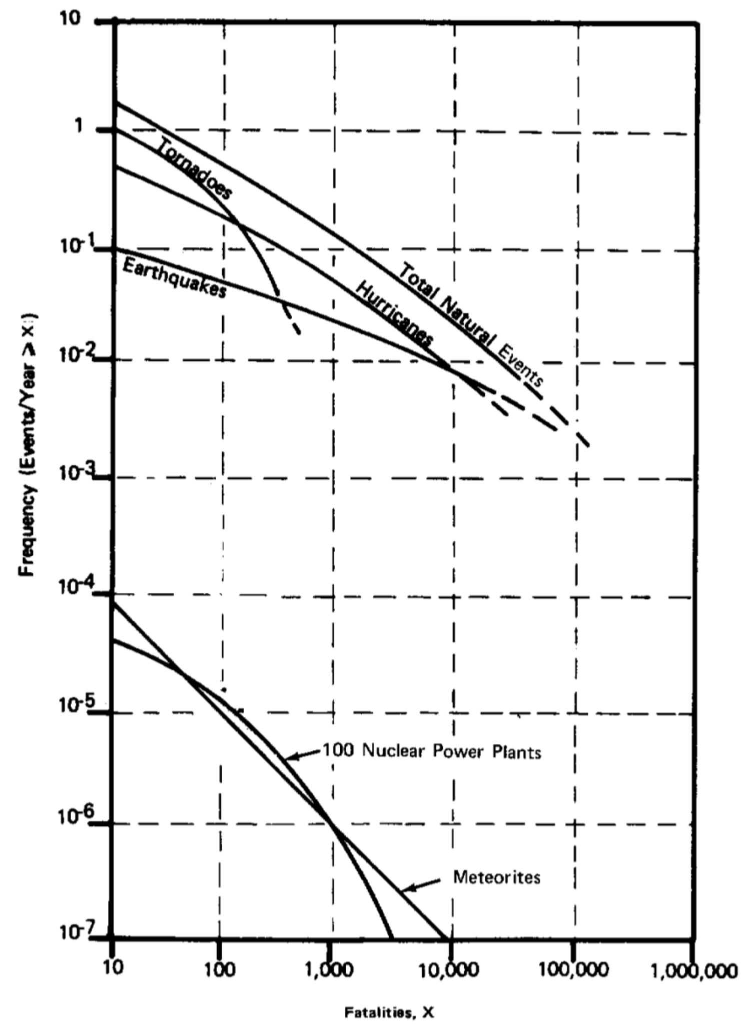


FIGURE 6-2 Frequency of Natural Events Involving Fatalities.

Historical PRA Context

- WASH-1400 drew some criticism:
  - The study failed to account for the many paths that could lead to major accidents
  - Data in the report did not support the conclusions about the relative risks of nuclear power
  - Weaknesses in the peer review process
- The Commission issued a policy statement in January 1979 that withdrew its full endorsement of the study's executive summary
  - However, it was noted that WASH-1400 was a substantial advance over previous attempts to estimate the risks of the nuclear power by making the study of reactor safety more rational, establishing the topology of many accident sequences, and in delineating procedures through which quantitative estimates of the risk can be derived

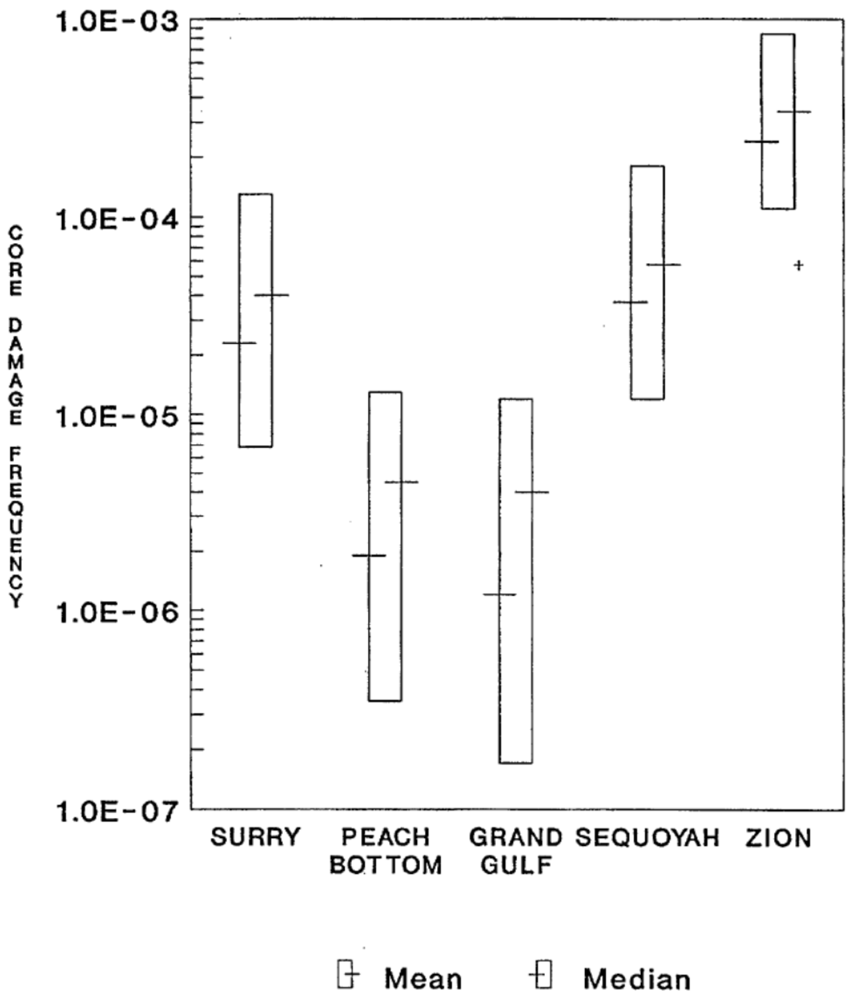
PRA Foundations

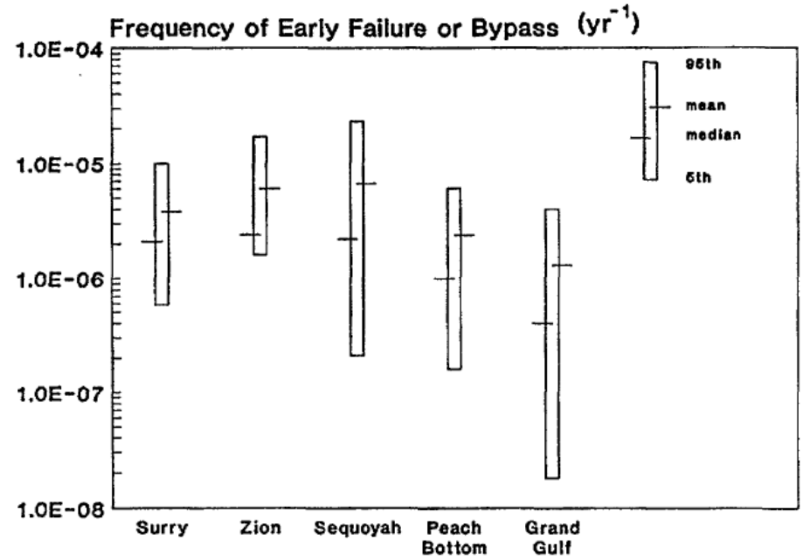
NUREG-1150,“Severe Accident Risks: An Assessment of Five U.S. Nuclear Power Plants” (1990)

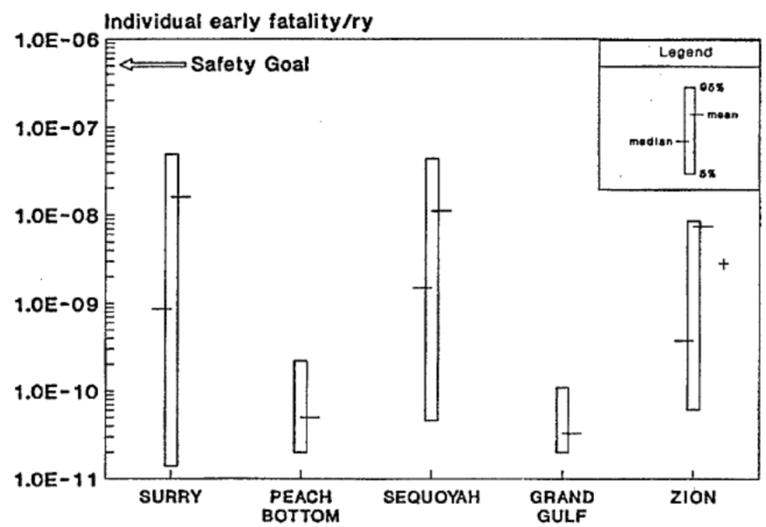
- Surry – W 3 Loop, Subatmospheric
- Zion – W 4 Loop, Large dry
- Sequoyah – W 4 Loop, Ice
- Peach Bottom – GE BWR4, Mark I
- Grand Gulf – GE BWR6, Mark III

- Updates the estimates of the WASH-1400 Reactor Safety Study;
- Includes quantitative estimates of risk uncertainty in response to a principal criticism of the Reactor Safety Study;
- Supported identification of plant-specific risk vulnerabilities for the five studied plants, supporting the development of the NRC's individual plant examination (IPE) process;

PRA Foundations







PRA Foundations

Severe Accident Vulnerabilities (GL 88-20)

- Individual Plant Examination for Severe Accident Vulnerabilities
  - NUREG-1560, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance "
- Individual Plant Examination of External Events (IPEEE) for Severe Accident
  - NUREG-1742, "Perspectives Gained From the Individual Plant Examination of External Events (IPEEE) Program"

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Policy Statements

Severe Reactor Accident Policy Statement (50 FR 32138; August 8, 1985)

- Clarified procedures for licensing new plant designs
  - Completion of PRAs for new designs
- Closed further action on operating plants
  - No need for severe accident rulemaking
  - Plans to conduct systematic search for severe accident vulnerabilities
- Noted importance of resolution of TMI action plan issues in the Commission's policy basis

Policy Statements

Safety Goal Policy Statement (51 FR 30028; August 21, 1986)

- Intended to answer the question how safe is safe enough
- Provided two quantitative health objectives (QHO) for the current generation of light water reactors
  - The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed 0.1% of the sum of prompt fatality risks resulting from other accident to which members of the U.S. population are generally exposed
  - The risk to the population in the area of nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed 0.1% of the sum of cancer fatality risks resulting from all other causes.

Policy Statements

- Based on CDC data (1), the QHO's translate to the following:
  - Unintentional Accidents: 41.3 per 100,000 per yr

    41.3/100,000 * 0.001 = ~5E-07/yr
  - Cancers: 185 per 100,000 per yr

    185/100,000 * 0.001 = ~2E-06/yr

Table 11. Death rates for 113 selected causes, United States, 2013

[Rates per 100,000 population in specified group. Populations used

Accidents (unintentional injuries) (V01-X59,Y85-Y86)	41.3
Malignant neoplasms (C00-C97)	185.0

(1) http://www.cdc.gov/nchs/data/nvsr/nvsr64/nvsr64_02.pdf

Policy Statements

- The following numerical objectives are used as surrogate safety goals (see NUREG-1860, App. D)
  - A LERF of <10-5 per year as a surrogate for the early fatality QHO
    - Worst case conditional probability of individual prompt early fatality (CPEF) for large early release from NUREG-1150 (Surry) is 3E-2.
    - 3E-2 fatality risk/large early release * 1E-05 LERF = 3E-7 individual prompt fatality risk/yr.
  - A CDF of <10-4 per year is the surrogate for the latent cancer QHO
    - Worst case conditional probability of latent cancer fatality (CPLF) from large release from NUREG-1150 (Surry) is 4E-03.
    - 4E-3 latent fatality/large release * 1E-4 core damage/year * 1 large release/core damage = 4E-07 individual latent cancer fatality risk/yr.

Policy Statements

PRA Policy Statement (60 FR 42622; August 16, 1995)

(1) The use of **PRA technology should be increased in all regulatory matters** to the extent supported by the state-of-the-art in PRA methods and data and in a manner that **complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy**.

(2) PRA and associated analyses should be used in regulatory matters, where practical **within the bounds of the state-of-the-art, to reduce unnecessary conservatism** associated with current regulatory requirements, regulatory guides, license commitments, and staff practices. Where appropriate, PRA should be used to support the proposal for additional regulatory requirements in accordance with 10 CFR 50.109 (Backfit Rule).

Policy Statements

PRA Policy Statement (con't)

(3) PRA evaluations in support of regulatory decisions should be as **realistic as practicable** and appropriate **supporting data should be publicly available for review**.

(4) The Commission's **safety goals for nuclear power plants and subsidiary numerical objectives are to be used** with appropriate consideration of uncertainties in making regulatory judgments on the need for proposing and backfitting new generic requirements on nuclear power plant licensees.

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Risk-Informed Regulation

- **Risk-Based** – regulatory decision-making approach that relies solely on the use of risk information from PRA results
- **Risk-Informed** - regulatory decision-making approach using a philosophy where risk insights are considered together with other factors to establish requirements that better focus licensee and regulatory attention on design and operational issues commensurate with their importance to health and safety.
- **Performance-Based** - regulatory approach that focuses on desired, measurable outcomes, rather than prescriptive processes, techniques, or procedures.

A few keywords…

- Regulations - http://www.nrc.gov/reading-rm/doc-collections/cfr/
- Regulatory Guide (RG) - http://www.nrc.gov/reading-rm/doc-collections/reg-guides/
- Standard Review Plan (SRP) - http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0800/
- NUREG Series Reports - http://www.nrc.gov/reading-rm/doc-collections/nuregs/
- Policy Statements - http://www.nrc.gov/reading-rm/doc-collections/commission/policy/
- Inspection Manual - http://www.nrc.gov/reading-rm/doc-collections/insp-manual/

A few keywords…

Regulation

- 10 CFR 50, Appendix A, Criterion 2
- **Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as** earthquakes, **tornadoes**, **hurricanes**, floods, tsunami, and seiches without loss of capability to perform their safety functions.

RG

- RG 1.76, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants"
  - NUREG/CR-4461, "Tornado Climatology of the Contiguous United States,"
- RG1.221, "Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants"
  - NUREG/CR-7004 Technical Basis for Regulatory Guidance on Design-Basis Hurricane-Borne Missile Speeds for Nuclear Power Plants
  - NUREG/CR-7005 Technical Basis for Regulatory Guidance on Design-Basis Hurricane Wind Speeds for Nuclear Power Plants

SRP

- Standard Review Plan Chapter 3.3.1, "Wind Loading"
- Standard Review Plan Chapter 3.5.1.4, "Missiles Generated By Tornadoes And Extreme Winds"

Regulatory Analysis and Backfitting

- PRA Screening Criteria:

CHANGE IN CORE DAMAGE FREQUENCY (ΔCDF)/RY

	1E-02	1E-01
1E-03 – 1E-04	PROCEED TO V/I PORTION OF REGULATORY ANALYSIS	PROCEED TO V/I PORTION OF REGULATORY ANALYSIS* (PRIORITY)
1E-04 – 1E-05	MANAGEMENT DECISION WHETHER TO PROCEED WITH V/I PORTION OF REGULATORY ANALYSIS	**PROCEED TO V/I PORTION OF REGULATORY ANALYSIS**
1E-05 – 1E-06	NO ACTION TAKEN**	**MANAGEMENT DECISION WHETHER TO PROCEED WITH V/I PORTION OF REGULATORY ANALYSIS**

1

ESTIMATED CONDITIONAL CONTAINMENT FAILURE PROBABILITY***

Regulatory Analysis and Backfitting

- Cost-benefit Guidelines (NUREG/BR-0158):

$$\text{Avoided Public Dose} = \left[\text{Accident Frequency} \times \text{Population Dose Factor}\right]_{\text{Status Quo}} - \left[\text{Accident Frequency} \times \text{Population Dose Factor}\right]_{\text{After Action}}$$

From PRA analysis…

Table 5.4 Weighted population dose factors for the five NUREG-1150 power reactors

Reactor	Type	Person-rem Within 50 miles from the Plant
Zion	PWR	1.95E+5
Surry	PWR	1.60E+5
Sequoyah	PWR	2.46E+5
Peach Bottom	BWR	2.00E+6
Grand Gulf	BWR	1.93E+5
Average		1.99E+5

Person-rem is converted to $ using conversion factor (currently $2000/person-rem) [NUREG-1530]

Risk-Informed Licensing

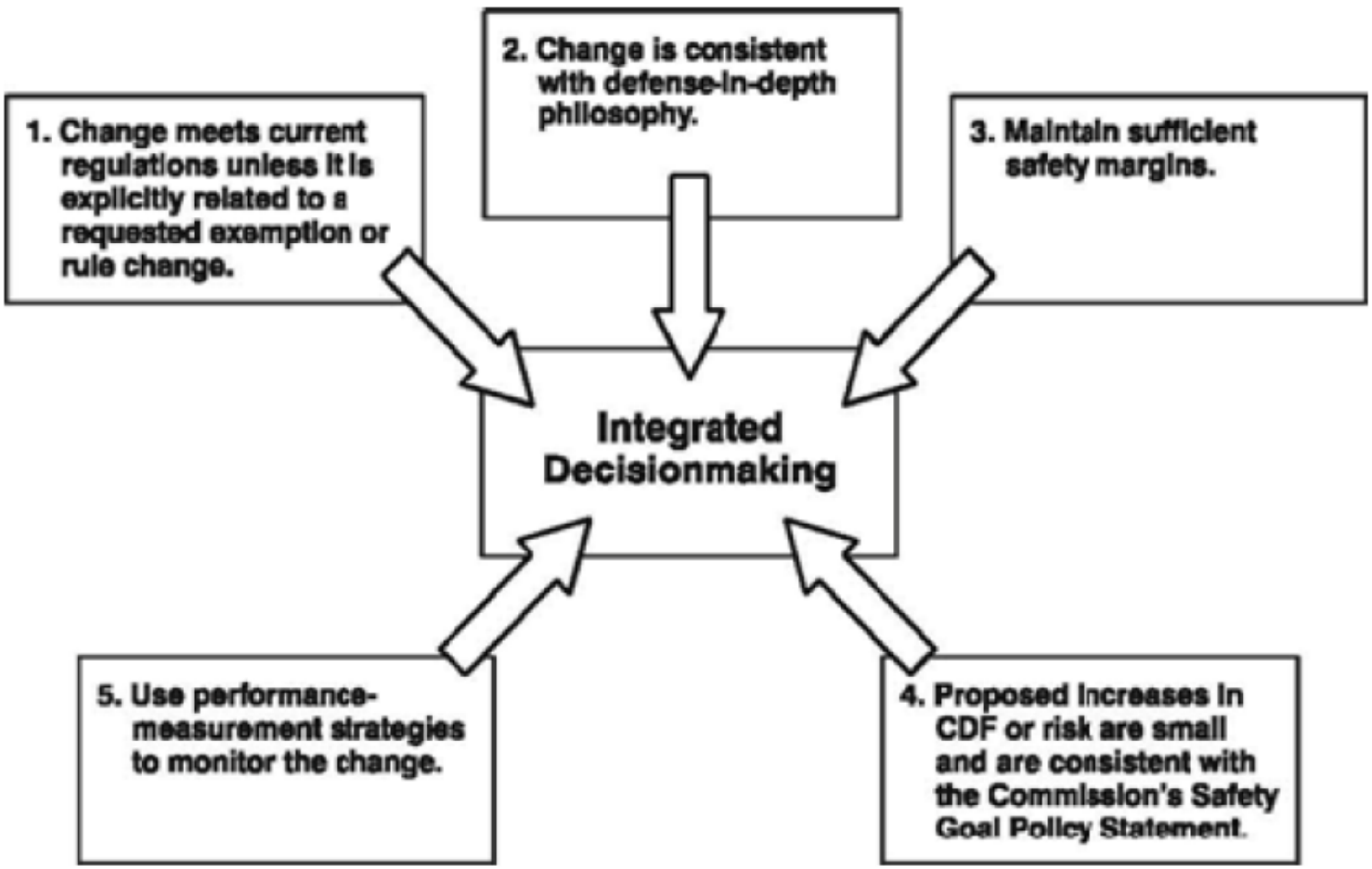

Figure 2 Principles of risk-informed integrated decisionmaking

Risk-Informed Licensing

- PRA Guidelines (RG 1.174)

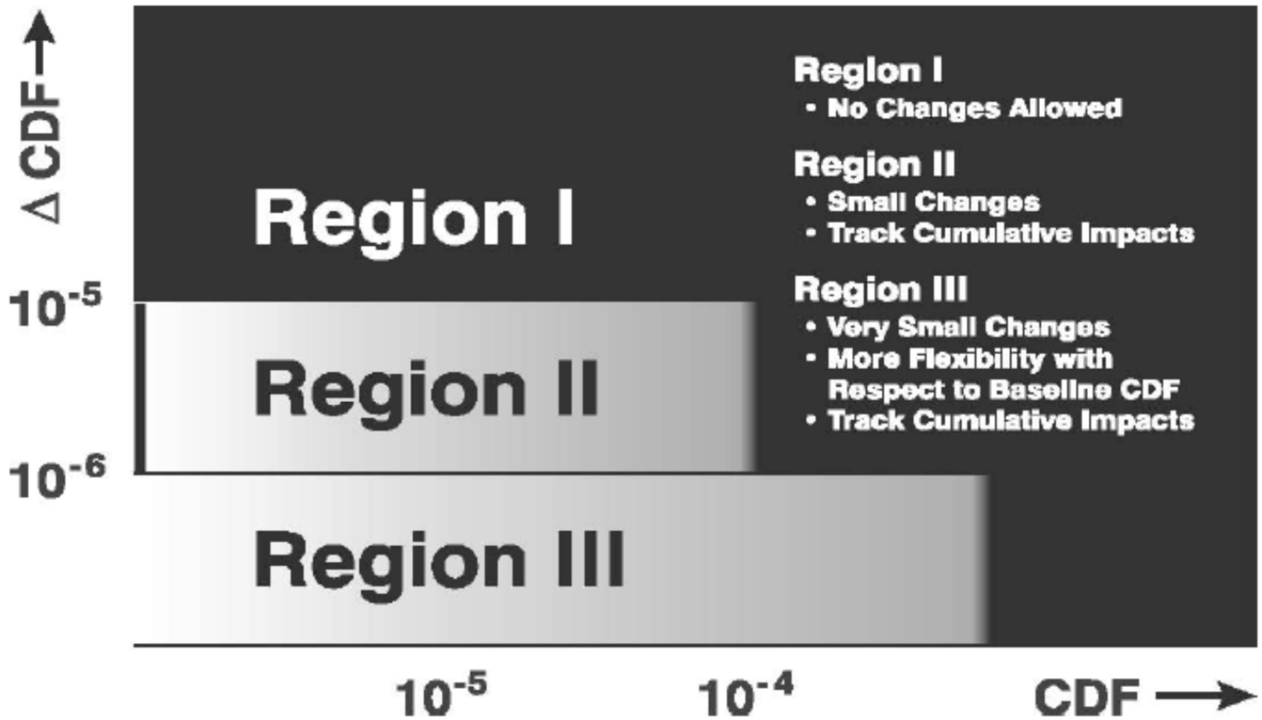

Figure 4 Acceptance guidelines* for core damage frequency

Risk-Informed Licensing

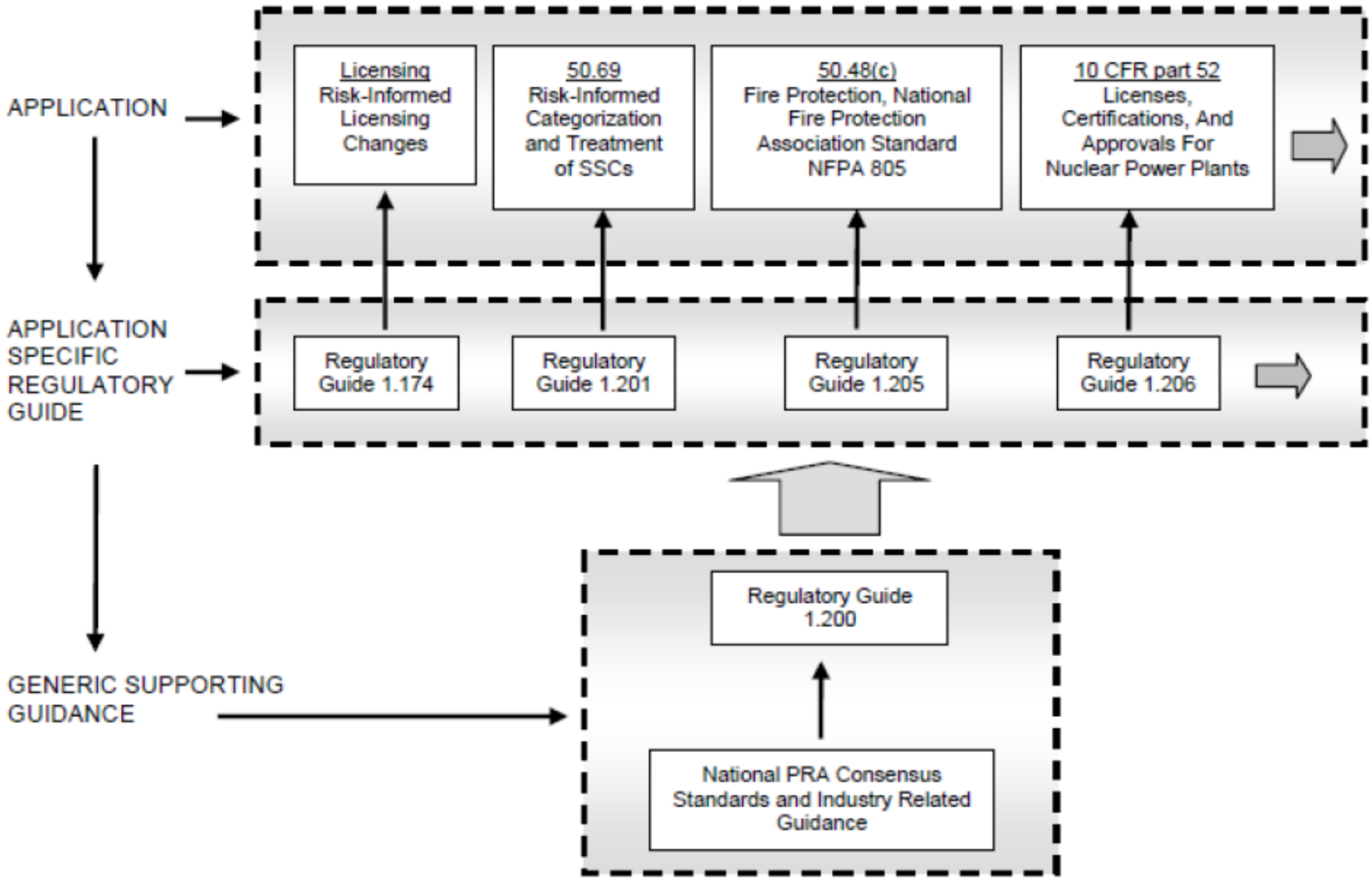

Figure 1 Relationship of Regulatory Guide 1.174 to other risk-informed guidance

Maintenance Rule

10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants" (1991)

- Requires monitoring of in-scope SSC performance to identify and address maintenance related issues
- Paragraph "(a)(4)" requires licensees to assess and manage the increase in risk that may result from the proposed maintenance activities
- Regulation is both risk-informed and performance-based

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Risk-Informed Licensing

Typical applications:

- Technical Specifications allowed outage time and surveillance frequency extensions
- Risk-informed In-service Inspection
- Risk-informed fire protection (NFPA-805)
- Quality categorization of systems, structures, and components
- Loss of Coolant Accident break size redefinition rulemaking

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Technology Neutral Framework

- NUREG-1860, “Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing”
  - documents a “framework" that provides an approach, scope and criteria that could be used to develop a set of requirements that would serve as an alternative to licensing future NPPs.
  - Has not been used, but has been investigated to support advanced reactor licensing.

Technology Neutral Framework

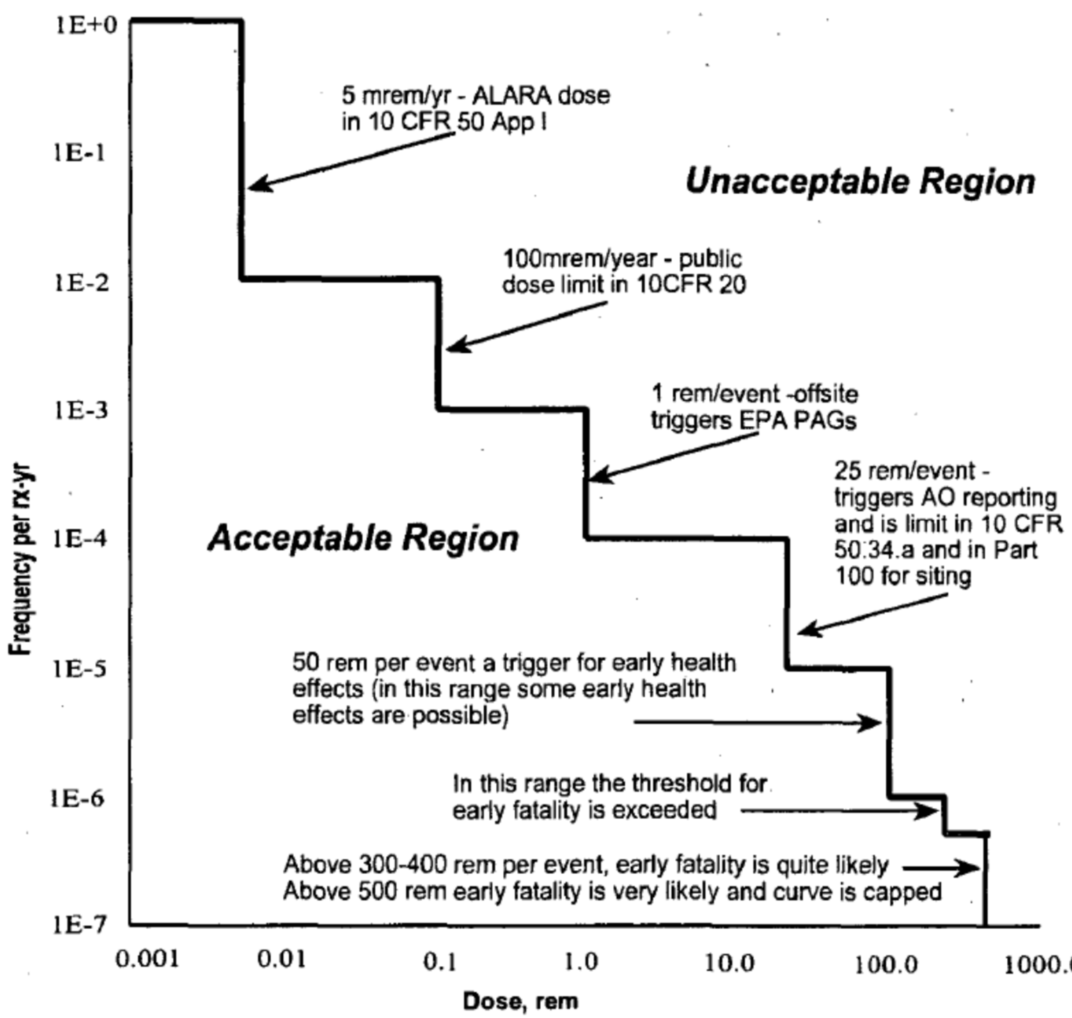

Figure 6-2 Frequency consequence curve

Frequency-Consequence Curve supports the identification of event sequences that the design and operation of the plant needs to be able to mitigate

Significance Determination Process

- Used to determine the safety or security significance of inspection findings identified within the seven cornerstones of safety at operating reactors.
- Risk-informed process
- Resulting safety significance of findings, combined with the results of the risk-informed performance indicator (PI) program, are used to define a licensee's level of safety performance, and to define the level of NRC engagement with the licensee

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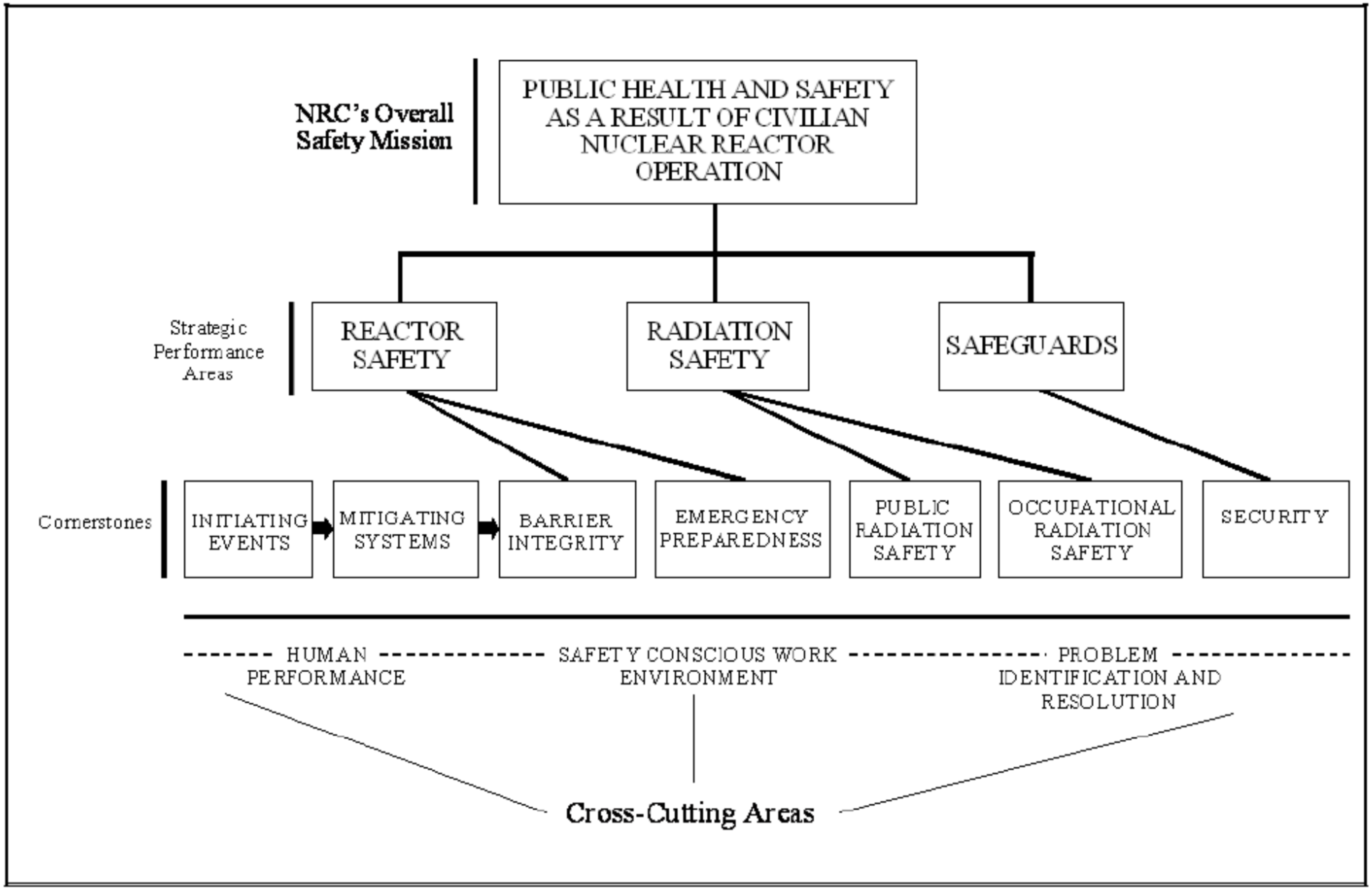
Significance Determination Process

- SDP Process
  - Characterization of licensee performance deficiency (violation or finding)
    - Screening
    - Detailed analysis (if required)
  - Significance And Enforcement Review Panel (if required)
  - Obtain licensee perspectives on analysis
  - Finalization of SDP characterization

Significance Determination Process



Significance Determination Process

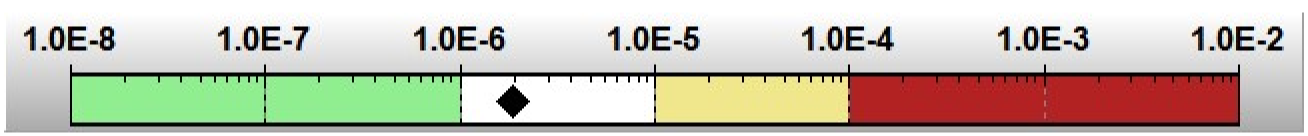
- **Green** - very low risk significance – cornerstone objectives are fully met with nominal risk and deviation
- **White** - low to moderate risk significance – indicates an acceptable level of performance by the licensee, but outside the nominal risk range.
- **Yellow** - substantive risk significance – indicates a decline in licensee performance that is still acceptable with cornerstone objectives met, but with significant reduction in safety margin.
- **Red** - high risk significance – indicates a decline in licensee performance that is associated with an unacceptable loss of safety margin

ΔCDF < 1E-6 **ΔLERF < 1E-7**	Green
1E-6 < ΔCDF < 1E-5 **1E-7 < ΔLERF < 1E-6**	White
1E-5 < ΔCDF < 1E-4 **1E-6 < ΔLERF < 1E-5**	Yellow
ΔCDF > 1E-4 **ΔLERF > 1E-5**	Red

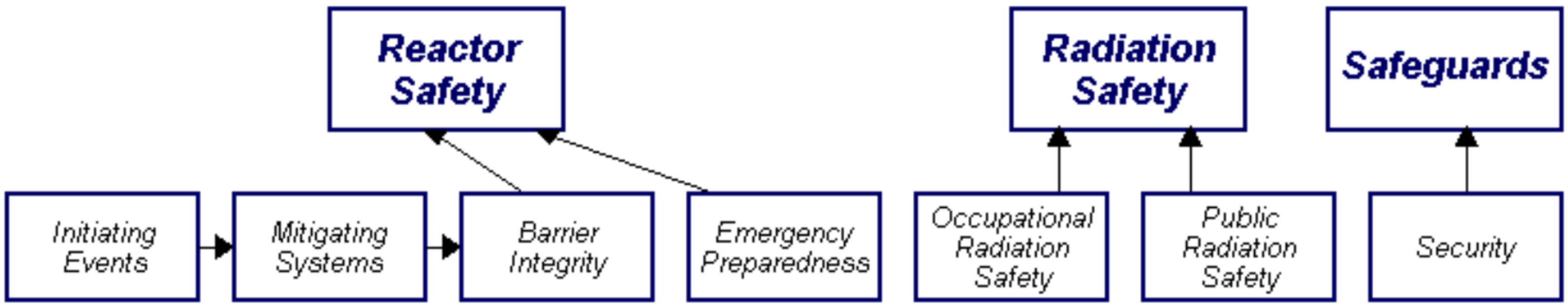
Simple example

- Train “A” DC Battery unavailable for one week due to a performance deficiency
- Battery “A” Risk Achievement Worth is 3.8
  - $RAW_i = CDF_{i,failed}/CDF_{Base}$
- Baseline CDF = 3.5E-5/ry
- $\Delta CDF = (RAW-1)\ CDF_{Base}$ (Exposure Fraction)

$$\Delta CDF = (3.8 - 1)\ (3.5E\text{-}5/ry)\ (1\ week/\ 52\ week)$$
$$= 1.9E\text{-}6\ (WHITE)$$



Significance Determination Process



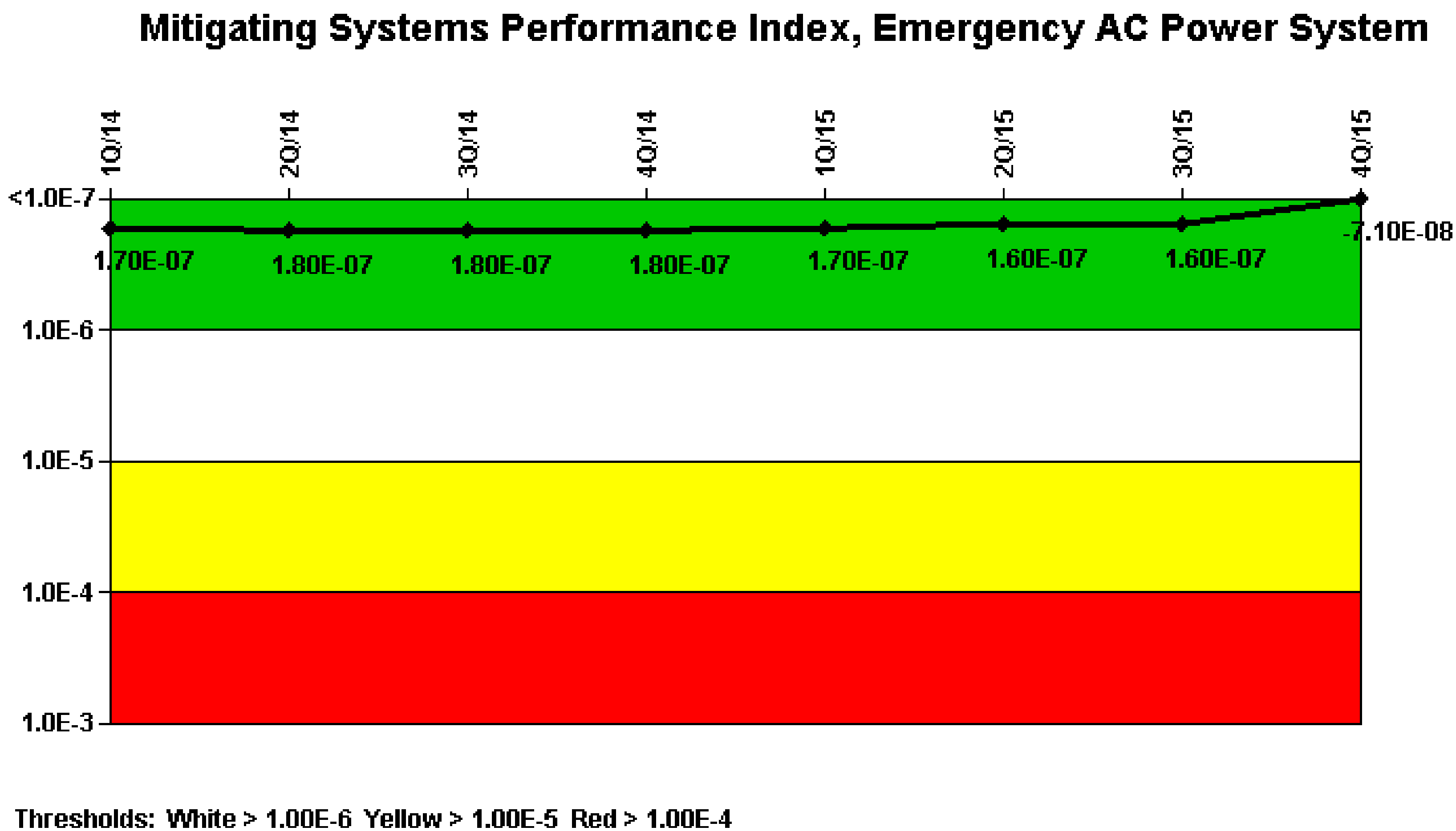
Most Significant Inspection Findings

	Initiating Events	Mitigating Systems	Barrier Integrity	Emergency Preparedness	Occupational Radiation Safety	Public Radiation Safety	Security
4Q/2015	G	G	G	No findings this quarter	No findings this quarter	G	G
3Q/2015	G	G	No findings this quarter	No findings this quarter	G	No findings this quarter	No findings this quarter
2Q/2015	G	W (1)	No findings this quarter	No findings this quarter	No findings this quarter	No findings this quarter	G
1Q/2015	G	G	No findings this quarter	No findings this quarter	No findings this quarter	No findings this quarter	GTG

Miscellaneous findings

See http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/pim_summary.html

Significance Determination Process



See http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/pi_summary.html

Significance Determination Process

Figure 1: Reactor Oversight Process Action Matrix

		Licensee Response Column (Column 1)	Regulatory Response Column (Column 2)	Degraded Performance Column (Column 3)	Multiple/Repetitive Degraded Cornerstone Column (Column 4)	Unacceptable Performance Column (Column 5)	IMC 0350 Process[1]
RESULTS		All assessment inputs (performance indicators and inspection findings) green; Cornerstone objectives fully met	One or Two white inputs in a strategic performance area; Cornerstone objectives met with minimal degradation in safety performance	One degraded cornerstone (3 white inputs or 1 yellow input), or Any 3 white inputs in a strategic performance area; Cornerstone objectives met with moderate degradation in safety performance	Repetitive degraded cornerstone, Multiple degraded cornerstones, Multiple yellow inputs, or One red input; Cornerstone objectives met with longstanding issues or significant degradation in safety performance	Overall unacceptable performance; Plants not permitted to operate within this band; Unacceptable margin to safety	Plants in a shutdown condition with performance problems are placed in the IMC 0350 process
RESPONSE	Regulatory Performance Meeting	None	Branch Chief or Division Director meets with licensee	Regional Administrator or designee meets with senior licensee management.	EDO/DEDO or designee meets with senior licensee management	EDO/DEDO or designee meets with senior licensee management	RA/EDO or designee meets with senior licensee management
	Licensee Action	Licensee corrective action	Licensee root cause evaluation and corrective action with NRC oversight	Licensee cumulative root cause evaluation with NRC oversight	Licensee performance improvement plan with NRC oversight		Licensee performance improvement & restart plan with NRC oversight
	NRC Inspection	Risk-informed baseline inspection program	Baseline and supplemental inspection (IP 95001)	Baseline and supplemental inspection (IP 95002)	Baseline and supplemental inspection (IP 95003)		Baseline and supplemental as practicable; Special inspections per restart checklist.
	Regulatory Actions[2]	None	Supplemental inspection only	Supplemental inspection only; Plant discussed at AARM if conditions met	10 CFR 2.204 DFI; 10 CFR 50.54(f) letter; CAL/Order; Plant Discussed at AARM	Order to modify, suspend, or revoke license; Plant discussed at AARM	CAL/Order requiring NRC approval for restart; Plant discussed at AARM
COMMUNICATION	Assessment Letters	Branch Chief or Division Director reviews and signs assessment letter w/ inspection plan	Division Director reviews/signs assessment letter w/ inspection plan	Regional Administrator reviews/signs assessment letter w/ inspection plan	Regional Administrator reviews/signs assessment letter w/ inspection plan		N/A. RA or 0350 Panel Chairman review/ sign 0350-related correspondence
	Annual Involvement of Public Stakeholders	Various public stakeholder options involving the senior resident inspector or Branch Chief	Various public stakeholder options involving the BC or DD	Regional Administrator or designee discusses performance with senior licensee management	EDO/DEDO or designee discuss performance with senior licensee management		N/A. 0350 Panel Chairman conducts periodic public status meetings
	External Stakeholders[3]	None	State Governors	State Governors, DHS, Congress	State Governors, DHS, Congress	State Governors, DHS, Congress	
	Commission Involvement	None	None	Possible Commission meeting if licensee remains for 3 years	Commission meeting with senior licensee management within 6 months.[4]	Commission meeting with senior licensee management	Commission meetings as requested; Restart approval in some cases.
	INCREASING SAFETY SIGNIFICANCE →						

Inspection Manual Chapter 305: http://www.nrc.gov/docs/ML1531/ML15317A147.pdf

Precursor Analysis

- Program recommended by WASH-1400 review group (1978)
- Program Objectives:
  - Provide a comprehensive, risk-informed view of nuclear plant operating experience and a measure for trending core-damage risk.
  - Provide a partial validation of the current state of practice in risk assessment.
  - Provide feedback to regulatory activities.
  - Input to the NRC’s Abnormal Occurrence Report
- Supports performance measurement for the NRC’s performance budget

Precursor Analysis

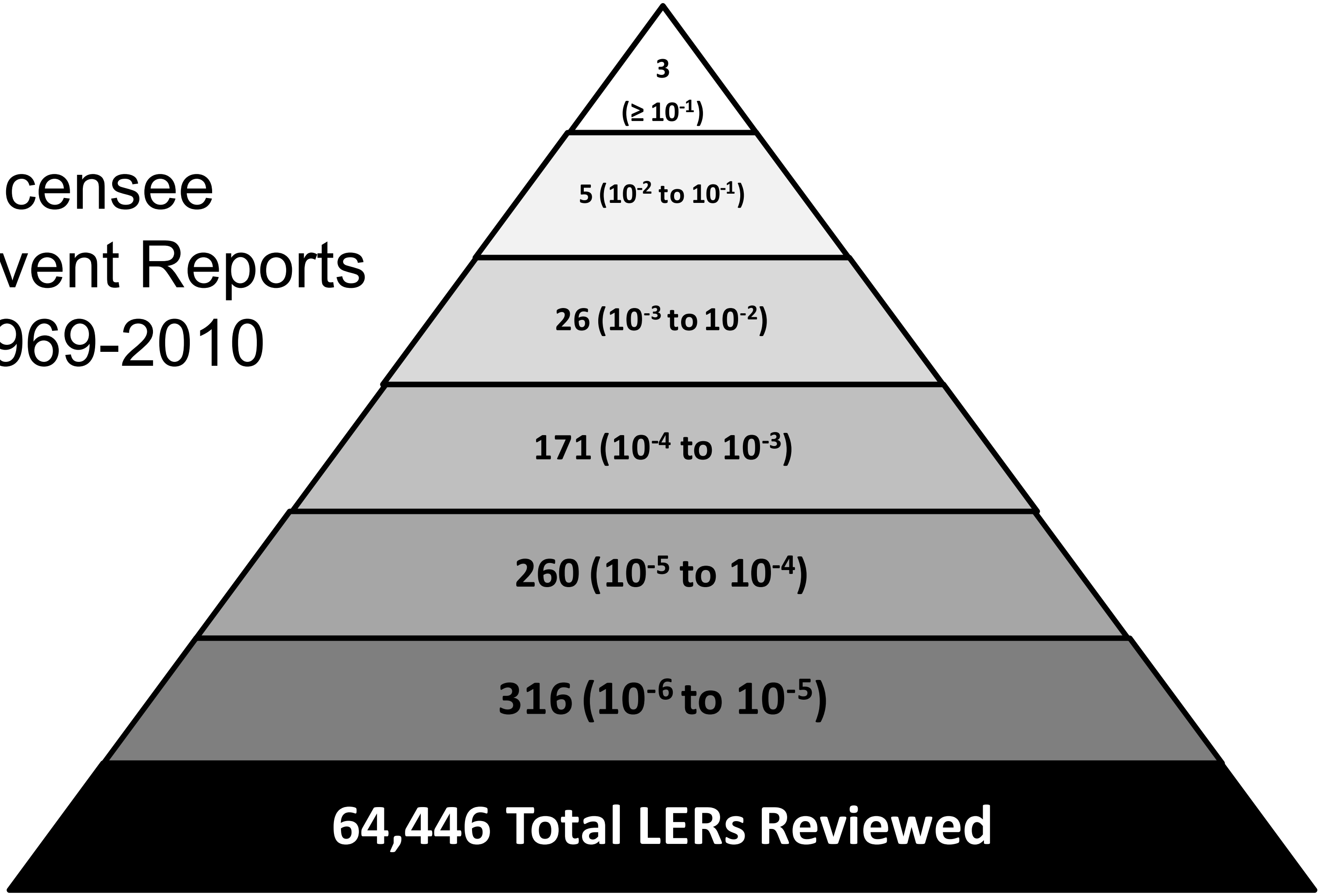
- What is an accident sequence precursor?
  - An observed event and/or condition at a plant, when combined with one or more postulated events (e.g., equipment failures, human errors), could result in core damage.
  - Conditional core damage probability (CCDP) or increase in core damage probability (ΔCDP) $\geq 10^{-6}$.
- What is an ASP analysis?
  - A plant-specific risk analysis performed to determine the conditional likelihood of a core damage accident given an initiating event and/or plant equipment failures or unavailability.
    - Concurrent events and/or failed conditions are set to “TRUE” in the risk model.
    - Observed successes are generally set at their nominal frequency or failure probability
    - This is known as the “failure memory” approach

Precursor Analysis

- **Precursor** - CCDP or ΔCDP greater than or equal to 10^{-6}
- **Significant Precursor** - CCDP or ΔCDP greater than or equal to 10^{-3}
- Precursor Subgroups and Trending
  - Important Precursors - CCDP or ΔCDP greater than or equal to 10^{-4}
  - initiating events
  - degraded conditions
  - loss of offsite power initiating events
  - precursors at boiling-water reactors (BWRs) and pressurized-water reactors (PWRs)

Precursor Analysis

Licensee
Event Reports
1969-2010



Precursor Analysis

- Top Precursors:

Plant	Description	*CCDP*	Event Date	Plant Type	Vendor
Three Mile Island 2	LOFW AND PORV FAILS OPEN. MAJOR ERRORS. CORE MELTDOWN. TMI2	1.E-0	03/28/1979	PWR	BW
Browns Ferry 1	CABLE TRAY FIRE CAUSED EXTENSIVE DAMAGE, TEST ERROR. BIG FIR	1.E-1	03/22/1975	BWR	GE
Rancho Seco	FAILURE OF NNI & SG DRYOUT, ERROR SHORTS NNI-Y. ICS FAILS. L	1.E-1	03/20/1978	PWR	BW
Davis-Besse	STUCK OPEN PORV, FALSE SFRCS/MWFP TRIP. OPERATOR NOT SEE POR	7.E-2	09/24/1977	PWR	BW
Turkey Point 3	FAILURE OF 3 AUX FEEDWATER PUMPS TO START AT TEST	1.E-2	05/08/1974	PWR	WE
Millstone 2	LOOP FROM GRID DISTURBANCE. ERRORS IN DG LOADING FAIL ECCS.	1.E-2	07/20/1976	PWR	CE
Salem 1	LOSS OF VITAL BUS AND SCRAM. 2AFW,1DG,1CP,1RHR FAIL.	1.E-2	11/27/1978	PWR	WE
Davis-Besse	LOFW/SCRAM/FALSE SFRCS. ERROR FAIL AFW/PORV FAIL OPEN/HPI F/	1.E-2	06/09/1985	PWR	BW

Precursor Analysis

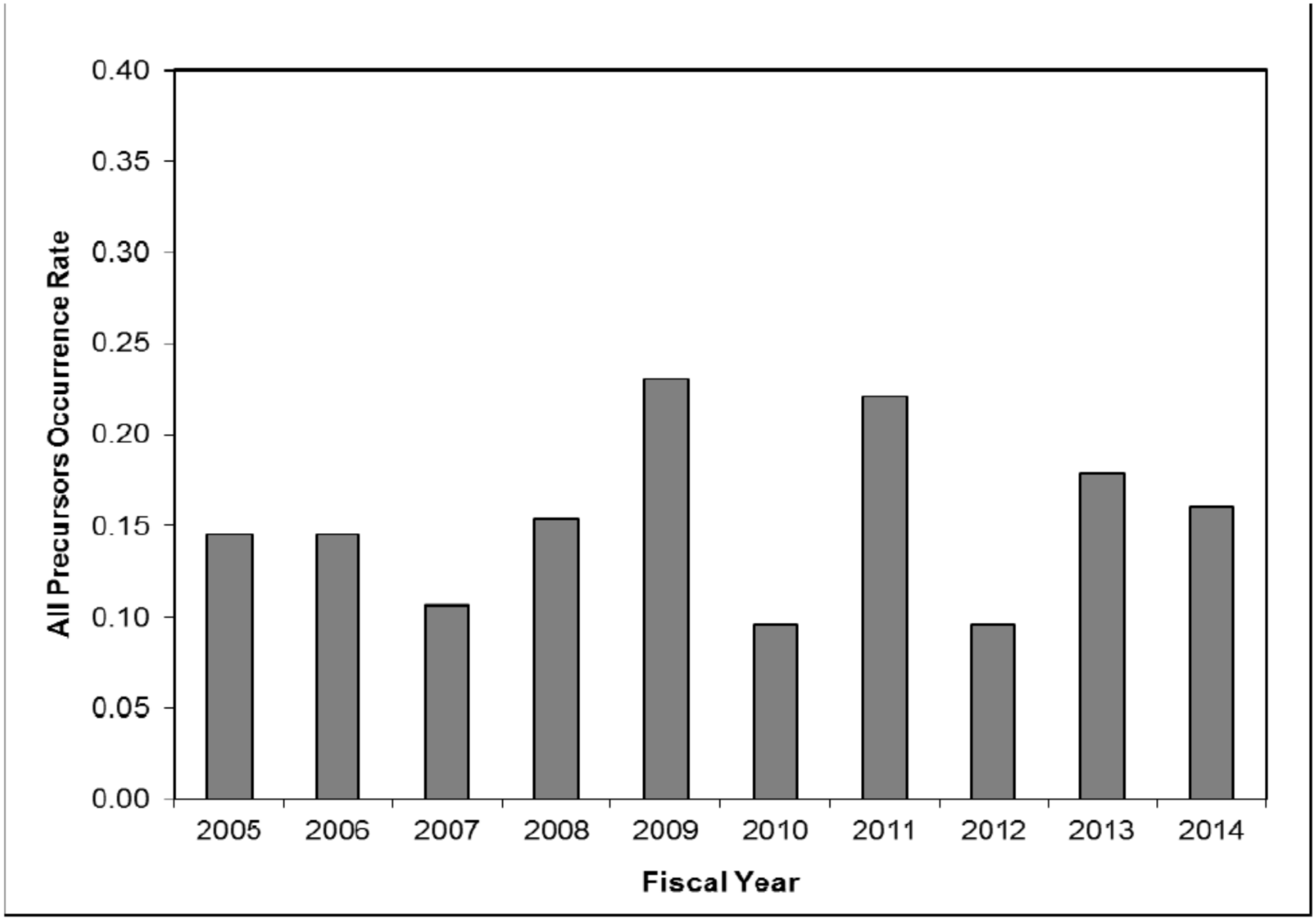
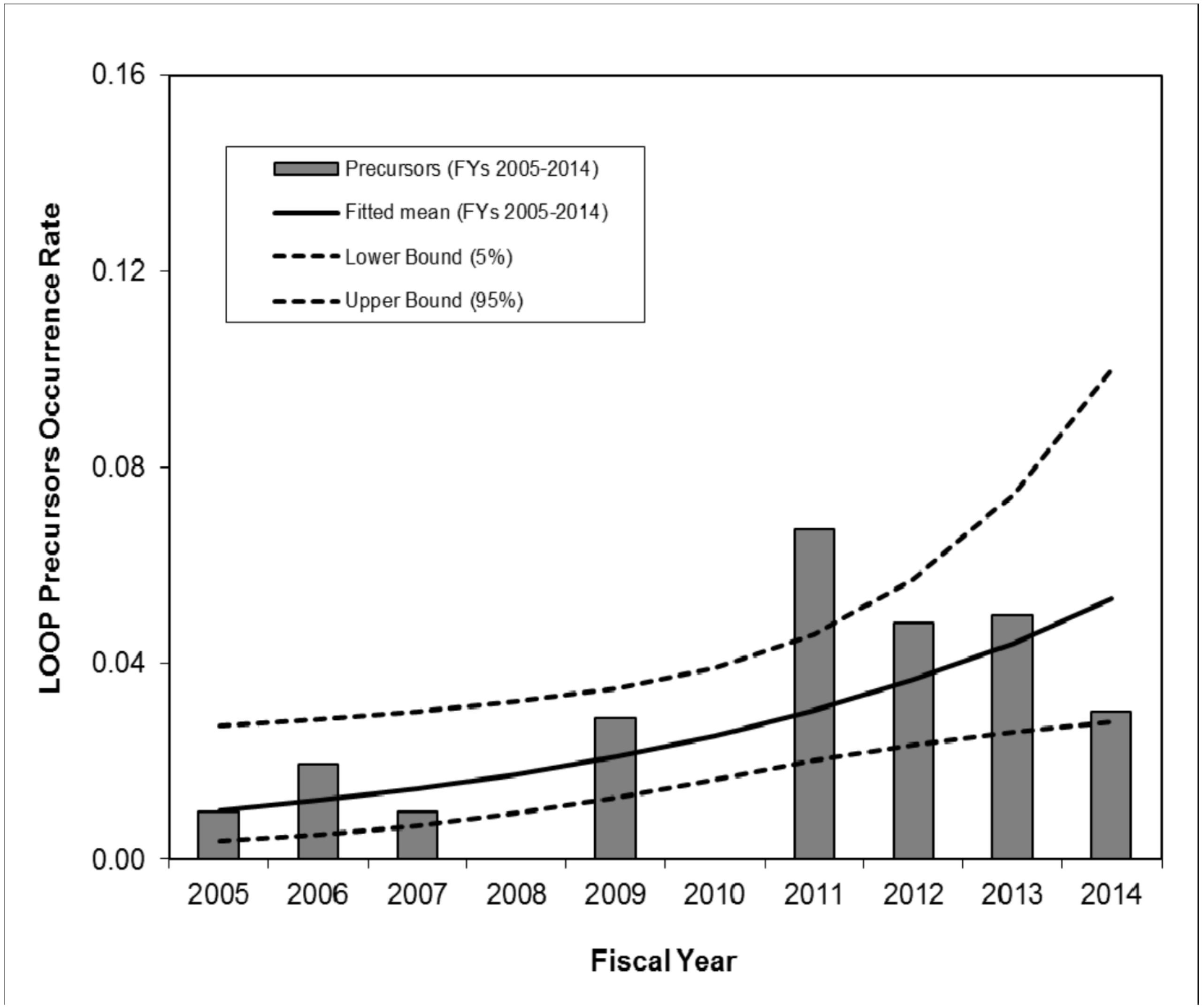


Figure 2. Occurrence Rate of All Precursors.

SECY 15-0124, http://www.nrc.gov/docs/ML1518/ML15187A434.html

Precursor Analysis



SECY 15-0124, http://www.nrc.gov/docs/ML1518/ML15187A434.html

Precursor Analysis

Why is Precursor Analysis Important?

- **Davis-Besse (9/24/1977)**
  - Partial loss of feedwater; stuck-open pressurizer PORV; operators failed to recognize stuck-open PORV (*LER 346/77-016*).
  - CCDP = 7×10^{-2} (4th highest risk ASP event)
- **Three Mile Island, Unit 2 (3/28/1979)**
  - Total loss of feedwater; stuck-open pressurizer PORV; operators failed to recognize stuck-open PORV; other operator errors led to core damage (*LER 320/79-012*).
  - CCDP = 1

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Risk Tools

- Standardized Plant Analysis Risk (SPAR) Model
  - 79 plant-specific models using standardized modeling conventions (covers all operating reactor sites)
  - All models cover level 1, internal events, at-power
  - Some external hazard, fire, shutdown, and level 2 capability
- SAPHIRE PRA Code
  - Event tree/fault tree code
  - Designed to support event and condition analysis (SDP and ASP)
  - Enhanced capabilities for Level 1 and Level 2 integration, enhanced computational approaches, and report generation

Outline

- Origins and Mission of the NRC
- Historical NRC PRA Studies and Foundations
- Policy
- Risk-Informed Regulation
  - Regulatory Analysis & Backfitting
  - Licensing
  - Oversight
- **Research Activities**

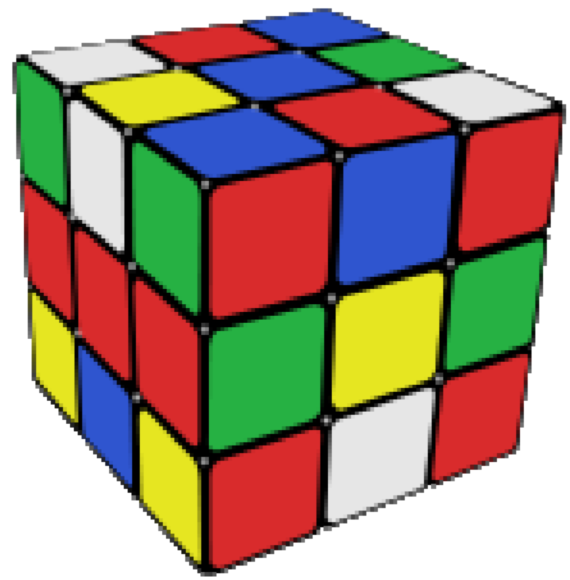
Research Activities

Vogtle Integrated Site Level 3 PRA Project

- Considers all reactor cores, spent fuel pools, and dry storage casks on site, all internal and external hazards, and all modes of reactor operation
- Incorporates improvements in PRA technology and changes in plant operational performance and safety since completion of NUREG-1150
- Study will be for a single multi-unit site; results in some limits in the generally applicability of risk insights

Research Activities

- Scope – all modes, all hazards
  - Reactor, at-power, Level 1
    - Internal events and floods
    - Internal fires
    - Seismic events
    - High winds, external flooding, and
    - other hazards
  - Reactor, at-power, Level 2
  - Reactor, at-power, Level 3
  - Reactor, low power and shutdown
  - Spent fuel pool (SFP)
  - Dry cask storage (DCS)
  - Integrated site risk



Research Activities

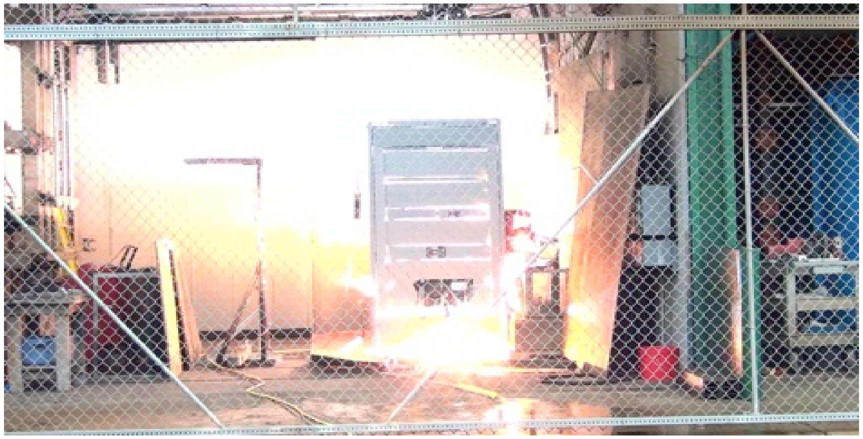
Human Reliability

- Operator performance data collection
  - Methodology and software developed (SACADA: Scenario Authoring, Characterization, and Debriefing Application) to collect operator simulator exercise data
  - Future work: expand the methodology to collect data of real events, emergency drills, and outside control room actions
- Generic human reliability analysis (HRA) methodology
  - Developing a single HRA methodology for all radiation sources, hazards, PRA levels, and NRC's risk-informed applications

Research Activities

Fire Research

- Research will improve state of knowledge of fire phenomena and risk assessment practices
  - reduce uncertainty, improve parameter estimates, and enhance modeling
  - Characterize Fire (intensity, growth, duration)
  - Determine impact (smoke, hot gas layer)
  - Assess equipment functionality
- Fire Dynamics
  - Cable heat release and ignition
  - High Energy Arcing Fault
- Equipment Functionality
  - Post-fire impact on cabling
  - Main Control Room abandonment



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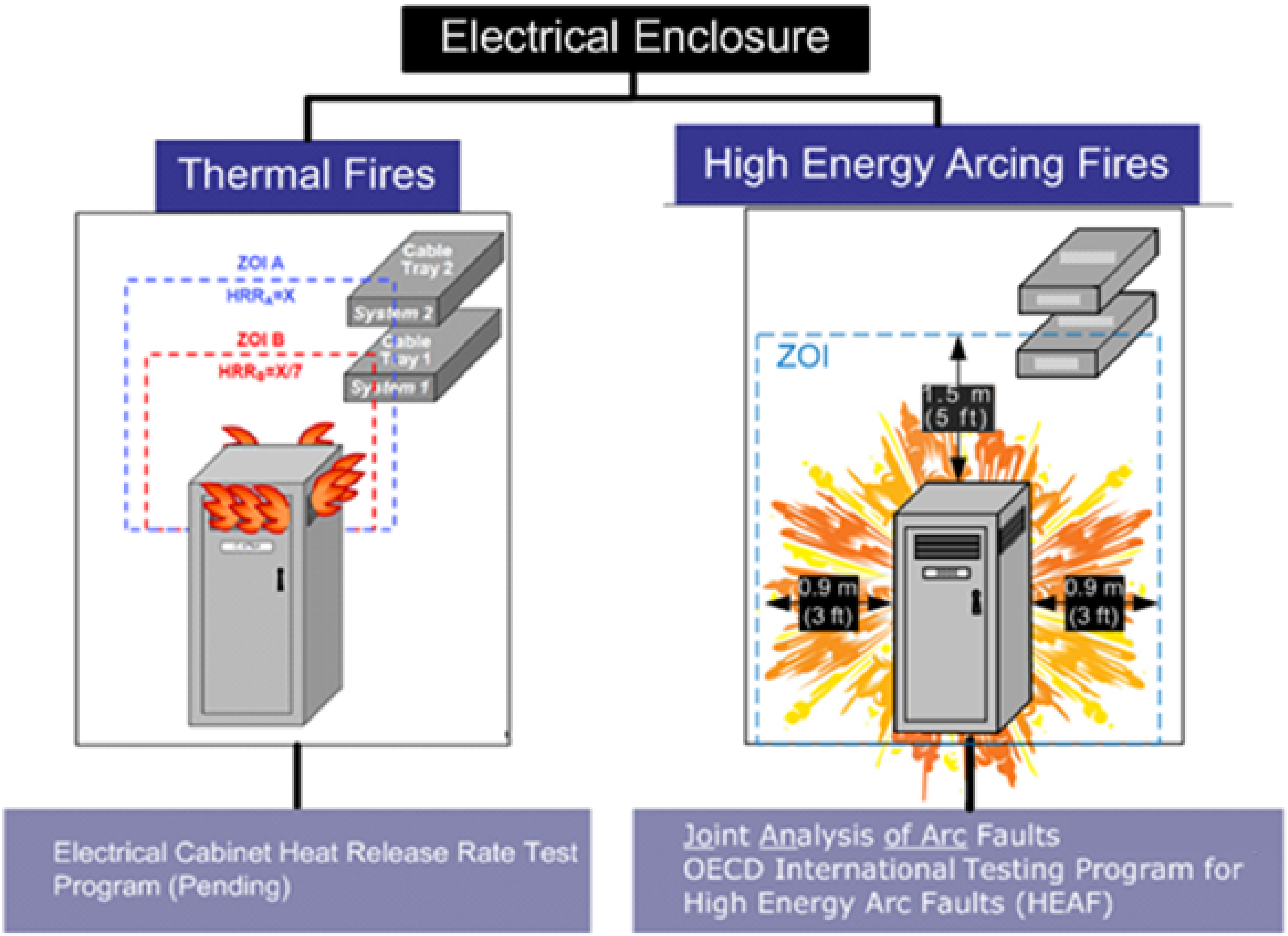
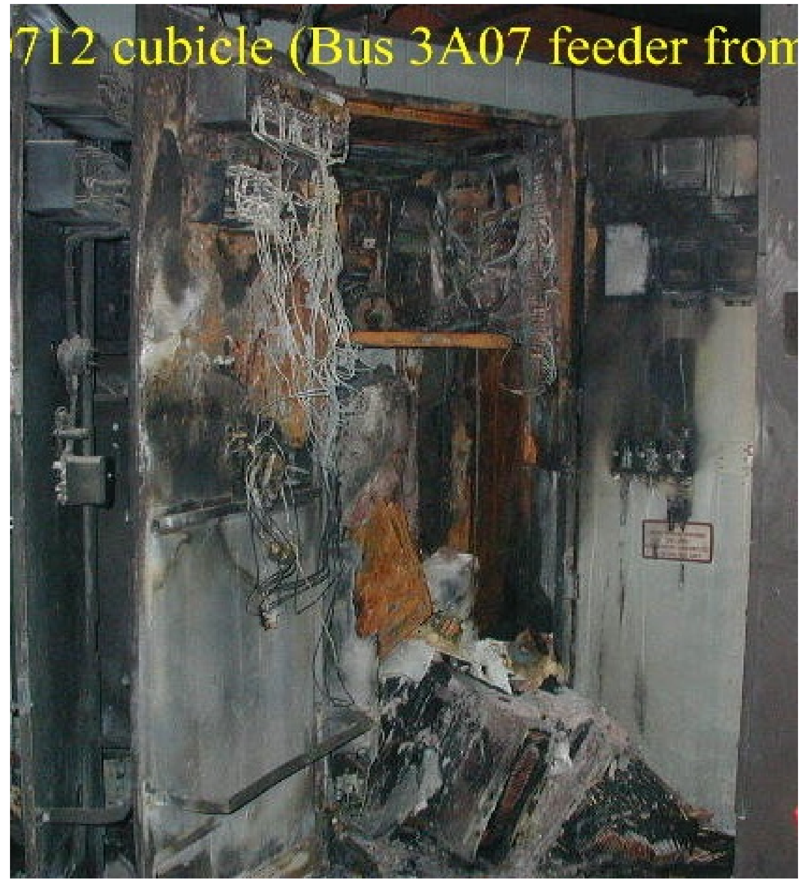
Research Activities



Research Activities

Recent High Energy
Arc Fault Events



Concluding Thoughts…

- Risk-informed regulatory approaches can benefit both the regulator and industry
  - Focus attention on issues most pertinent to public health
  - Help ensure regulatory response is commensurate with safety impact
  - Provides a framework to organize complex information
  - Can foster a more open and transparent regulatory process

More Information…

- NRC History - http://www.nrc.gov/about-nrc/history.html
- NRC Document Collections - http://www.nrc.gov/reading-rm/doc-collections/
- Commission Policy Statements - http://www.nrc.gov/reading-rm/doc-collections/commission/policy/
- Nuclear Plant Oversight - http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/
- Risk Informed, Performance Based Plan (RPP) - http://www.nrc.gov/about-nrc/regulatory/risk-informed/rpp.html