9. Sequence Quantification

Sequence Quantification

- Purpose: This topic will provide students with an understanding of the quantitative basis of PRA. Elements of accident sequence quantification and importance analysis will be presented.
- Objectives: At the conclusion, students will be able to:
  - ✧ Describe the major processes for accident sequence quantification
  - ✧ Explain the concepts of importance analysis
- References: NUREG/CR-2300, NUREG-1489 (App. C)

Quantification Inputs

- Initiating events and frequencies
- Event trees to define accident sequences
- Fault trees and Boolean expressions for all systems (front line and support)
- Data (component failures and human errors)

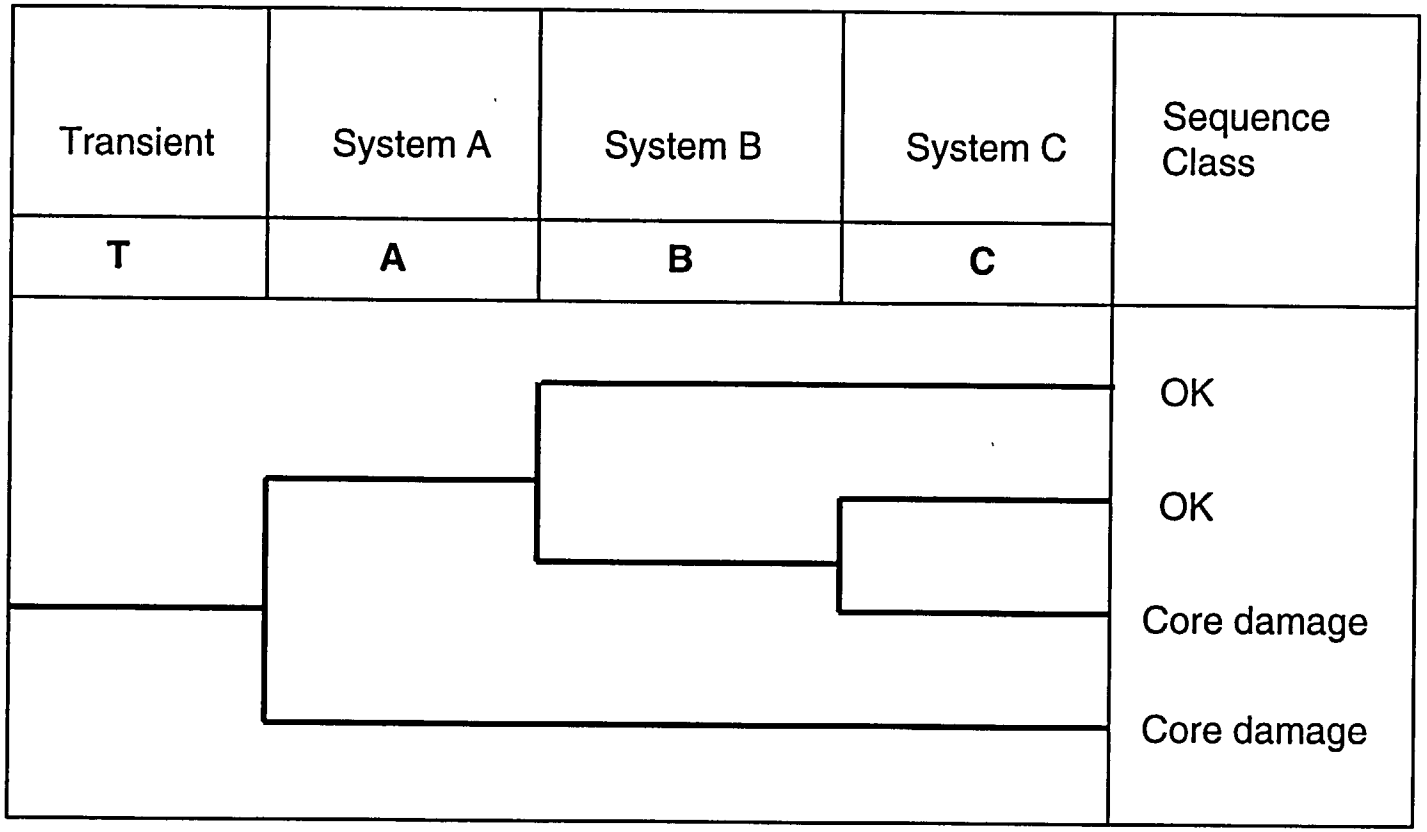
Parameter Inputs for Sequence Quantification

- Initiating event frequencies
  - λ_{IE}
- Demand failures
  - $Q_d = p$
- Mission time failures (failure to run)
  - $Q_r \approx \lambda_h t_m$
- Standby failures
  - $Q_s \approx \lambda_s t_i/2$
- Test and maintenance unavailability
  - $Q_m = \lambda_m d_m$
- Common-cause parameters
  - β

Fault-Tree Linking Approach to Accident Sequence Quantification

- Link fault tree models on sequence level using event trees
- Evaluate each sequence for minimal cut sets (Boolean reduction)
- Quantify sequence minimal cut sets with data
- Add operator recovery actions and common cause failures
- Determine dominant accident sequences
- Place in plant damage state bins
- Perform sensitivity, importance, and uncertainty analysis

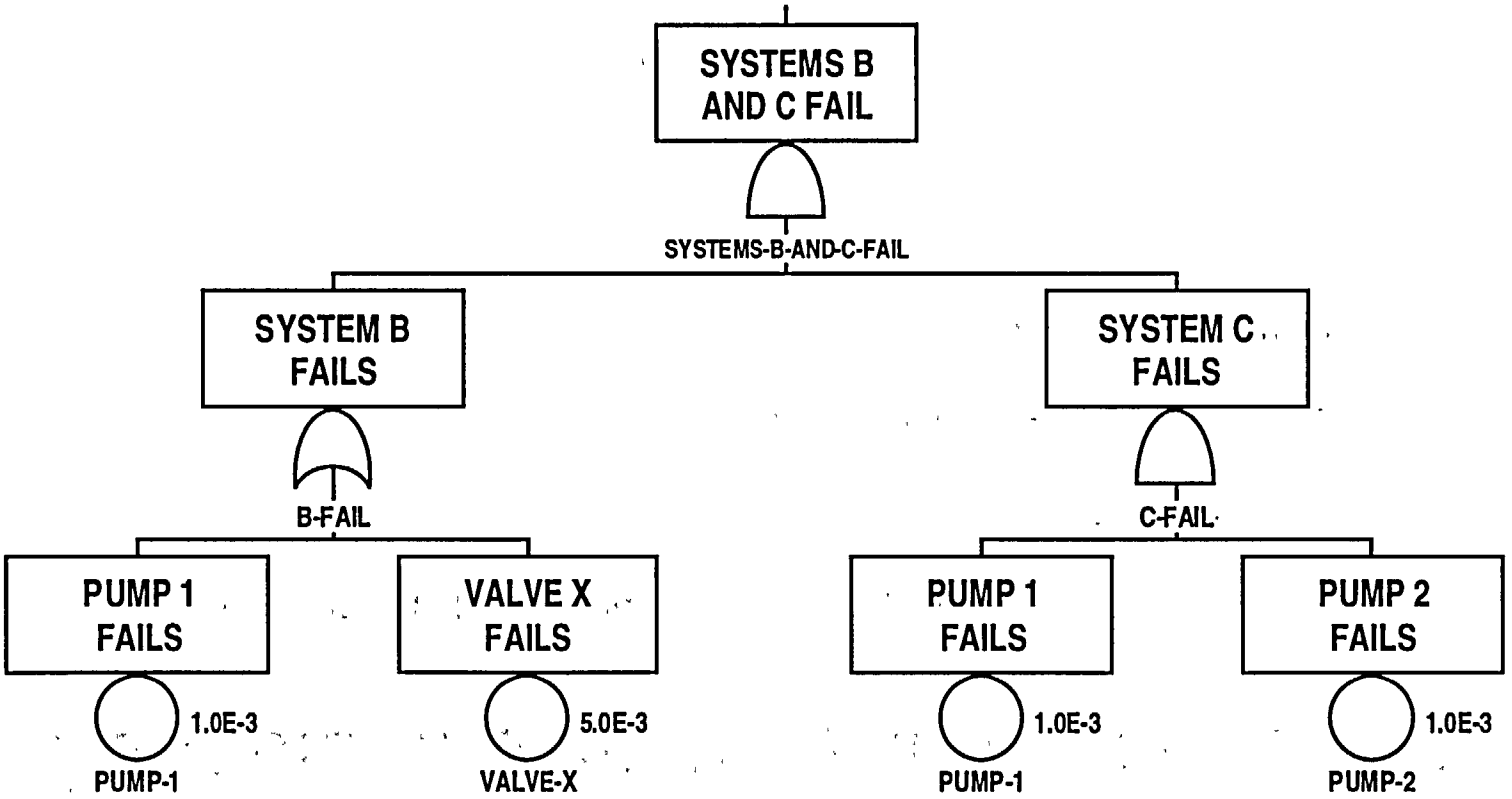
Example of Quantification Process



.....Let's look at Sequence **TBC**

Example of Quantification Process (cont.)

T = 10 transients (demands) / year



Example of Quantification Process (cont.)

Systems B AND C Fail = System B Fails * System C Fails

= (Pump 1 + Valve X) * (Pump 1 * Pump 2)

= (Pump 1 * Pump 1 * Pump 2) + (Valve X * Pump 1 * Pump 2)

= (Pump 1 * Pump 2) + (Valve X * Pump 1 * Pump 2)

= Pump 1 * Pump 2

= (1E-3) (1E-3)

= 1E-6 (Probability)

Sequence TBC = T * System B Fails * System C Fails

= 10/Year * 1E-6

= 1E-5/Year (Frequency)

Recovery Analysis

- Analysis on accident sequence level
  - Examination of contributors to failure
  - Identification of potential for recovery
- Recovery factors
  - Critical time for recovery
  - Action required
  - Time for action
  - Time versus probability of recovery
- Final accident sequence frequency includes recovery

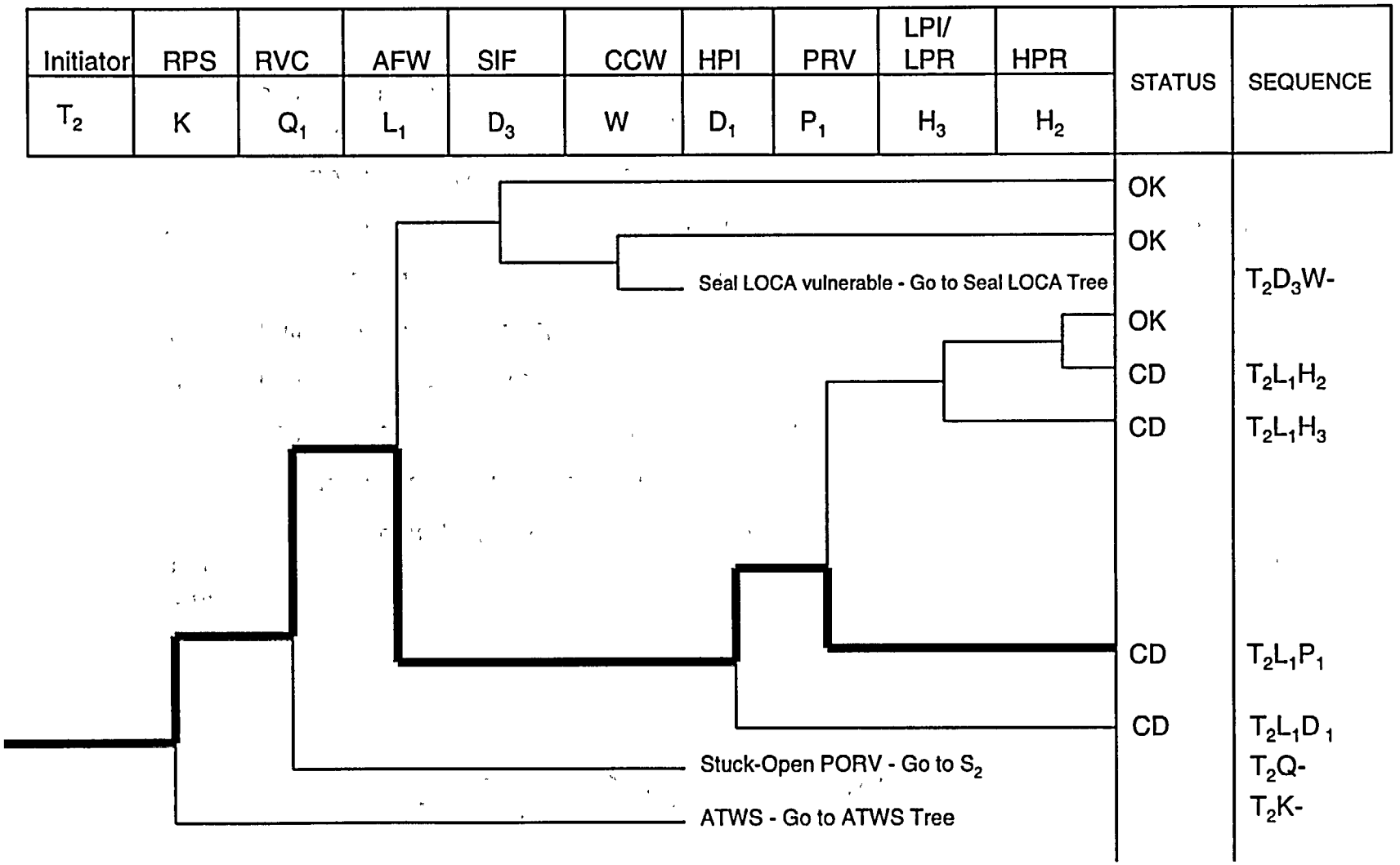
Summary of Sequence $T_2L_1P_1$

- This sequence is initiated by a loss of main feedwater (T2), followed by failure of the auxiliary feedwater (AFW) system, and failure of feed and bleed cooling due to the inability to open both power operated relief valves (PORVs).

- The loss of main feedwater initiator places a demand on auxiliary feedwater to remove core decay heat. Failure of the AFW system causes a demand for feed and bleed cooling. Failure to initiate feed and bleed and various failures which prevent one of the two PORVs from opening contribute to this sequence. Success criteria require that two PORVs open for successful feed and bleed.

- The dominant contributors to AFW failure are common cause failure of the air-operated steam generator level control valves and the common cause failure of all three AFW pumps due to steam binding. The dominant contributor to failure of feed and bleed is operator failure to open PORVs, followed by mechanical failures of the PORV block valves and PORVs.

Event Tree for T_2 - Loss of Main Feedwater



Identifiers for T_2 Event Tree

Event Identifier	Description	System Identifier
D_1	Failure of charging pump system with 1 of 4 success requirements	HPI
D_3	Failure of charging pump system in seal injection flow mode	SIF
H_2	Failure of charging pump system in the high pressure recirculation mode	HPR
H_3	Failure of low pressure injection/recirculation	LPI/LPR
K	Failure of reactor protection system	RPS
L_1	Failure of auxiliary feedwater required for transients with reactor trip	AFW
P_1	Failure of both pressurizer PORVs to open for feed & bleed	PRV
Q_1	Failure of any relief valve to reclose	RVC
W	Failure of component cooling water to the thermal barrier of all reactor coolant pumps	CCW

Dominant Contributors to Sequence $T_2L_1P_1$

Minimal Cut Set	Minimal Cut Set Frequency
T_2 * AFW-AOV-CC * BETA-8AOV * HPI-XHE-FO-FDBLD	5.4E-7
T_2 * STEAM-BINDING * HPI-XHE-FO-FDBLD	1.6E-7
T_2 * AFW-AOV-CC * BETA-8AOV * PPS-SOV-FT-334	1.6E-7
T_2 * AFW-AOV-CC * BETA-8AOV * PPS-SOV-FT-340A	1.6E-7
T_2 * AFW-TDP-FS-1AS * AFW-MDP-FS * BETA-AFW * HPI-XHE-FO-FDBLD	8.0E-8
T_2 * AFW-TDP-FR-1AS6H * AFW-MDP-FS * BETA-AFW * HPI-XHE-FO-FDBLD	8.0E-8
T_2 * STEAM-BINDING * PPS-SOV-FT-334	4.6E-8
T_2 * STEAM-BINDING * PPS-SOV-FT-340A	4.6E-8
T_2 * AFW-ACT-FA-TRNA * AFW-ACT-FA-TRNB * HPI-XHE-FO-FDBLD	4.1E-8
T_2 * AFW-TDP-TM-1AS * AFW-MDP-FS * BETA-AFW * HPI-XHE-FO-FDBLD	2.7E-8
Total $T_2L_1P_1$	**1.3E-6**

Term Descriptions

T_2	Loss of main feedwater	7.2E-1/reactor year
STEAM-BINDING	Steam-binding of all AFWS pumps	1.0E-5
PPS-SOV-FT-334	PORV 334 fails to open	6.3E-3
PPS-SOV-FT-340A	PORV 340A fails to open	6.3E-3
AFW-TDP-FS-1AS	AFWS turbine pump fails to start	3.0E-2
AFW-TDP-FR-1AS6H	AFWS turbine pump fails to run 6 hours	3.0E-2
AFW-TDP-TM-1AS	AFWS turbine pump unavailable test and maintenance	1.0E-2
AFW-AOV-CC	AFWS AOV fails to open	1.0E-3
BETA-AFW	Common cause failure factor of 2 motor pumps	5.6E-2
BETA-8AOV	Common cause failure factor of 8 AOVs	3.4E-2
AFW-MDP-FS	AFWS motor pump fails to start	3.0E-3
HPI-XHE-FO-FDBLD	Operator fails to initiate feed and bleed	2.2E-2
AFW-ACT-FA-TRNA	AFWS Train A actuation fails	1.6E-3
AFW-ACT-FA-TRNB	AFWS Train B actuation fails	1.6E-3

Importance Measures

- Provide quantitative perspective on dominant contributors to risk and sensitivity of risk to changes in input values
- Usually calculated at core damage frequency level
- Three are encountered most commonly:
  - Fussell-Vesely
  - Risk Reduction
  - Risk Increase or Risk Achievement

Fussell-Vesely Importance

- Measures overall contribution of an event to risk (CDF)
- Calculated by adding up frequencies of cutsets containing event of interest and dividing by total CDF

  $FV_x = \sum$Cutsets with event x / F(x)

  or

  $FV_x = [F(x) - F(0)] / F(x)$

  where,

  F(x) is risk with event x at nominal failure probability, and

  F(0) is risk when event x is never failed (failure probability = 0)

- Range is from 0 to 1

Fussell-Vesely Importance (cont.)

- Consider these minimal cut sets:

$A = 6 \times 10^{-4}$ $= 6 \times 10^{-4}$

$B * C = 1 \times 10^{-2} * 3 \times 10^{-3} = 3 \times 10^{-5}$

$C * D = 3 \times 10^{-3} * 1 \times 10^{-3} = 3 \times 10^{-6}$

$F_{(x)} = 6.33 \times 10^{-4}$

where,

$A = 6 \times 10^{-4}$

$B = 1 \times 10^{-2}$

$C = 3 \times 10^{-3}$

$D = 1 \times 10^{-3}$

- Fussell-Vesely Importance

$FV_A = 6.0 \times 10^{-4}/6.33 \times 10^{-4} = 0.948$

$FV_B = 3.0 \times 10^{-5}/6.33 \times 10^{-4} = 0.047$

$FV_C = 3.3 \times 10^{-5}/6.33 \times 10^{-4} = 0.052$

$FV_D = 3.0 \times 10^{-6}/6.33 \times 10^{-4} = 0.005$

Risk Reduction Importance

- Measures amount by which CDF would decrease if event's failure probability were set to 0 (never fails)
- Calculated as either ratio or difference between baseline CDF and CDF with event failure probability at 0

  Ratio: RRR(x) = F(x)/F(0)

  Difference (or Interval): RRI(x) = F(x) - F(0)

  where,

  F(x) is risk with event x at nominal failure probability, and

  F(0) is risk when event x is never failed (failure probability = 0)
- Ratio - Range is from 1 to ∞
- Gives same ranking as Fussell-Vesely
- For Maintenance Rule (10 CFR 50.65), NUMARC Guide 93-01 (endorsed by NRC) uses a RRR significance criterion of 1.005
  - ✧ Equivalent to Fussell-Vesely importance of 0.005

Risk Reduction Importance (cont.)

- Consider these minimal cut sets:

$A = 6 \times 10^{-4} = 6 \times 10^{-4}$

$B * C = 1 \times 10^{-2} * 3 \times 10^{-3} = 3 \times 10^{-5}$

$C * D = 3 \times 10^{-3} * 1 \times 10^{-3} = 3 \times 10^{-6}$

$F_{(x)} = 6.33 \times 10^{-4}$

where,

$A = 6 \times 10^{-4}$

$B = 1 \times 10^{-2}$

$C = 3 \times 10^{-3}$

$D = 1 \times 10^{-3}$

- Risk Reduction Ratio Importance

$RRR_A = 6.33 \times 10^{-4}/3.3 \times 10^{-5} = 19.18$

$RRR_B = 6.33 \times 10^{-4}/6.03 \times 10^{-4} = 1.05$

$RRR_C = 6.33 \times 10^{-4}/6.00 \times 10^{-4} = 1.06$

$RRR_D = 6.33 \times 10^{-4}/6.30 \times 10^{-4} = 1.00$

Risk Increase Importance

- Measures amount by which CDF would increase if event's failure probability were set to 1 (e.g., component taken out of service)
- Calculated as either ratio or difference between CDF with event failure probability at 1 and baseline CDF

  Ratio: RAW(x) or RIR(x) = F(1)/F(x)

  Difference (or Interval): RII(x) = F(1) - F(x)

  where,

  F(x) is risk with event x at nominal failure probability, and

  F(1) is risk when event x is always failed (failure probability = 1)
- Ratio measure referred to as risk achievement worth (RAW)
- RAW - Range is ≥ 1
- For Maintenance Rule (10 CFR 50.65), NUMARC Guide 93-01 (endorsed by NRC) uses a RAW significance criterion of 2

Risk Increase Importance (cont.)

- Consider these minimal cut sets:

$A = 6 \times 10^{-4} = 6 \times 10^{-4}$

$B * C = 1 \times 10^{-2} * 3 \times 10^{-3} = 3 \times 10^{-5}$

$C * D = 3 \times 10^{-3} * 1 \times 10^{-3} = 3 \times 10^{-6}$

$F_{(x)} = 6.33 \times 10^{-4}$

where,

$A = 6 \times 10^{-4}$

$B = 1 \times 10^{-2}$

$C = 3 \times 10^{-3}$

$D = 1 \times 10^{-3}$

- Risk Achievement Worth Importance

$RAW_A = 1.0 / 6.33 \times 10^{-4} = 1579.78$

$RAW_B = 3.603 \times 10^{-3}/6.33 \times 10^{-4} = 5.69$

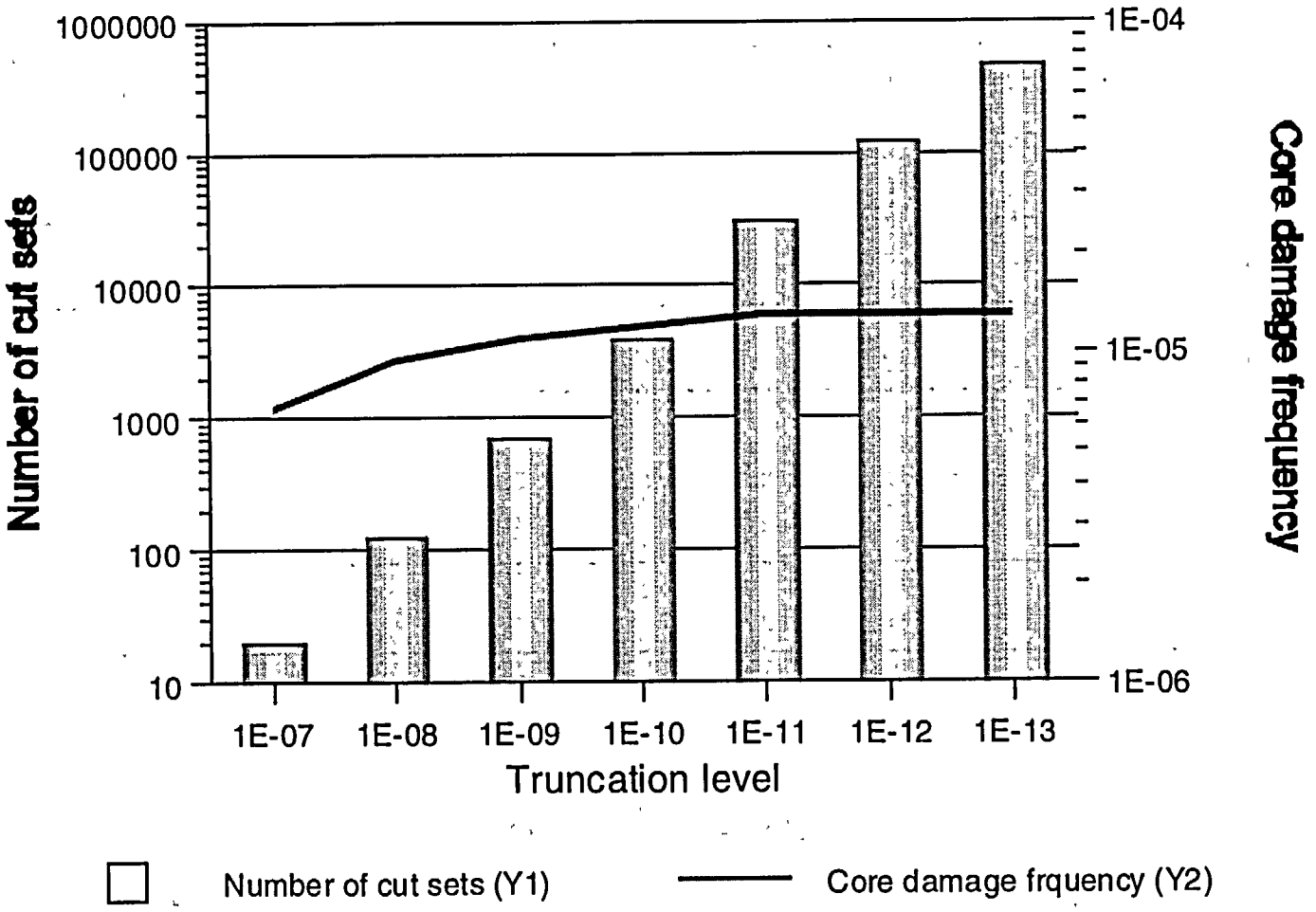
$RAW_C = 1.16 \times 10^{-2}/6.33 \times 10^{-4} = 18.33$

$RAW_D = 3.63 \times 10^{-3}/6.33 \times 10^{-4} = 5.73$

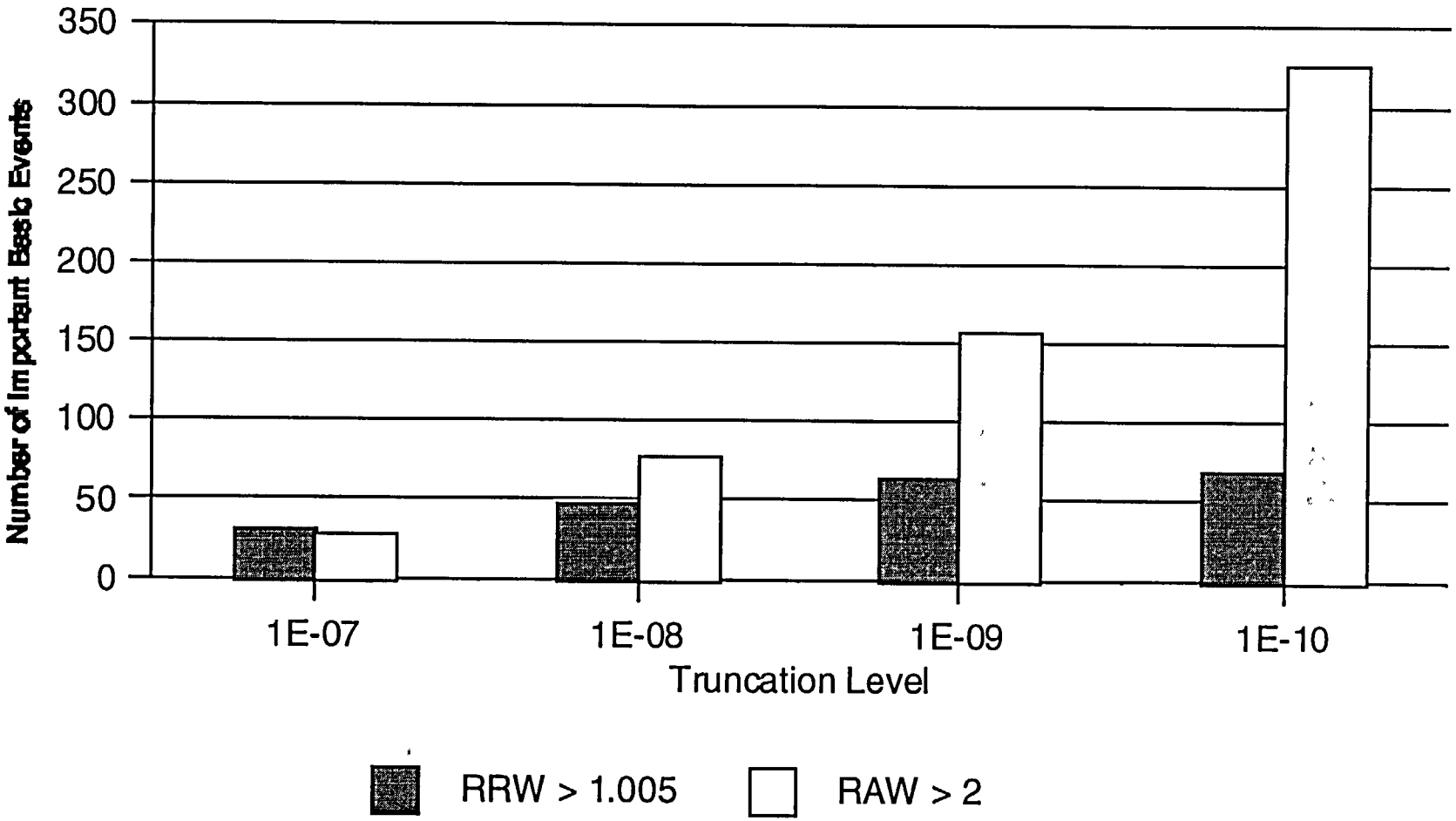
Limitations of Risk Importance Measures

- Numerical values can be affected by:
  - ✧ Exclusion of equipment from PRA model
  - ✧ Model truncation during quantification
  - ✧ Parameter values used for other events in model
  - ✧ Present configuration of plant (equipment that is already out for test/maintenance)

Core Damage Frequency and Number of Cutsets Sensitive to Truncation Limits



Truncation Limits Affect Importance Rankings



Limitations of Risk Importance Measures (cont.)

- Risk rankings are not always well-understood in terms of their issues and engineering interpretations
- RAW provides indication of risk impact of taking equipment out of service but full impact may not be captured
  - ✧ That is, taking component out of service for test and maintenance may increase likelihood of initiating event due to human error

Other Considerations When Using Importance Measures

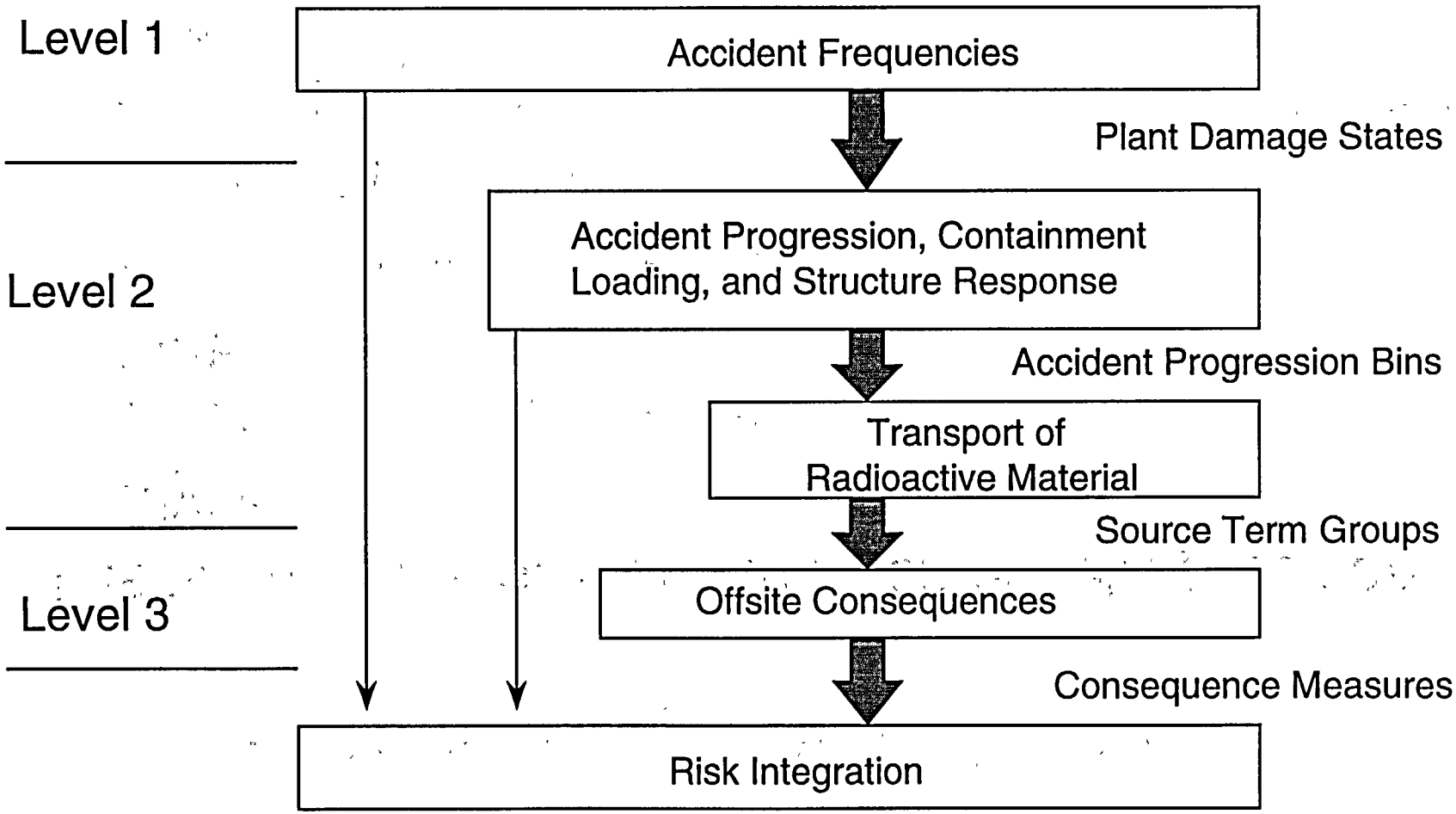
- F-V and RAW rankings can differ significantly when using different risk metrics
  - ✧ Such as, core damage frequency due to internal events versus external events, shutdown risk, etc.
- Individual F-V or RAW measures cannot be combined to obtain risk importance for combinations of events
  - ✧ Critical combinations can be extremely important due to failure of redundant components whereas individual components in one train may have low rankings

10. Accident Progression & Consequence Analysis

Accident Progression Analysis, Containment Response, Fission Product Transport, and Consequence Analysis

- **Purpose:** Students receive a brief introduction to accident progression (Level 2 PRA) and consequence analysis (Level 3 PRA).
- **Objectives:** At the conclusion of this topic, students will be able to:
  - ✧ List primary elements which comprise accident phenomenology
  - ✧ Explain how accident progression analysis is related to full PRA
  - ✧ Explain general factors involved in containment response
  - ✧ Explain general factors involved in fission product transport & consequences
  - ✧ Name the major computer codes used in accident process and consequence analysis
- **Reference:** NUREG/CR-2300, NUREG-1489 (App. C)

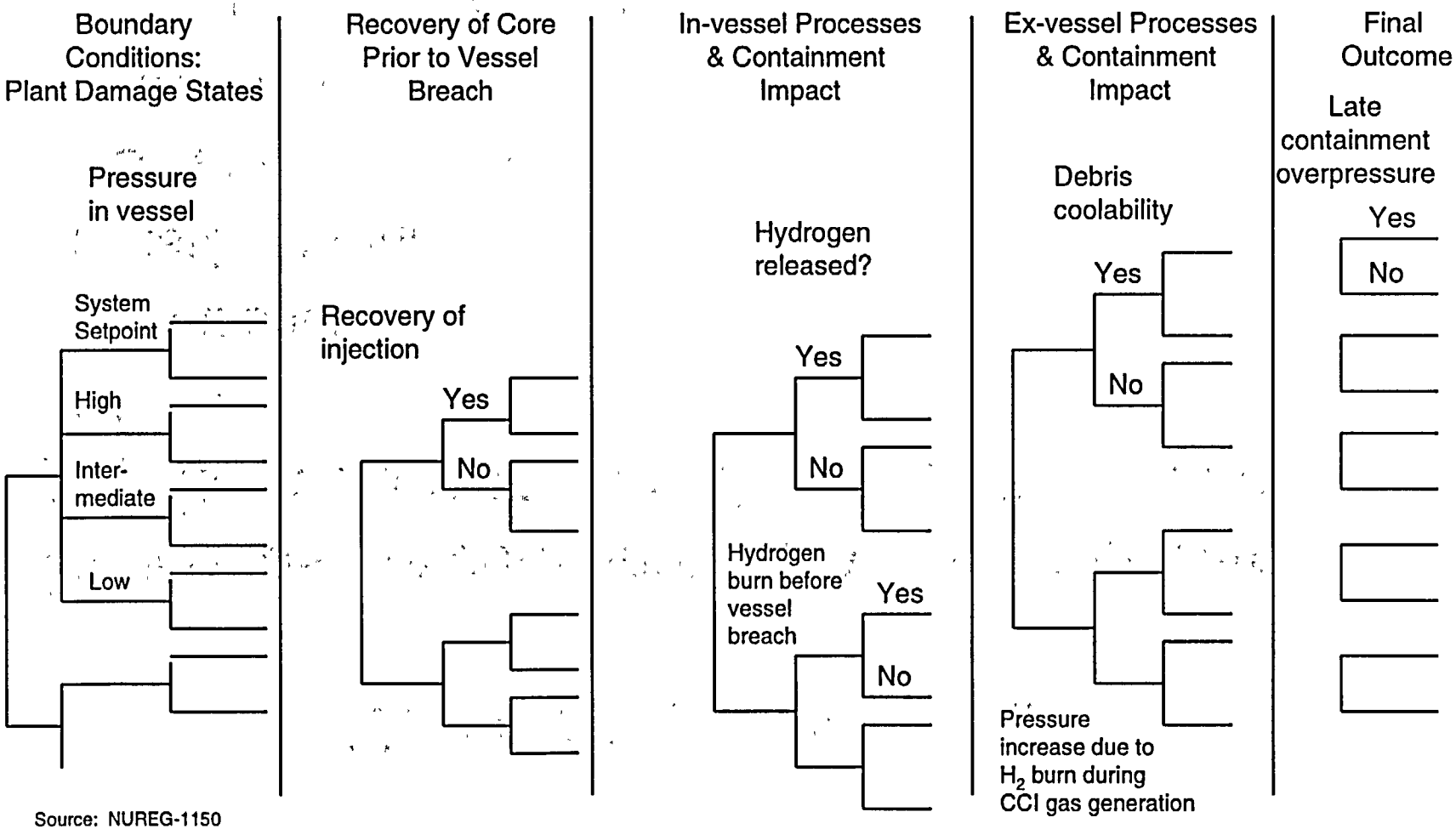
Principal Steps in PRA Process



Accident Progression Analysis

- There are 4 major steps in Accident Progression Analysis
  - 1. Develop the Accident Progression Event Trees (APETs)
  - 2. Perform structural analysis of containment
  - 3. Quantify APET issues
  - 4. Group APET sequences into accident progression bins

Schematic of Accident Progression Event Tree

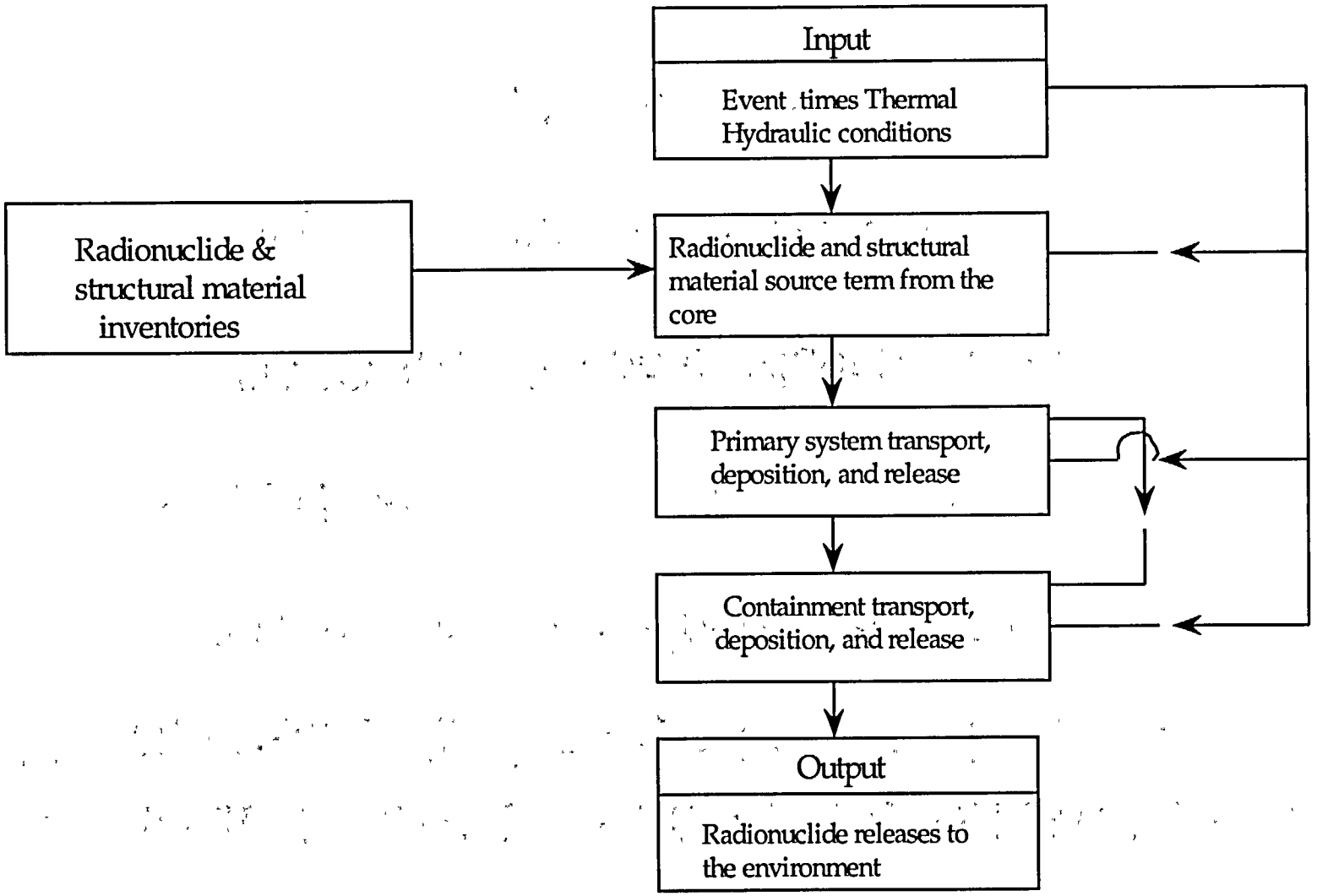


Source: NUREG-1150

Containment Response

- How does the containment system deal with physical conditions resulting from the accident?
  - ✧ Pressure
  - ✧ Heat sources
  - ✧ Fission products
  - ✧ Steam and water
  - ✧ Hydrogen
  - ✧ Other noncondensables

Elements in the Analysis of Radionuclide Behavior in the Reactor



Computer codes used to model Accident Progression & Fission Product Behavior

- RELAP5/SCDAP - in-vessel behavior
- CONTAIN - containment behavior
- VICTORIA - fission product behavior
- Integrated, comprehensive codes
  - ✧ MAAP - industry code
  - ✧ MELCOR - NRC code

Fission Product Source Term Outcomes of Interest

- Fractions Released Outside Containment
  - Noble Gases
  - Iodine
  - Cesium - Rubidium
  - Tellurium - Antimony
  - Barium - Strontium
  - Ruthenium - Molybdenum - Rhenium - Technetium - Cobalt
  - Lanthanum and other rare earth metals
- Parameters for Consequence Model
  - Time of release
  - Duration of release
  - Warning time for evacuation
  - Elevation of release
  - Energy of release

Source Term Calculation Models

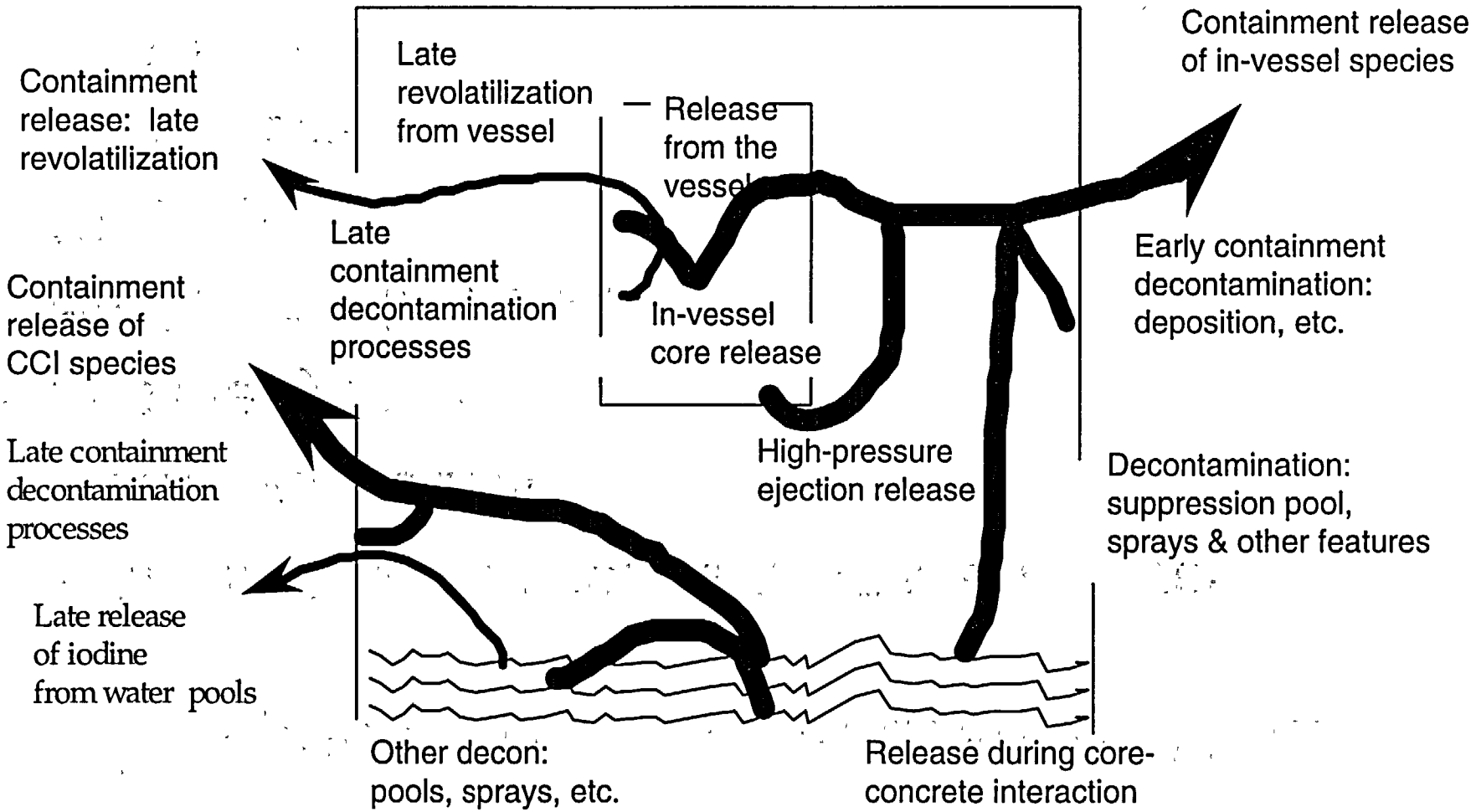
Integrated Deterministic Code (MELCOR)

- Point estimate radionuclide release calculations for scenarios important to risk
- Selected sensitivity calculations to explore uncertainties that can be modeled by the code

Parametric Source Term Code

- Point estimate radionuclide release calculations for scenarios less important to risk (simulation of source code package)
- Extensive sensitivity calculations to explore uncertainties that cannot be modeled by code package

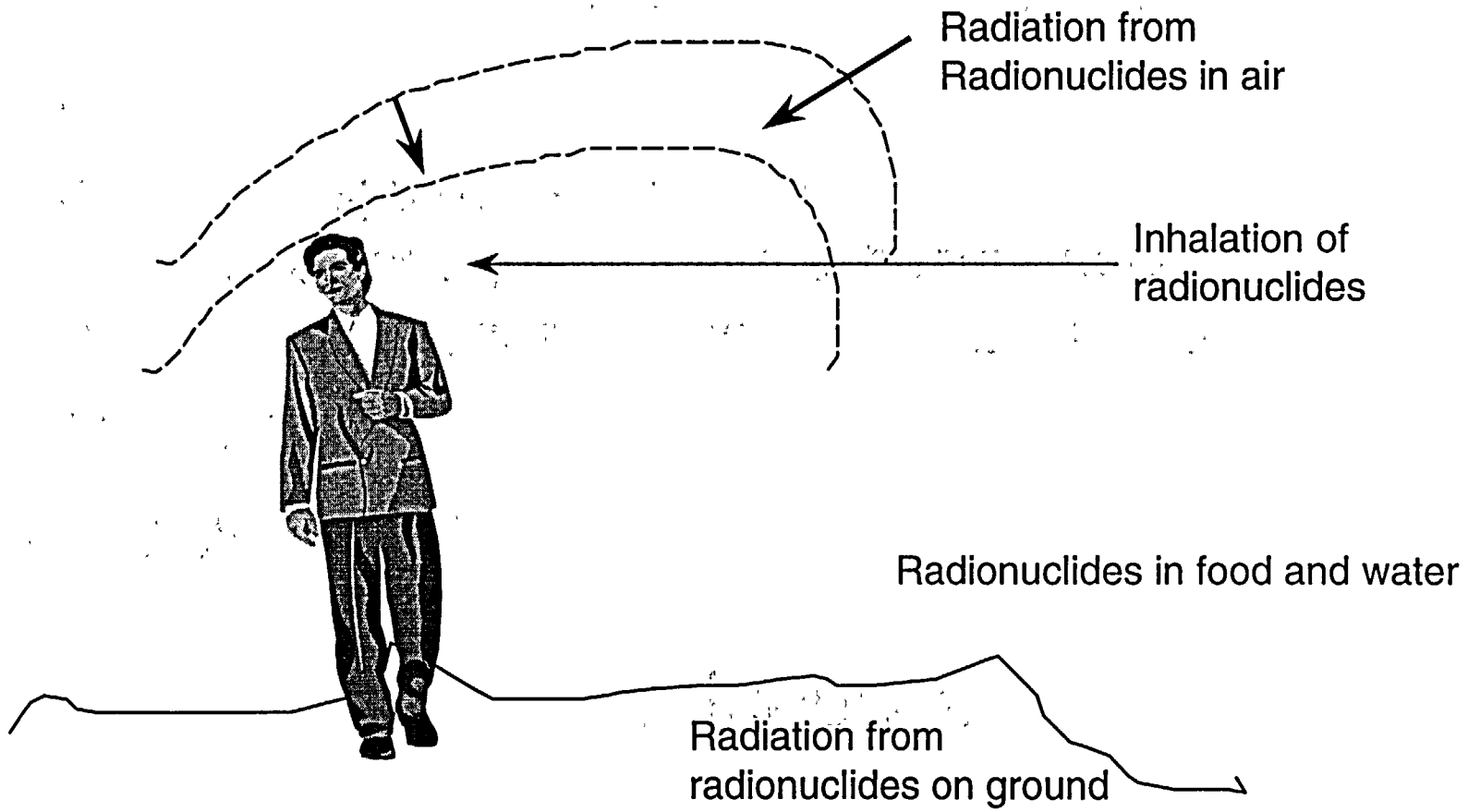
Schematic of Parametric Source Term Algorithm



Components of a Consequence Model

- Atmospheric transport and diffusion model
- Pathways models
- Dosimetry models
- Health effects model
- Other models:
  - ✧ Evacuation
  - ✧ Interdiction
  - ✧ Decontamination
  - ✧ Economic effects

Pathways to People



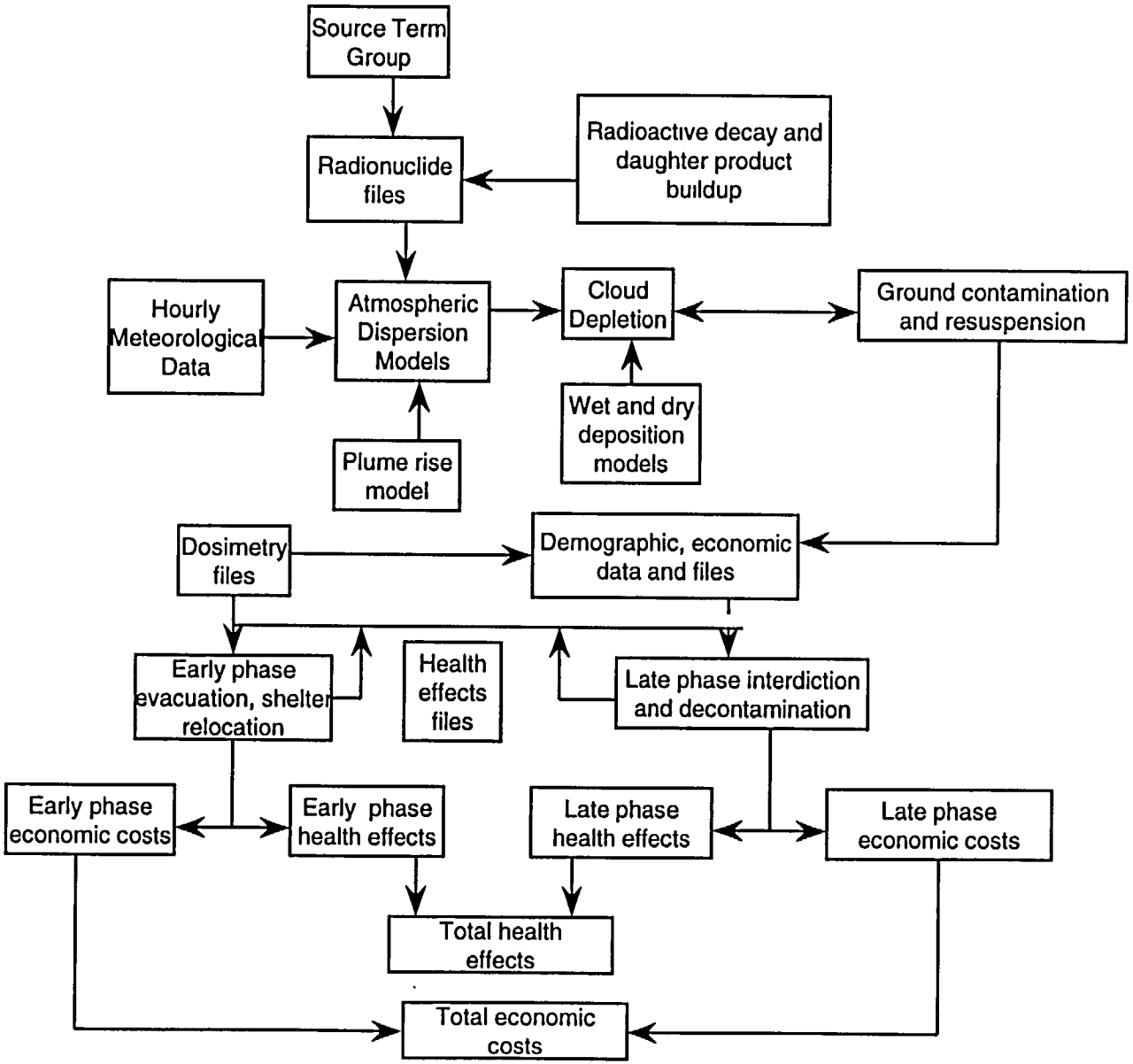
Consequences

- Population dose
- Acute effects
  - ✧ Number of fatalities, injuries, and illnesses occurring within one year due to initial exposure to radioactivity; nonlinear with dose equivalent
- Latent effects
  - ✧ Number of delayed effects and time of appearance as functions of dose for various organs; linear, no-threshold model typically used

Consequence Evaluation Models

- MACCS (MELCOR Accident Consequence Code System)
- Improved environmental transport, dosimetry, health effects, and economic cost models
- Improved wet deposition model for rainout
- Dependence of dry deposition velocity on particle size
- Multi-plume dispersion model including multi-step crosswind concentration profile
- Improved code architecture

Block Diagram of MACCS Models



Dominant Risk Contributors Sometimes Not Dominant With Respect to CDF

- For PWRs, SGTR and bypass sequences (e.g., ISLOCA) dominate LERF and therefore early fatalities
- SGTR and bypass not dominant contributors to core damage frequency
  - ✧ If SGTR or bypass occur, consequences are large
  - ✧ Remember: risk = frequency × consequence

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11. External Events

External Events

- Purpose: This topic will acquaint students with the definition of external events and the IPEEEs.
- Objectives:
  - ✧ Define external events and understand how they differ from internal events
  - ✧ List several of the more significant external events, including those analyzed in the IPEEEs
  - ✧ Know the objectives of the IPEEE and the acceptable approaches for seismic events and fires
  - ✧ Explain the ways in which external events may be evaluated and how this evaluation is related to the overall PRA task flow.
- Reference: NUREG/CR-2300, PRA procedures Guide; Generic Letter 88-20 Supplements 4 and 5, NUREG-1407

Overview of External Events Analysis

- External Events (EE) refers to those events that are external to system being analyzed
  - e.g., fires, floods, earthquakes
    - Includes on-site events such as flooding of various rooms within plant
- Concern is with dependent nature of EE
  - i.e., EE both initiates potential core damage accident AND results in failure of safety systems
- General approach
  - Identify hazard and its intensity
  - Conditional probability of plant SSCs failure
  - Assess overall plant response to event

NPP External Events Risk First Analyzed 1979

- 1979 - Oyster Creek (first seismic PRA)
- 1979 - HTGR (first fire PRA)
- 1981 - Big Rock Point
- 1982 - Zion/Indian Point
- 1983 - NUREG/CR-2300 (PRA Procedures Guide includes external events)
- 1988 - GL 88-20 (IPEs to include internal floods)
- 1989 - NUREG-1150 (fire and seismic)
- 1991 - GL-88-20, Supplement 4 (IPEEE, revised in 1995 with supplement 5, which revised seismic requirements)

Initial List of Potential External Event Hazards Very Extensive (1 of 2)

- Aircraft
- Avalanche
- *Earthquake
- *Fire in plant
- Fire outside plant but on site
- Fire off site
- Flammable fluid release
- Fog
- *Flooding, external (including seiche, storm surge, dam failure, and tsunami)
- Flooding, internal
- *High winds (including tornadoes)
- Hurricane
- Ice
- Industrial or military accident offsite
- Landslide
- Lightning
- Meteorite impact

Initial List of Potential External Event Hazards Very Extensive (2 of 2)

- Pipeline accident
- Sabotage
- Ship impact
- Toxic gas release
- Transportation accident
- Turbine missile
- Volcanic activity
- War
- Blizzard/Snow
- Drought
- Erosion
- Hail
- Heavy rain
- High temperature
- Low Temperature
- River diversion or change in lake level

Most Hazards Excluded for Various Reasons

- IPEEE required analysis of hazards believed to dominate external event risk
  - Seismic
  - Internal fires
  - High winds and tornadoes
  - External floods (internal flood analysis required in IPE)
  - Transportation and nearby facility accidents
  - Any known plant-unique hazards

External Events Analyses Performed at Various Levels of Detail

- Seismic
  - ✧ Seismic PRA or Seismic Margins Assessment (includes HCLPF - high confidence of low probability of failure assessment)
- Fire
  - ✧ Fire PRA or Fire Vulnerability Evaluation (FIVE)
- Other
  - ✧ EE PRA or screening analysis

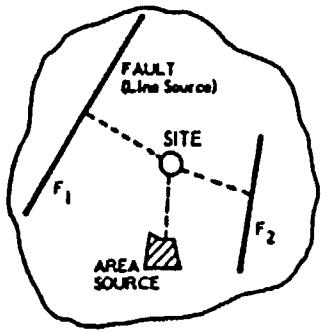
Seismic Hazard PRA - 3 Basic Steps

- Hazards analysis (frequency-magnitude relationship for earthquakes)
  - ✧ Location-specific hazard curves produced by NRC (LLNL) and EPRI
- Fragility analysis ("strength" of component)
  - ✧ Conditional probability of failure given a specific earthquake severity
- Accident sequence analysis

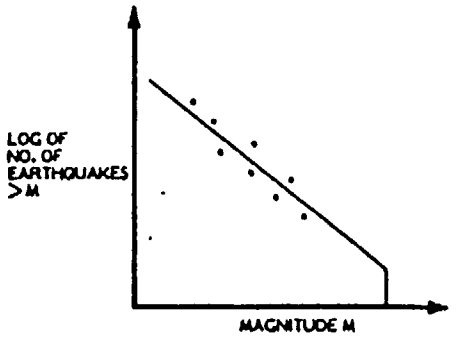
Analysis process briefly looked at in following slides

Four Steps in Seismic Hazard Curve Development

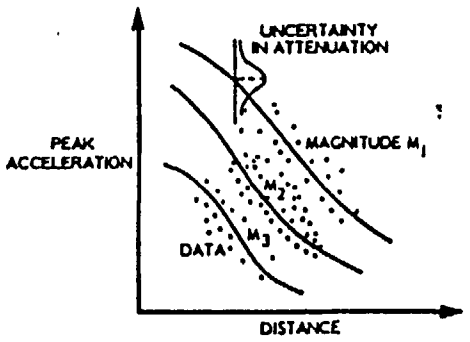
1. *Identify seismic sources*
2. *Develop frequency-magnitude model for each source*
3. *Develop ground motion model for each source*
4. *Integrate over sources*



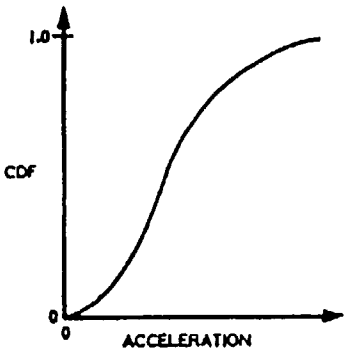
STEP 1
SOURCES



STEP 2
RECURRENCE



STEP 3
ATTENUATION



STEP 4
PROBABILITY OF NON- EXCEEDENCE WITHIN A TIME PERIOD t

Frequencies Estimated for Various Ground Acceleration Levels

- Frequency of 0.1g, 0.2g, 0.3g, etc. earthquake estimated
- Each g-level earthquake analyzed separately (i.e., as a separate and unique event)
- Failure probabilities of plant SSCs calculated based on g-level and fragility of SSC
- Internal events PRA re-evaluated using "new" seismic failure probabilities

Seismic Fragility Expressed in Terms of Peak Ground Acceleration

- Fragility (A) = A_m β_R β_U (lognormal model assumed)
  - A_m = median ground acceleration capacity of SSC
  - β_R β_U = Measure of the uncertainty in median fragility due to randomness and confidence, respectively (can also be labeled aleatory and epistemic, respectively).
  - A_m derived from various safety and response factors ($F_c F_{RE} F_{RS} A_{SSE}$), in turn are products of other factors
    - F_C - Capacity Factor
    - F_{RE} - Response factor for equipment
    - F_{RS} - Response factor for structure
    - A_{SSE} - Safe Shutdown Earthquake acceleration

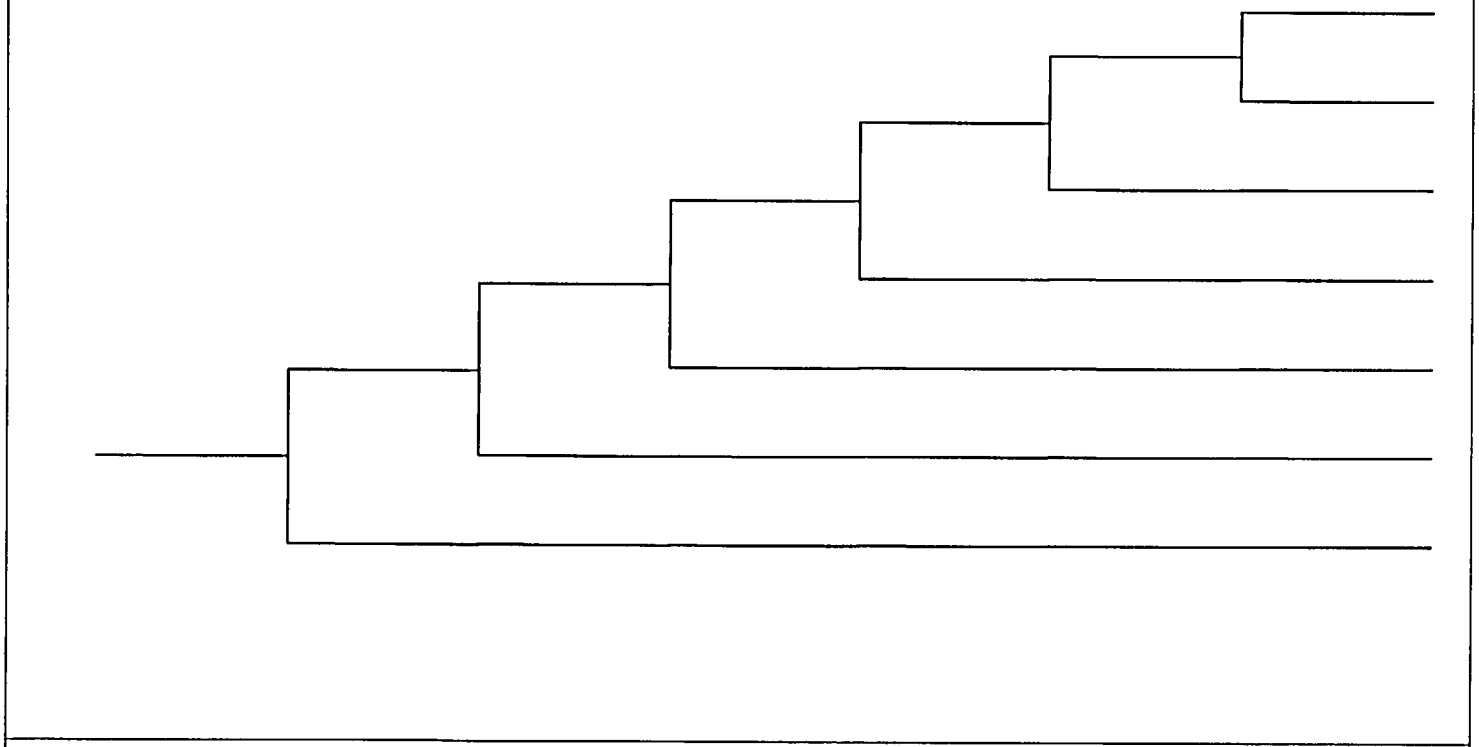
Range of Seismic Fragilities for Selected Components*

Componenent/Structure	*Dominant Failure Mode*	*Median Fragility Range (g)*
Concrete containment building	*Shear failure*	*2.50-9.20*
Reactor Pressure Vessel	*Anchor bolt*	*1.04-5.70*
Flat-bottom tank	*Shell wall buckling*	*0.20-1.00*
Batteries and racks	*Cases and plates*	*0.90-5.95*
Motor control centers	*Chattering*	*0.06-4.20*
Diesel generator	*Anchor bolt*	*0.70-3.89*
Offsite power	*Ceramic insulators*	*0.20-0.62*

* *Y. J. Park, etal, Survey of Seismic Fragilities Using in PRA Studies of Nuclear Power Plants, Reliability Engineering and System Safety, Vol. 62, pages 185-195, 1998.*

Probability of "Initiating Events" Estimated Given Occurrence of EE (Provides Link to Sequence Analysis)

Seismic Event Occurs	Reactor Vessel Rupture	Large LOCA	Medium LOCA	Small LOCA	Loss of Off-Site Power	Rx-Trip with FW nominally available	
EQ	RVR	LLOCA	MLOCA	SLOCA	LOSP	T	



SEISMIC - Seismic IE 2002/05/29 Page 5

Fire Analysis Follows Phased Approach

- Qualitative Screening
  - Fire in area does not cause a demand for reactor trip
  - Fire area does not contain safety-related equipment
  - Fire area does not have credible fire source or combustibles
- Quantitative Screening
  - Utilized existing internal events PRA
  - Estimate fire frequency for area and assume all equipment in fire area failed by fire, calculate CDF
- Detailed Analysis

Detailed Fire Analysis Includes

- Fire occurrence frequency assessment
  - ✧ Either location based or component based
  - ✧ Generic data updated with plant-specific experience
- Fire growth and propagation analysis
  - ✧ Considers: Combustible loading, fire barriers, and fire suppression
  - ✧ Modeled with specialized computer codes (COMPBRN IIIe)
- Component fragilities and failure mode evaluation
- Fire detection and suppression modeling
- Detailed fire scenarios analyzed using transient ET

Fire-Induced Vulnerability Evaluation (FIVE)

- Developed by EPRI as an alternative to a fire PRA for satisfying IPEEE requirements
- Equivalent to a fire-area screening analysis
  - ✧ worksheet-based systematic evaluation using information from Appendix R implementation
  - ✧ does not produce detailed quantification of fire CDF
- Most FIVE users (IPEEE) also quantified fire CDF of unscreened areas

Other External Events Analyzed Using Structured Screening Process

- IPEEE Guidance - Progressive Screening approach (see Figure 5.1 of NUREG-1407)
  - Review Plant Specific Hazard Data and Licensing Basis (FSAR)
  - Identify Significant Changes, if any, since OP Issuance
  - Does Plant/Facility Design Meet 1975 SRP Criteria (via quick screening & confirmatory walkdown)
    - If yes, no further analysis is needed
    - If no, continue analysis (next slide)

Examples of SRP Non-Conformance

- Flood
  - Probable Maximum Precipitation (PMP) at site based on old National Weather Service data
- High-Wind/Tornado
  - Design basis tornado missile spectrum different from that specified in SRP

If 1975 SRP Criteria Not Met

- Is Hazard Frequency Acceptably Low (<1E-5/yr)?

If Not:

- Does bounding analysis estimate CDF <1E-6/yr?

If Not:

- Perform detailed PRA
  - ✧ Details of analysis are tailored to particular hazard

12. SHUTDOWN RISK

Low-Power and Shutdown Risk

- Purpose: Discusses why low-power and shutdown modes of operation are thought to be of concern from a risk perspective, and introduces approaches to analyzing shutdown risk.
- References:
  - NUREG-1449 - Review of shutdown events
  - NUREG/CR-6143 and -6144 - Analysis of low-power shutdown risks at Grand Gulf and Surry
  - NUREG/CR-6616 - Risk comparison of scheduling preventive maintenance at shutdown vs at power operation for PWRs

Risk From LP/SD Operations Was Not Considered in Early PRAs

- Low-power and shutdown (LP/SD) encompasses operation when the reactor is subcritical or in transition between subcriticality and power operations up to ~15% of rated power
- In early risk studies, risk from full power operation was assumed to be dominant because during shutdown:
  - Reactor is subcritical
  - Decay heat is decreasing with time
    - Longer time is available to respond to accidents

LP/SD Operational Events Established the Credibility of LP/SD Risk

- Precursor events implied that potential generic vulnerabilities existed:
  - April 87 Diablo Canyon event resulting in loss of RHR while in mid-loop operation (and numerous similar events at other plants)
  - March 90 Vogtle plant loss of all AC power while shutdown
  - Two generic letters were subsequently issued relating to low-power and shutdown operations:
    - GL 87-12 -- Loss of RHR while the RCS is partially filled
    - GL 88-17 -- Loss of Decay Heat Removal

Operating Experience Insights Reinforced by Early LP/SD Risk Studies

- Limited risk studies of low-power and shutdown operations have suggested that shutdown risk may be significant because
  - Systems may not be available as Tech. Specs. allow more equipment to be inoperable than at power
  - Initiating events can impact operable trains of systems providing critical plant safety functions
  - Human errors are more prevalent because operators may find themselves in unfamiliar conditions not covered by training and procedures
  - Plant instruments and indications may not be available or accurate

Subsequent LP/SD Risk Studies Examined a Range of Issues

- Studies included:
  - ✧ Further review of operating experience for domestic and foreign reactors (discussed on next slide)
  - ✧ Analysis of selected significant events to estimate conditional probability of core damage using ASP models
  - ✧ Review of PRAs that included LP/SD operations
  - ✧ NRC sponsored Level 1 PRAs for LP/SD operations for Surry and Grand Gulf

Operating Experience Analysis

- AEOD* investigation of approximately 90 significant shutdown events out of 348 that occurred between January 1988, and July 1990 yielded the following major categories:
  - ✧ Loss of S/D cooling due to loss of system flow or loss of heat sink (27 events: 16 PWR and 11 BWR), e.g., errors during emergency power switching logic circuit testing caused a loss of AC power, resulting in loss of RHR for 15 minutes
  - ✧ Loss of reactor coolant inventory (22 events: 10 PWR and 12 BWR), e.g., opening RHR pump suction relief valve or PORV, or valve lineup errors
  - ✧ Loss of electrical power (19 events: 13 PWR and 6 BWR), e.g., loss of an AC, DC or instrument bus due to maintenance errors
  - ✧ Flooding and spills (3 PWR events)
  - ✧ Inadvertent reactivity addition (10 events: 4 PWR and 6 BWR), e.g., boron dilution without operator's knowledge
  - ✧ Breach of containment integrity (8 events, all human error)

** AEOD Special Report - Review of Operating Events Occurring During Hot and Cold Shutdown and Refueling, December 4, 1990*

NRC Continued Monitoring Operating LP/SD Experience

- AEOD performed follow-up investigation of shutdown events that occurred between January 1993 and May 1995, after licensees had time to implement NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management" (December 1991), and found:
  - ✧ Significant number of events during shutdown still occurring (486 during the 29-month investigation period), with 64 events having some measure of risk significance
  - ✧ Events similar to those of earlier investigation and still dominated by human errors during test and maintenance

NRC Staff's Evaluation of LP/SD Risk

- Vogtle (1990) SBO Investigation Motivated Broader Look at LP/SD Risk (NUREG-1449)
  - Study published in Sept 1993 documented significant technical findings including:
    - Outage planning is crucial to safety during S/D
    - Significant maintenance activities increase potential for fires during shutdown
    - PWRs are more likely to experience events than BWRs; dominant contributor to PWRs is loss of RHR during operations with reduced inventory (midloop operation)
    - Extended loss of RHR in PWRs can lead to LOCAs caused by failure of temporary pressure boundaries in RCS or rupture of RHR system piping

Subsequent LP/SD PRA Studies

- Although risks associated with shutdown and refueling conditions have not been studied as extensively as those for power operation, several limited PRAs have been completed for both PWRs and BWRs (e.g., Zion, Seabrook, Surry, Grand Gulf), as well as shutdown decay heat removal studies (Sequoyah, Brunswick); significant findings include:
  - ✧ Quantitative core damage frequency estimates for certain shutdown modes of operation are comparable to estimates for full power operation

Subsequent PRA Studies (Cont.)

- Most significant issues identified from a LP/SD risk perspective are:
  - Mid-loop operation (PWRs) of particular concern
  - Operator errors, especially
    - failure to determine proper actions to restore shutdown cooling
    - procedural deficiencies
  - Loss of RHR shutdown cooling, especially
    - operator induced
    - suction valve trips
    - cavitation due to overdraining of the RCS
  - Loss of offsite power

Few LP/SD PRA Have Been Developed

- Perception continues that LP/SD operations pose less risk than full-power
- LP/SD PRA developed reputation of being very expensive and complicated process
  - ✧ NUREG/CR-6143, -6144
- Most utilities have opted to manage LP/SD risk using simple configuration management approach
  - ✧ Vital safety functions defined - systems/trains needed to perform vital safety function maintained in-service

How Utilities are Addressing LP/SD Risk

- Some utilities have performed limited PRA studies of selected modes of operation
- Most utilities have adopted non-PRA approach
  - ✧ Approach based on guidance in NUMARC 91-06
  - ✧ Approach based on maintaining barriers during shutdown
  - ✧ EPRI sponsored development of software to implement this approach (ORAM*)

* *Outage Risk Assessment and Management*

SPAR Program Developing Limited Number of LP/SD Models

- Scheduled to produce 8 LP/SD models (Mar-02 to Mar-04)
- Models organized using 15 Plant Operating States (POSs) based on plant configuration evolutions and 4 Time Windows (time after reactor shutdown, i.e., different decay heat levels)
- Initiating Events include:
  - ✧ Loss of RHR
  - ✧ Loss of RHR given primary reactor coolant is at reduced inventory level
  - ✧ Loss of Offsite Power
  - ✧ Loss of primary reactor coolant Inventory

13. Uncertainties in PRA

Uncertainties in PRA

- **Purpose:** To acquaint students with how PRA treats uncertainty, including the identification of two types of uncertainty, aleatory and epistemic, and the characterization of one type of epistemic uncertainty with probability distributions.
- **Objectives:** Students will be able to identify the two types of uncertainty, along with their sources, and interpret probability distributions as an expression of epistemic uncertainty.
- **References:**
  - G. Apostolakis, "*The Concept of Probability in Safety Assessments of Technological Systems*," Science, 250, 1990.
  - NUREG-1489
  - G. Parry, "*The Characterization of Uncertainty in Probabilistic Risk Assessments of Complex Systems,*" Reliability Engineering and System Safety, **54** (1996), 119-126.
  - R. Winkler, "Uncertainty in Probabilistic Risk Assessment," Reliability Engineering and System Safety, **54** (1996), 127-132.
  - N. Siu and D. Kelly, "Bayesian Parameter Estimation in PRA," tutorial paper published in Reliability Engineering and System Safety **62** (1998).

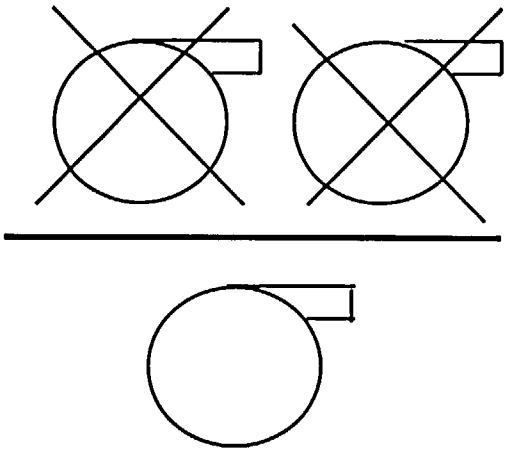
Uncertainty Arises From Many Sources

- Inability to specify initial and boundary conditions precisely
  - ✧ Cannot specify result with deterministic model
  - ✧ Instead, use probabilistic models (e.g., tossing a coin)
- Sparse data on initiating events, component failures, and human errors
- Lack of understanding of phenomena
- Modeling assumptions (e.g., success criteria)
- Modeling limitations (e.g., inability to model errors of commission)
- Incompleteness (e.g., failure to identify system failure mode)

Key Terminology:
Frequentist Interpretation of Probability

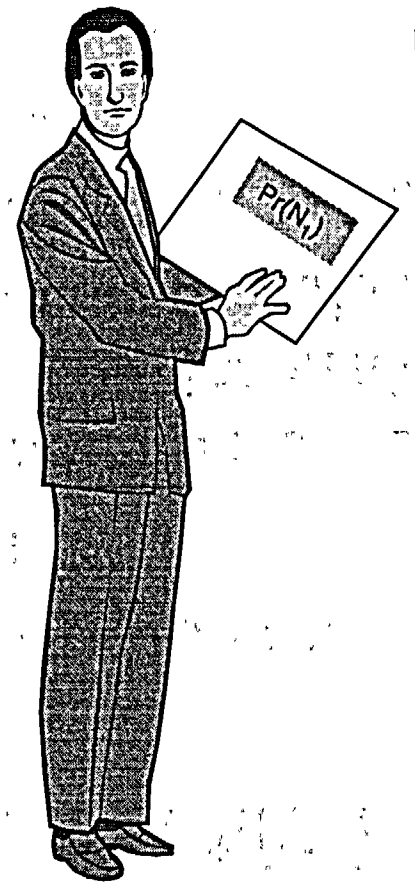
$$\Pr(N_1) = \lim_{N \to \infty} N_1 / N$$

$$\hat{p} = \frac{\text{(2)}}{\text{(100)}}$$



= 1/50
= 0.02
= 2E-2

Key Terminology: Subjectivist (Bayesian) Interpretation of Probability



- $Pr(N_1)$ is the degree of belief the analyst holds about the likelihood of event N_1 occurring

PRAs Identify Two Types of Uncertainty

- Distinction between aleatory and epistemic uncertainty:
  - ✧ "Aleatory" from the Latin Alea (dice), of or relating to random or stochastic phenomena. Also called "random uncertainty or variability."
  - ✧ "Epistemic" of, relating to, or involving knowledge; cognitive. [From Greek episteme, knowledge]. Also called "state-of-knowledge uncertainty."

Aleatory Uncertainty

- Variability in or lack of precise knowledge about underlying conditions makes events unpredictable. Such events are modeled as being probabilistic in nature. In PRAs, these include initiating events, component failures, and human errors.
- For example, PRAs model initiating events as a Poisson process, similar to the decay of radioactive atoms
- Poisson process characterized by frequency of initiating event, usually denoted by parameter l

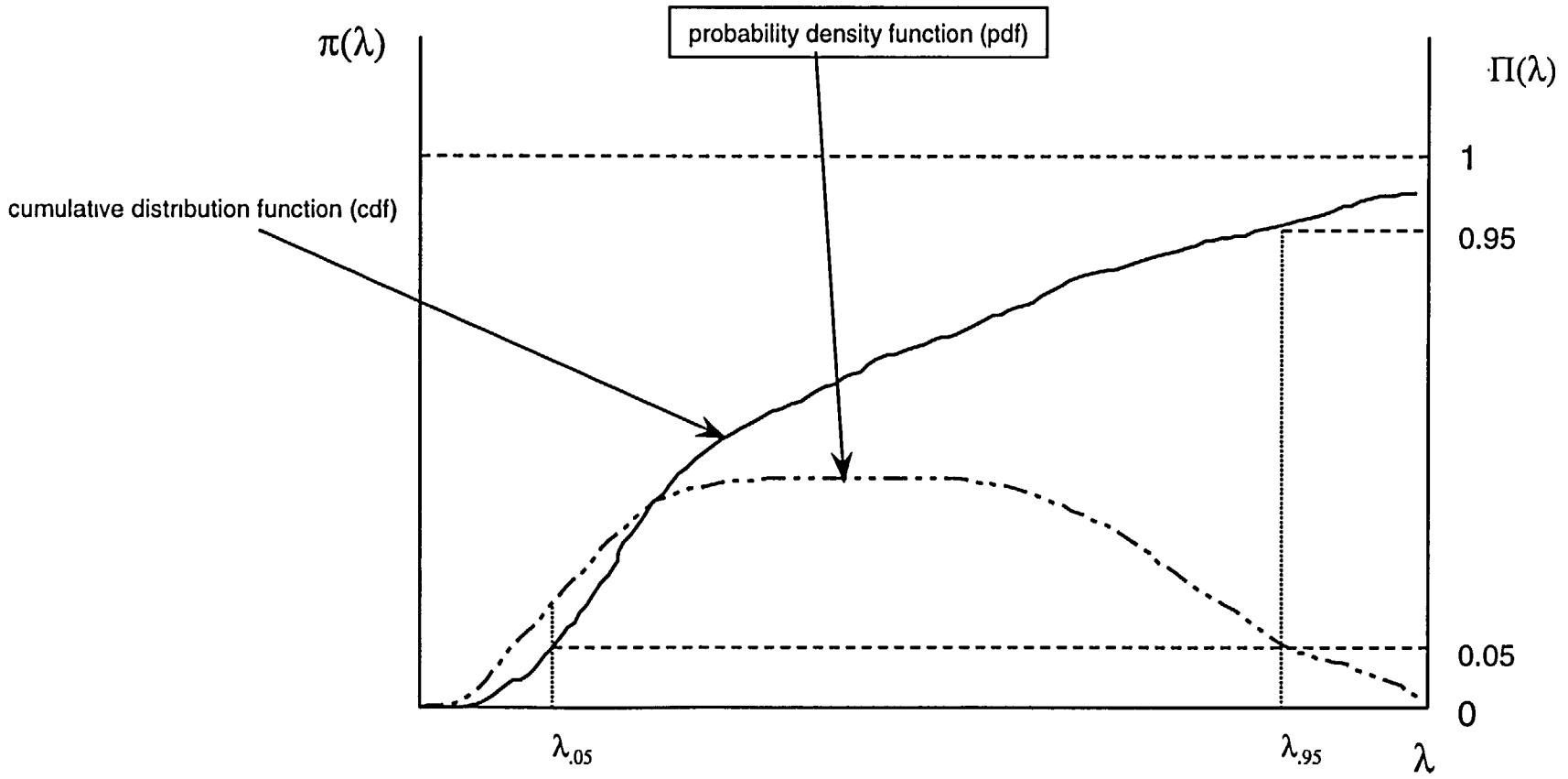
Epistemic Uncertainty

- Value of λ is not known precisely
- Could model uncertainty in estimate of λ using statistical confidence interval
  - ✧ Can't propagate confidence intervals through PRA models
  - ✧ Can't interpret confidence intervals as probability statements about value of λ
- PRAs model lack of knowledge about value of λ by assigning (usually subjectively) a probability distribution to λ
  - ✧ Probability distribution for λ can be generated using Bayesian methods.

Epistemic Uncertainty (cont'.)

- Advantages to Bayesian Approach
  - ✧ Allows uncertainties to be propagated easily through PRA models
  - ✧ Allows probability statements to be made concerning I and outputs that depend upon I
  - ✧ Provides unified, consistent framework for parameter estimation

Uncertainty in λ Expressed as Probability Distribution



Uncertainty Propagation

- Uncertainties propagated via Monte Carlo sampling
- In this approach, output probability distribution is generated empirically by repeated sampling from input parameter distributions

Other Epistemic Uncertainties in PRA

- Modeling uncertainty
  - ✧ System success criteria
  - ✧ Accident progression phenomenology
  - ✧ Health effects models (linear versus nonlinear, threshold versus nonthreshold dose-response model)

Other Epistemic Uncertainties in PRA (cont.)

- Completeness
  - Complex errors of commission
  - Design and construction errors
  - Unexpected failure modes and system interactions
  - All modes of operation not modeled
- Errors in analysis
  - Failure to model all trains of a system
  - Data input errors
  - Analysis errors

Addressing Other Epistemic Uncertainties

- Modeling uncertainty usually addressed through sensitivity studies
  - ✧ Research ongoing to examine more formal approaches
- Completeness addressed through comparison with other studies and peer review
  - ✧ Some issues (e.g., design errors) are simply acknowledged as limitations
  - ✧ Other issues (e.g., errors of commission) are topics of ongoing research
- Analysis errors may be difficult to catch; addressed through peer review and validation process

Uncertainty in PRA

For additional information:

Probability & Statistics for PRA (P-102) course covers modeling and propagation of uncertainty in great detail. It covers both the frequentist and Bayesian approaches and compares and contrasts the two.

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14. Configuration Risk Management

Configuration Risk Management

- Purpose: To acquaint students with the basic concepts of using PRA models to control configuration risk by planning maintenance.
- Objectives: Students will be able to explain;
  - ✧ Why base case PRA results cannot be used for maintenance planning
  - ✧ What is meant by "configuration risk management"
  - ✧ How configuration risk management is related to risk-informed regulation
- Reference: NUREG/CR-6141, Handbook of Methods for Risk-Based Analyses of Technical Specifications

Configuration Risk Management
Why an Issue?

- Economics - Plants are moving towards increased maintenance while at power, to reduce outage durations
- Safety
  - ✧ Increased maintenance while at power not covered in IPEs/PRAs
  - ✧ Increased on-line maintenance can produce high-risk plant configurations

Configuration Risk Management
Why an Issue?

"In general, the industry appears to be adopting the practice of on-line maintenance faster than it is developing and implementing effective controls to manage the safety (risk) implications of this practice."

[Temporary Instruction (TI) 2525/126, "Evaluation of On-line Maintenance, February 1995," page 5]

Observed Preventive Maintenance Practices of Concern

- Multiple components simultaneously out of service, as allowed (implicitly) by technical specifications
- Repeated entries into Action Statements to perform PM + long equipment downtimes
- Significant portions of power operations may be spent in Action Statements to carry out PMs

Configuration Risk Management Traditional Approaches

- Technical Specifications and Limiting Conditions for Operation
  - Identify systems/components important to safety based on traditional engineering approach
  - Limit component out-of-service times for individual and combinations of component outages (not based on formal risk analysis)
- Maintenance planning guidelines such as 12-week rolling schedule, etc.
  - Provide guidance to work week planners on allowable maintenance/testing
  - Based on train protection concept and Technical Specifications
- Operator judgment

Configuration Risk Management Traditional Approaches

- Weaknesses of Traditional Approaches
  - ✧ Generally based on and limited to Technical Specification equipment
  - ✧ No limit on frequencies of equipment outages - only on duration of each outage
- Is the traditional approach good enough, given the increased emphasis on on-line maintenance?
- How can PRA help?

Configuration Risk Management

- Configuration risk management: one element of risk-informed regulation
- Can be forward-looking or retrospective
  - ✧ Forward-looking to plan maintenance activities & outage schedules
  - ✧ Retrospective to evaluate risk significance of past plant configurations

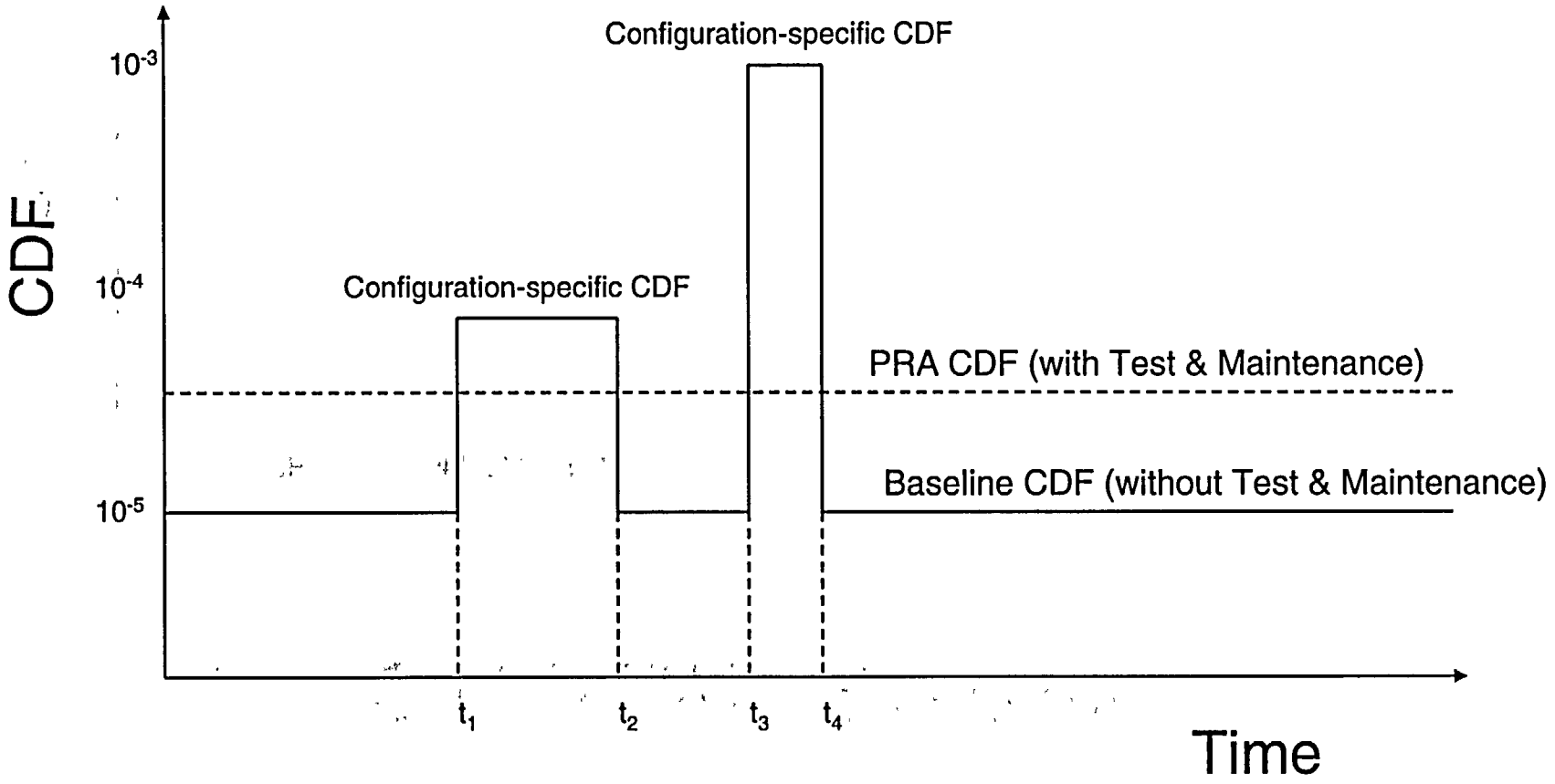
Configuration Risk Management

- Plant configuration: state of the plant as defined by status of plant components
- Involves taking measures to avoid risk-significant configurations, limit duration and frequency of such configurations that cannot be avoided

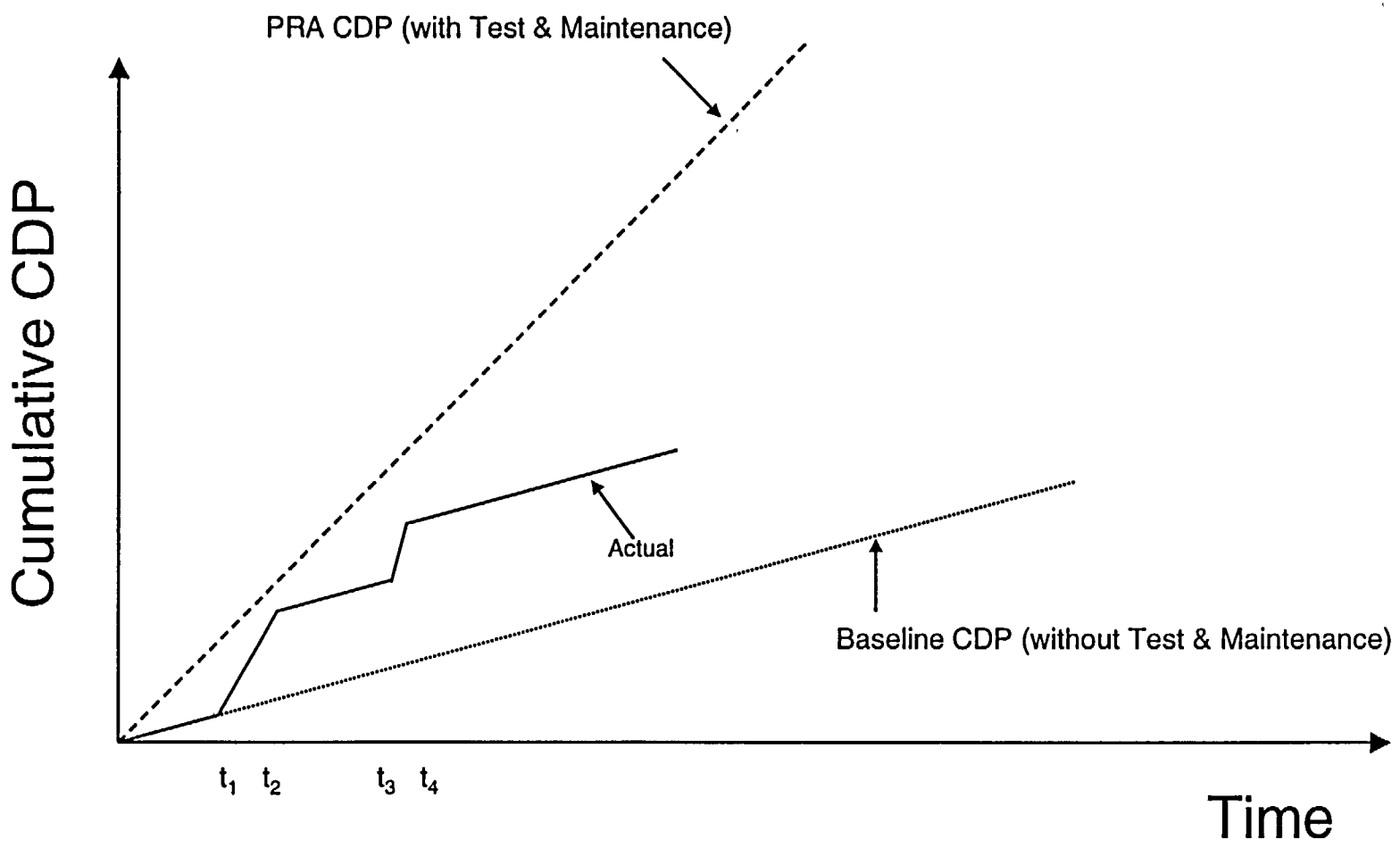
Configuration Risk Management

- Configuration risk has various measures
  - Core damage frequency (instantaneous)
    - Baseline CDF (the zero maintenance CDF)
    - Configuration-specific CDF
  - Incremental CDF
    - = Configuration-specific CDF - Baseline CDF
  - Core damage probability (CDP)
    - = CDF * duration
  - Incremental core damage probability (ICDP)
    - = ICDF * duration
    - = CCDP - CDP
  - Incremental large early release probability (ICLERP)
    - = ILERF * duration
    - = CLERP - LERP

CDF Profile


Configuration-specific CDF
10^{-3}
10^{-4}
10^{-5}
CDF
Configuration-specific CDF
PRA CDF (with Test & Maintenance)
Baseline CDF (without Test & Maintenance)
t_1
t_2
t_3
t_4
Time

Cumulative CDP Profile



Configuration Risk Management

- Requires management of:
  - OOS components
    - instantaneous CDF (configuration-specific CDF)
  - Outage time of components & systems
    - configuration duration
    - CCDP
    - ICDP
  - Backup components
    - instantaneous CDF
  - Configuration frequency
    - cumulative CDP over time

Managing OOS Components

- Involves scheduling maintenance and tests to avoid having critical combinations of components or systems out of service concurrently
- For Maintenance Rule; 10 CFR 50.65
  - A value of 1E-3/year is suggested in NUMARC 93-01 for a ceiling for configuration-specific CDF
    - Subject of such a ceiling value being studied by the NRC
    - NRC neither endorses nor disapproves 1E-3/year value

Managing Outage Time

- Must determine how long configuration can exist before risk incurred becomes significant
  - Many utilities using EPRI PSA Application Guide numerical criteria, although not endorsed by NRC
  - NRC has no numerical criteria at present for temporary changes to plant
  - For Maintenance Rule,
    - Configuration Should not normally be entered voluntarily
      - >1E-5 ICDP
      - >1E-6 ILERP
    - Assess non quantifiable factors and establish risk management actions
      - 1E-6 to 1E-5 ICDP
      - 1E-7 to 1E-6 ILERP
    - Normal work controls
      - <1E-6 ICDP
      - <1E-7 ILERP
  - For risk-informed Tech. Specs., for single AOT:
    - ICCDP < 5E-7
    - ICLERP < 5E-8
- Must know compensatory measures to take to extend outage time without increasing risk

Managing Backup Components

- Must determine which components can carry out functions of those out of service

Controlling Frequency

- Must track frequency of configurations and modify procedures & testing to control occurrences, as necessary and feasible

Why Configuration Risk Management is Needed...

- PRA/IPE assumes random failures of equipment (including equipment outages for testing & maintenance)
- PRA/IPE baseline model does not correctly model simultaneous outages of critical components
- Simultaneous outages (i.e., plant configurations) can increase risk significantly above the PRA/IPE baseline
- Lack of configuration management can affect initiating events and equipment designed to mitigate initiating events, leading to increased risk

Preventive Maintenance Risk Calculations

- Risk impact of PM on single component
- Risk impact of maintenance schedule
- Risk impact of scheduling maintenance (power operations versus shutdown)

Risk Monitors

- On-line risk monitors can be used to evaluate plant configurations for a variety of purposes:
  - To provide current plant risk profile to plant operators
  - As a forward-looking scheduling tool to allow decisions about test and maintenance actions weeks or months in advance of planned outages
  - As a backward-looking tool to evaluate the risk of past plant configurations

Current Risk Monitor Software Packages

- Erin Engineering Sentinel
- Scientech/NUS Safety Monitor
  - The NRC acquired this package from Scientech, and has an agency-wide license covering its use
- EPRI R&R Workstation
- Commonwealth Edison OSPRE

Requisite Features

- Risk monitor software requires (at a minimum) the following features:
  - PRA solution engine for analysis of the plant logic model
  - Database to manage the various potential plant configurations
  - Plotting program to display results

Risk Monitor Capabilities

- As a tool for plant operators to evaluate risk based on real-time plant configuration:
  - Calculates measure of risk for current or planned configurations
  - Displays maximum time that can be spent in that particular configuration without exceeding pre-defined risk threshold
  - Provides status of plant systems affected by various test and maintenance activities
  - Operators can do quick sensitivity studies to evaluate the risk impacts of proposed plant modifications

Risk Monitor Capabilities (cont.)

- As a tool for plant scheduling for maintenance and outage planning:
  - ✧ Generates time-line that shows graphically the status of plant systems and safety functions
  - ✧ Generates risk profile as plant configuration varies over time
  - ✧ Identifies which components have strongest influence on risk

Risk Monitor Strengths and Weaknesses

- Risk Monitor Strengths
  - Provides risk determinations of current and proposed plant configurations
  - Compact model
  - Many current PRA models can be converted into risk monitor format
  - Can obtain importance and uncertainty information on results
  - Provides risk management guidance by indicating what components should be restored first

Risk Monitor Strengths and Weaknesses (cont.)

- Risk Monitor Limitations
  - For some PRA codes, difficulty of converting PRA models into master logic diagram (e.g., Large Event Tree approach models)
  - Effort required to set up databases to link master logic diagram events to plant components and electronic P&IDs, and interface with scheduling software
  - Analysis Approximations
    - CCF adjustments
    - Human recovery modeling
    - Consideration of plant features not normally modeled in PRA studies
    - Cut set updating versus logic model solution
    - Truncation limits

Additional Sources of Information

- Further details on configuration risk management can be found in NUREG/CR-6141, Handbook of Methods for Risk-Based Analyses of Technical Specifications.
- Risk Assessment for Event Evaluation (P-302) course in the PRA Technology Transfer Program curriculum explores the use of PRA techniques for evaluating the risk significance of operational events, as well as plant configuration risk management, discusses the other risk measures mentioned in this module (e.g., CCDP and event importance), and illustrates use of the GEM code to perform the necessary PRA calculations.

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15. Introduction to Risk-Informed Decision-Making

Introduction to Risk-Informed Decision-Making

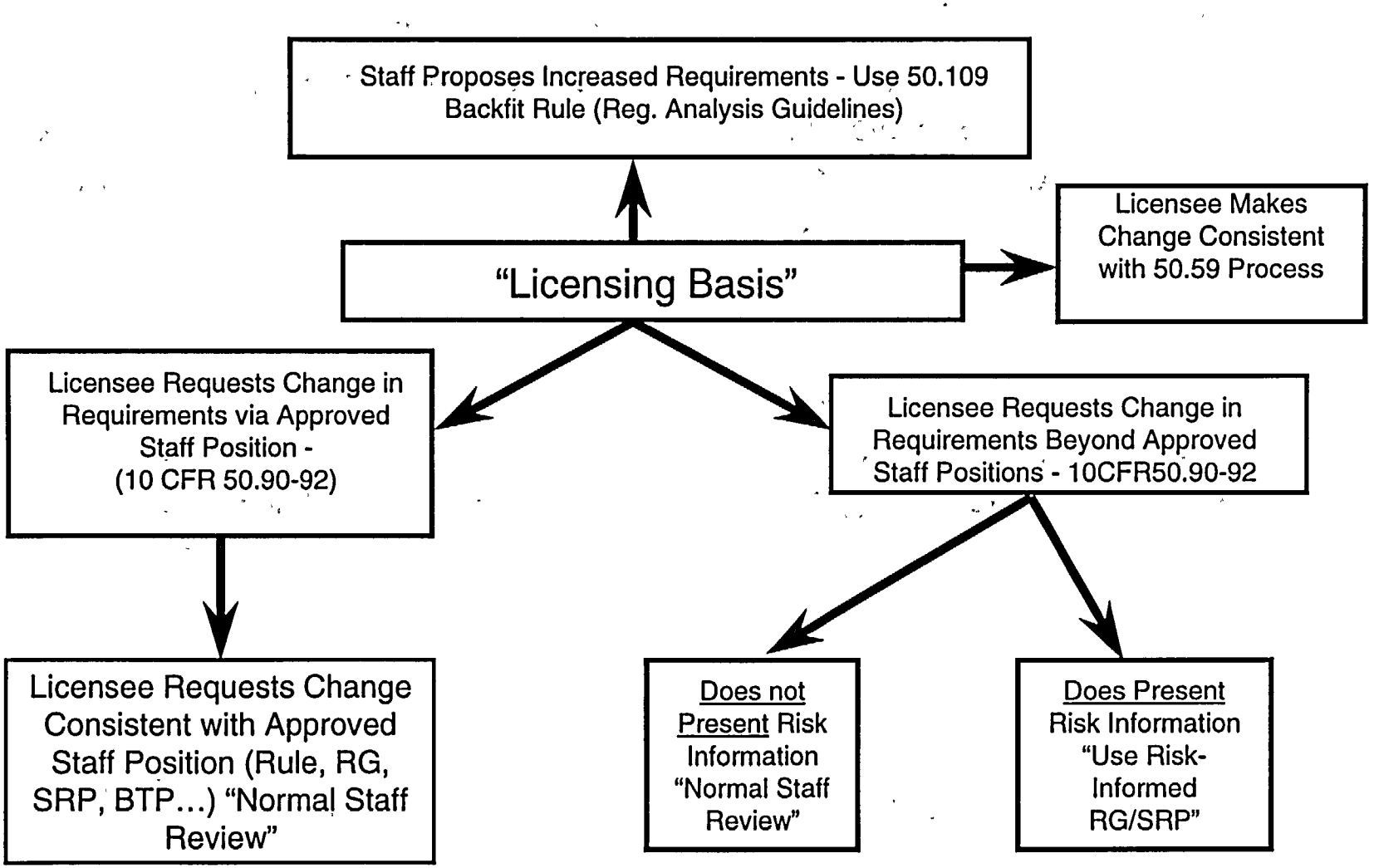
- Purpose: Discuss the principal steps in making risk-informed regulatory decisions, including the acceptance guidance contained in the draft SRPs addressing this subject.

Risk-Informed Regulatory Guides and SRPs

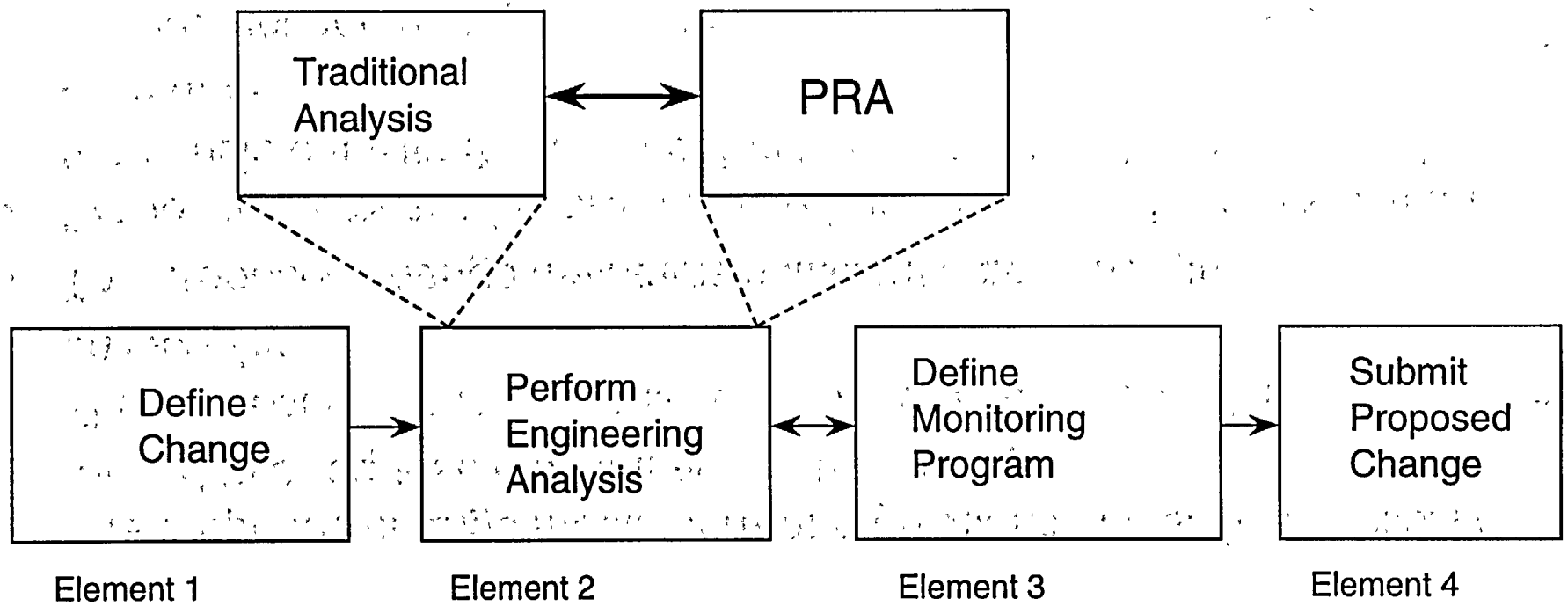
- R. G. 1.174 - General guidance to licensees
- R.G.-1.175 - Application-specific guidance on in-service testing
- R.G. – 1.176 - Application-specific guidance on graded quality assurance
- R.G. – 1.177 - Application-specific guidance on technical specifications
- R.G. – 1.178 - Application-specific guidance on in-service inspection

- SRP Chapter 19 - General guidance to staff
- SRP Section 3.9.7 - Application-specific guidance on IST
- Inspection guidance - under development
- SRP Section 16.1 - Application-specific guidance on technical specifications
- SRP Section 3.9.8 - Application-specific guidance on ISI

Decision Logic for Submittal Reviews



Principal Steps in Risk-Informed Plant-Specific Decision-Making



Principles of Risk-Informed Regulation

- The proposed change meets current regulations unless it is explicitly related to a requested exemption or rule change
- The proposed change is consistent with the defense-in-depth philosophy
- The proposed change maintains sufficient safety margins
- Proposed increases in core damage frequency and risk are small and are consistent with the intent of the Commission's Safety Goal Policy Statement
- The impact of the proposed change should be monitored using performance measurement strategies

Expectations from Risk-Informed Regulation

- All safety impacts of the proposed change are evaluated in an integrated manner as part of an overall risk management approach in which the licensee is using risk analysis to improve operational and engineering decisions broadly by identifying and taking advantage of opportunities for reducing risk, and not just to eliminate requirements the licensee sees as undesirable. For those cases where risk increases are proposed, the benefits should be described and should clearly outweigh the proposed risk increases. The approach used to identify changes in requirements should be used to identify areas where requirements should be increased, as well as where they could be reduced.

Expectations from Risk-Informed Regulation (cont.)

- Acceptability of proposed changes should be evaluated by the licensee in an integrated fashion that ensures that all principles are met
- The use of core damage frequency (CDF) and large early release frequency (LERF) as bases for probabilistic risk assessment acceptance guidelines is an acceptable approach. Use of the Commission's Safety Goal Quantitative Health Objectives (QHOs) for this purpose is acceptable in principle and licensees may propose their use; however, in practice, implementing such an approach would require careful attention to the methods and assumptions used in the analysis, and treatment of uncertainties.

Expectations from Risk-Informed Regulation (cont.)

- Increases in estimated CDF and LERF resulting from proposed changes will be limited to small increments and the cumulative effect of such changes should be tracked
- The scope and quality of the engineering analyses (including traditional and probabilistic analyses) conducted to justify the proposed change should be appropriate for the nature and scope of the change and should be based on the as-built and as-operated and maintained plant, including reflection of operating experience at the plant
- Appropriate consideration of uncertainty is given in analyses and interpretation of findings
- A program of monitoring, feedback, and corrective action should be used to address significant uncertainties

Expectations from Risk-Informed Regulation (cont.)

- The plant-specific PRA supporting licensee proposals has been subjected to quality controls such as an independent peer review or certification
- Data, methods, and assessment criteria used to support regulatory decision-making must be scrutable and available for public review

Acceptance Guidelines

- Defense-in-depth is maintained
  - A reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation is preserved
  - Over-reliance on programmatic activities to compensate for weaknesses in plant design is avoided
  - System redundancy, independence, and diversity are preserved commensurate with the expected frequency and consequences of challenges to the system (e.g., no risk outliers)
  - Defenses against potential common-cause failures are preserved and the potential for introduction of new common-cause failure mechanisms is assessed

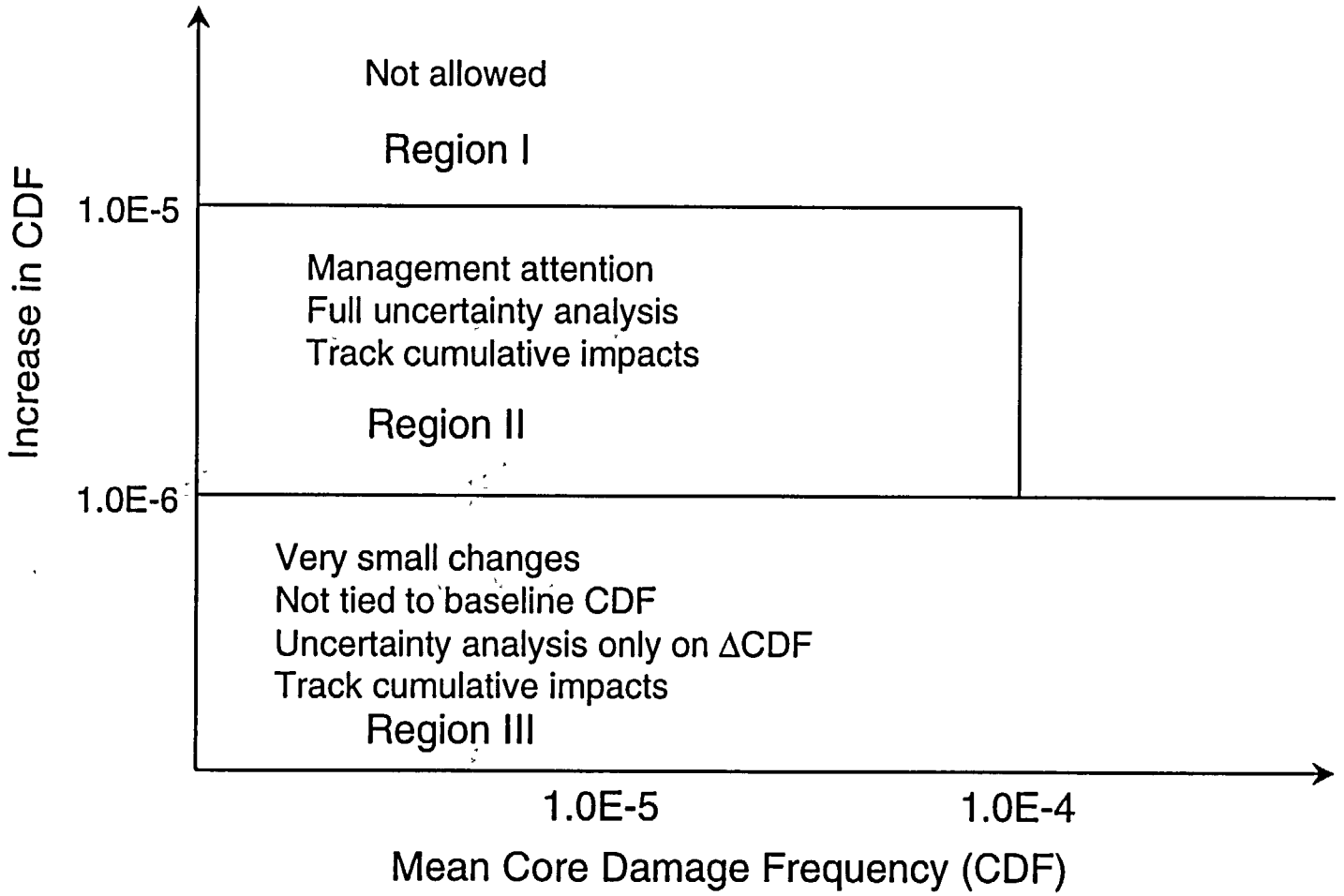
Acceptance Guidelines (cont.)

- Defense-in-depth is maintained (cont.)
  - ✧ Independence of barriers is not degraded
  - ✧ Defenses against human errors are preserved
  - ✧ The intent of the General Design Criteria in 10 CFR 50, App. A, are maintained
- Sufficient safety margins are maintained
  - ✧ Codes and standards or alternatives approved for use by the NRC are met
  - ✧ Safety analysis acceptance criteria in the licensing basis (e.g., FSAR, supporting analyses) are met, or proposed revisions provide sufficient margin to account for analysis and data uncertainty

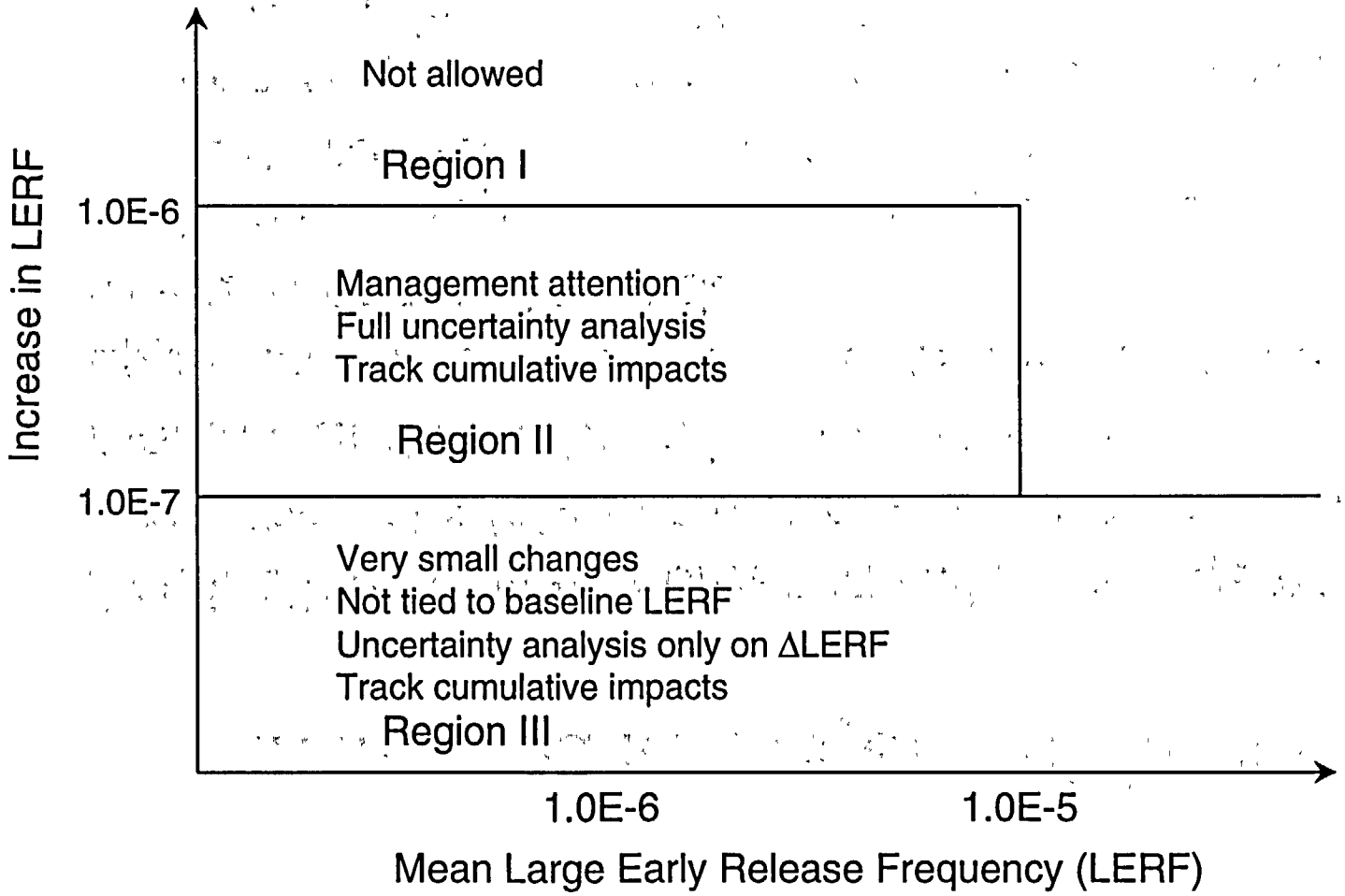
Acceptance Guidelines (cont.)

- Risk guidelines on following slides are met
  - Risk guidelines are intended for comparison with full-scope PRA results
    - Internal events (full power, low power, shutdown)
    - External events (seismic, fire, etc.)
    - Use of less than full scope PRA may be acceptable

Mean Core Damage Frequency Acceptance Guidelines



Mean Large Early Release Frequency Acceptance Guidelines



Increased Management Attention

- Application is given increased NRC management attention when the calculated values of the changes in the risk metrics, and their baseline values when appropriate, approach the guidelines. The issues addressed by management will include
  - ✧ Cumulative impact of previous changes and trend in CDF and LERF (licensee's risk management approach)
  - ✧ Impact of proposed change on operations complexity, burden on operating staff, and overall safety practices
  - ✧ Benefit of the change with respect to its risk increase
  - ✧ Level 3 PRA information, if available

Consideration of Uncertainties

- Use mean values for comparison with guidelines
- Identify important sources of uncertainty
  - ✧ Parameter
  - ✧ Modeling
  - ✧ Completeness
- Perform sensitivity calculations on parameter and modeling uncertainties
- Perform quantitative or qualitative analysis on completeness uncertainties
- Results of sensitivity studies should generally meet guidelines
- Region III - no need to calculate uncertainty on baseline CDF/LERF

Combined Change Requests

- Several changes can be combined in one submittal
- Will be reviewed against acceptance guidelines
  - ✧ Individually with respect to defense in depth
  - ✧ Cumulatively
- Combined changes should be related. For example
  - ✧ Be associated with same system, function, or activity
  - ✧ Changes reviewed individually against risk criteria if not closely related
- Combined changes should not trade many small risk decreases for a large risk increase (i.e., create a new significant contributor to risk)

Key Issues in PRA Quality

- Ensure that, within scope, PRA analysis is complete and has appropriate level of detail
  - Consideration of relevant initiating events, plant systems, and operator actions
  - Analysis reflects plant-specific operating experience, design features, and accident response
  - All calculations are documented
- PRA methodology and associated input
  - Influence of models, input data, and assumptions on results and conclusions
- Licensee review and QA process
  - Peer review
  - Certification
  - Standards

NRC Staff and Management Responsibilities

- Ensure that licensing submittals are identified and processed in accordance with risk-informed guidance
- Identify current requirements that could be significantly enhanced with a risk-informed and/or performance-based approach
- Ensure objectives of risk-informed regulation are met
  - ✧ Enhanced safety decisions
  - ✧ Efficient use of NRC resources
  - ✧ Reduced unnecessary industry burden
- Ensure adequate staff training on use of risk-informed guidance and underlying PRA technical disciplines
- Maintain current levels of safety

16. Acronyms and Abbreviations

Acronyms and Abbreviations (1 of 4)

AC	Alternating current
ACRS	Advisory Committee on Reactor Safeguards
ADS	Automatic depressurization system
ADV	Atmospheric dump valve
AEOD	Office for Analysis and Evaluation of Operational Data
AFW	Auxiliary feedwater
AOP	Abnormal Operating Procedure
AOT	Allowed outage time
AOV	Air-operated valve
APB	Accident progression bin
APET	Accident progression event tree
ASEP	Accident Sequence Evaluation Program
ASP	Accident Sequence Precursor
ATHEANA	A Technique for Human Event Analysis
ATWS	Anticipated transient without scram
BC	Boundary condition
BNL	Brookhaven National Laboratory
BTP	Branch Technical Position
BWR	Boiling water reactor
BWROG	BWR Owners' Group
BWST	Borated water storage tank
CCDF	Complementary cumulative distribution flinction
CCDP	Conditional core damage probability
CCF	Common-cause failure
CCI	Core-concrete interaction
CCW	Component Cooling Water
CDF	Core damage frequency
CDFM	Conservative Deterministic Failure Margin
CDP	Core damage probability
CE	Combustion Engineering
CEOG	Combustion Engineering Owners' Group
CFR	Code of Federal Regulations
CLB	Current licensing basis
CRD	Control rod drive
CSIP	Charging/safety injection pump
CST	Condensate storage tank
CW	Circulating water
DBA	Design basis accident
DC	Direct current
DCH	Direct containment heating
DF	Decontamination factor
DFSD	Dominant functional sequence diagram
DHR	Decay heat removal
ECCS	Emergency core-cooling system
EDG	Emergency diesel generator
EOOS	Equipment Out of Service System
EOP	Emergency Operating Procedure
EPA	Environmental Protection Agency
EPIX	Equipment performance and information exchange system
EPRI	Electric Power Research Institute

Acronyms and Abbreviations (2 of 4)

ESF	Engineered safeguards feature
ESW	Emergency service water
ESWGR	Emergency switchgear
ET	Event tree
FCI	Fuel-coolant interaction
FIVE	Fire-Induced Vulnerability Evaluation
FMEA	Failure modes and effects analysis
FSAR	Final Safety Analysis Report
FT	Fault tree
F-V	Fussell-Veseley (importance)
FW	Feedwater
GE	General Electric
GL	Generic Letter
HCLPF	High confidence, low probability of failure
HCR	Human Cognitive Reliability
HEP	Human error probability
HHSI	High-head safety injection
HLW	High-level waste
HPCI	High-pressure coolant injection
HPCS	High-pressure core spray
HPI	High-pressure injection
HPR	High-Pressure re-circulation
HPSI	High-pressure safety injection
HRA	Human reliability analysis
HVAC	Heating, ventilation, and air conditioning
HTGR	High-Temperature Gas Reactor
HX	Heat exchanger
ICCDP	Incremental conditional core dame probability
ICLERP	Incremental conditional large early release probability
IE	Initiating event
INEEL	Idaho National Engineering and Environmental Laboratory
INPO	Institute for Nuclear Plant Operations
IPE	Individual Plant Examination
IPEEE	Individual Plant Examination for External Events
IREP	Interim Reliability Evaluation Program
ISA	Integrated Safety Analysis
ISI	In-service inspection
ISLOCA	Interfacing system loss-of-coolant accident
IST	In-service testing
JCO	Justification for Continued Operation
LB	Licensing basis
LCO	Limiting Condition for Operation
LER	Licensee Event Report
LERF	Large early release frequency
LERP	Large early release probability
LLNL	Lawrence Livermore National Laboratory
LLW	Low-level waste
LOCA	Loss-of-coolant accident
LOOP	Loss of offsite power
LOSP	Loss of offsite power

Acronyms and Abbreviations (3 of 4)

LP&S	Low power and shutdown
LPCI	Low-pressure coolant injection
LPCS	Low-pressure core spray
LPI	Low-pressure injection
LPR	Low-pressure re-circulation
LPSI	Low-pressure safety injection
LPZ	Low population zone
LWR	Light water reactor
MAAP	Modular Accident Analysis Program
MACCS	MELCOR Accident Consequence Code System
MCS	Minimal cut set
MDP	Motor-driven pump
MGL	Multiple Greek letter
MOV	Motor-operated valve
MSIV	Main steam isolation valve
MSP	Maintenance and Surveillance Program
NCV	Non-cited violation
NEI	Nuclear Energy Institute
NMSS	Office of Nuclear Materials Safety and Safeguards
NOED	Notice of Enforcement Discretion
NPRDS	Nuclear Plant Reliability Data System
NRC	Nuclear Regulatory Commission
NRR	Office Nuclear Reactor Regulation
NUMARC	Nuclear Management and Resources Council
OOS	Out of service
ORAM	Outage Risk Assessment and Management
ORNL	Oak Ridge National Laboratory
OSHA	Occupational Safety and Health Administration
P&ID	Piping and instrumentation diagram
PA	Performance assessment
PCC	PRA Coordinating Committee
PCS	Power conversion system
PDS	Plant damage state
PM	Preventive maintenance
PORV	Power-operated relief valve
POS	Plant operating state
PRA	Probabilistic risk assessment
PRT	Plant response tree
PRV	Pressurizer power-operated relief valves
PSA	Probabilistic safety assessment
PSF	Performance shaping factor
PTFG	PRA Training Focus Group
PTS	Pressurized thermal shock
PWR	Pressurized water reactor
QA	Quality Assurance
QHO	Quantitative health objective
QRA	Quantitative risk analysis
RAW	Risk achievement worth
RBCCW	Reactor building closed cooling water
RCIC	Reactor core isolation cooling

Acronyms and Abbreviations (4 of 4)

RCP	Reactor coolant pump
RCS	Reactor coolant system
RES	Office of Nuclear Regulatory Research
RG	Regulatory Guide
RHR	Residual heat removal
RI	Resident Inspector
RPS	Reactor protection system
RRW	Risk reduction worth
RSS	Reactor Safety Study
RVC	Relief valve re-close
RWST	Refueling water storage tank
S/D	Shutdown
SAR	Safety Analysis Report
SBO	Station blackout
SDC	Shutdown cooling
SER	Safety Evaluation Report (Staff Evaluation Report for IPE/IPEEE)
SG	Steam generator
SGTR	Steam generator tube rupture
SHARP	Systematic Human Action Reliability Procedure
SI	Safety injection
SIF	Seal injection flow
SIT	Safety injection tank
SLOCA	Small loss-of-coolant accident
SNL	Sandia National Laboratory
SRA	Senior Reactor Analyst
SRI	Senior Resident Inspector
SRP	Standard Review Plan
SRV	Safety/relief valve
SSC	Systems, structures, and components
SSET	Support state event tree
STG	Source term group
SW	Service water
SWGR	Switch gear
TBCCW	Turbine building closed cooling water
TDP	Turbine-driven pump
TER	Technical Evaluation Report
THERP	Technique for Human Error Rate Prediction
TRC	Time reliability correlation
VCT	Volume control tank
WOG	Westinghouse Owners' Group