Risk-Informed Changes to the Licensing Basis - I

22.39 Elements of Reactor Design, Operations, and Safety

Lecture 12

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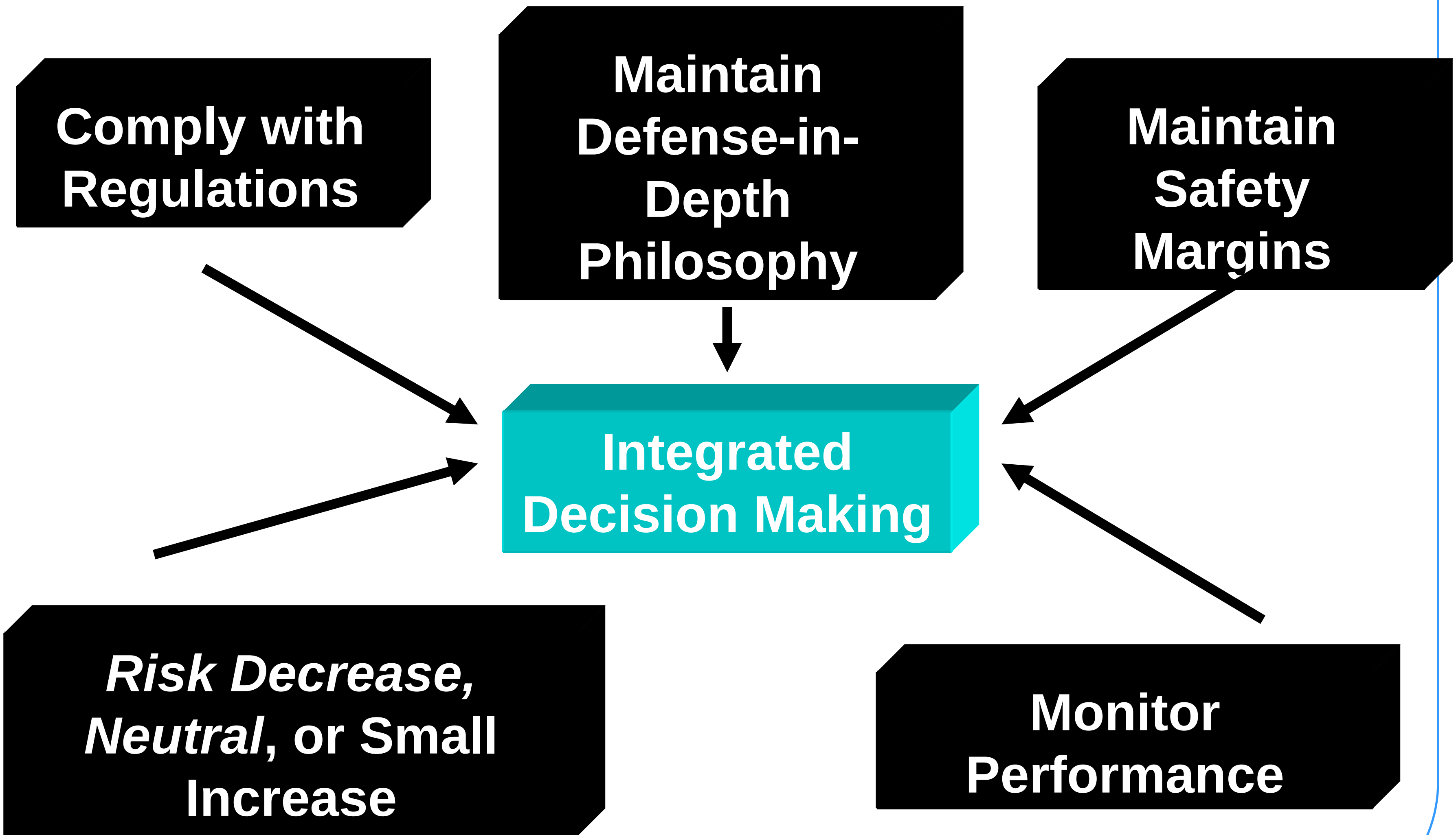
Licensing Basis Changes

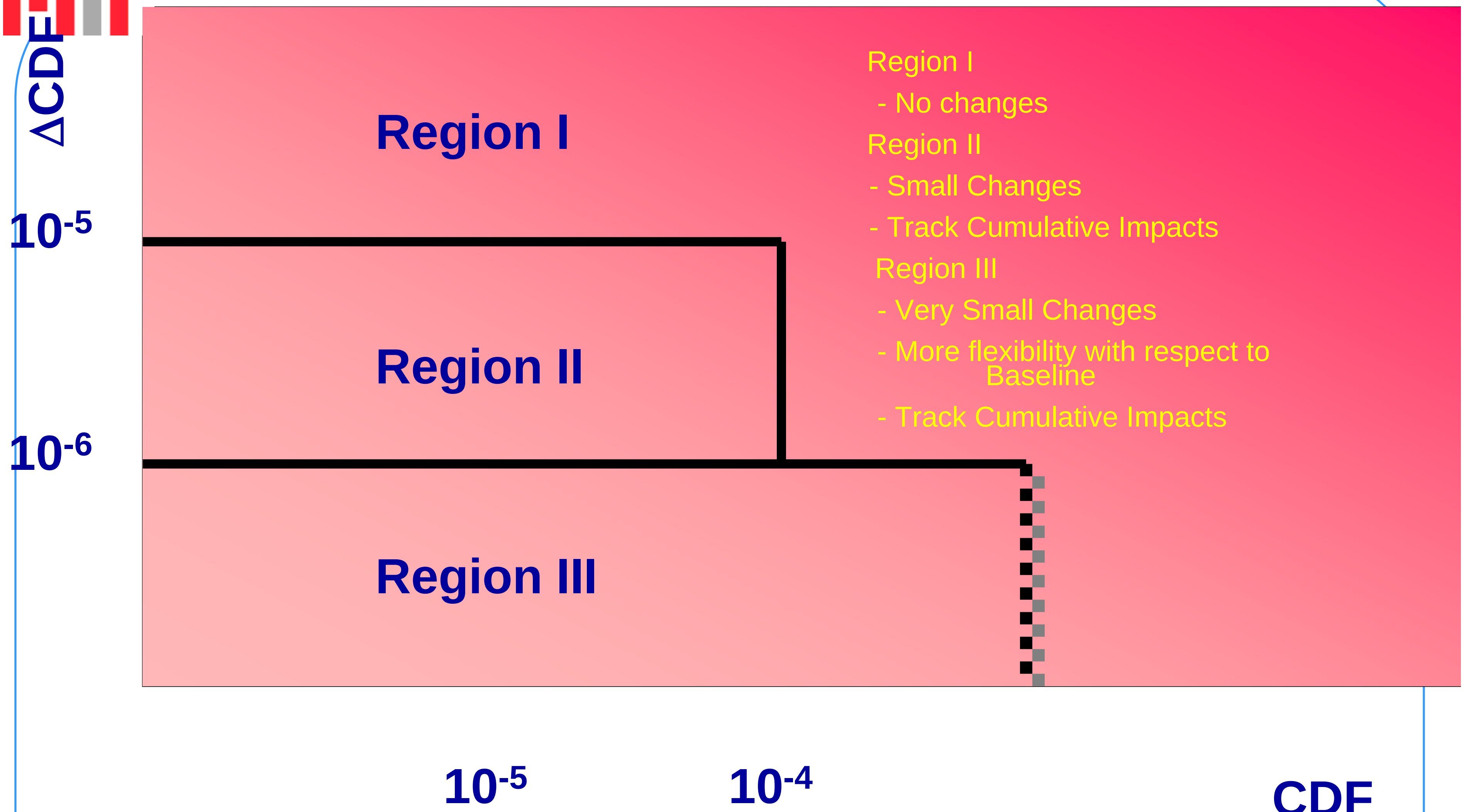
- These are modifications to a plant's design, operation, and other activities that require NRC approval.
- Regulatory Guide 1.174 (General Guidance) was issued in 1998 and revised in 2002.
- In-Service Testing (RG 1.175)
- Graded Quality Assurance (RG 1.176)
- Technical Specifications (RG 1.177)
- In-Service Inspection (RG 1.178)

The Integrated Decision-Making Process (RG 1.174)



Defense In Depth (RG 1.174)

- A reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation.
- 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, consequences of challenges to the system, and uncertainties (e.g., no risk outliers).
- Defenses against common-cause failures are preserved, and the potential for the introduction of new common-cause failure mechanisms is assessed.
- Independence of barriers is not degraded.
- Defenses against human errors are preserved.
- The intent of the GDC in Appendix A to 10 CFR Part 50 is maintained.



Acceptance Guidelines for Core Damage Frequency

Uncertainties

- **Aleatory uncertainty is built into the structure of the PRA model itself.**
- **Epistemic uncertainties:**
  - Parameter uncertainties are those associated with the values of the fundamental parameters of the PRA model, such as equipment failure rates, initiating event frequencies, and human error probabilities that are used in the quantification of the accident sequence frequencies.
  - In many cases, understanding of certain processes or phenomena is incomplete, and there may be different opinions on how the models should be formulated. Examples: modeling human performance, common cause failures, and reactor coolant pump seal behavior upon loss of seal cooling. This gives rise to model uncertainty.
  - Completeness is not in itself an uncertainty, but a reflection of scope limitations. The problem with completeness uncertainty is that, because it reflects an unanalyzed contribution, it is difficult (if not impossible) to estimate its magnitude. Examples: the analysis of some external events and the low power and shutdown modes of operation, and influences of organizational performance.

Comparison with Acceptance Guidelines

- The acceptance guidelines were established with the Commission's Safety Goals and subsidiary objectives in mind, and these goals were intended to be compared with mean values. Therefore, the mean values of the distributions should be used.
- For the distributions generated in typical PRAs, the mean values typically corresponded to the region of the 70th to 80th percentiles, and coupled with a sensitivity analysis focused on the most important contributors to uncertainty, can be used for effective decision-making.
- Approach: Address parametric uncertainty and any explicit model uncertainties in the assessment of mean values; perform sensitivity studies to evaluate the impact of changes in key assumptions or the use of alternate models for the principal implicit model uncertainties; and use quantitative analyses or qualitative analyses as necessary to address incompleteness as appropriate to the decision and the acceptance guidelines.

Important Note

"The analysis will be subject to increased technical review and management attention as indicated by the darkness of the shading of the figure. In the context of the integrated decision-making, the boundaries between regions should not be interpreted as being definitive; the numerical values associated with defining the regions in the figure are to be interpreted as indicative values only."

Regulatory Guide 1.174

Increased Management Attention

Consider:

- The cumulative impact of previous changes and the trend in CDF (the licensee's risk management approach);
- The cumulative impact of previous changes and the trend in LERF (the licensee's risk management approach);
- The impact of the proposed change on operational complexity, burden on the operating staff, and overall safety practices;
- Plant-specific performance and other factors, including, for example, siting factors, inspection findings, performance indicators, and operational events; and Level 3 PRA information, if available;
- The benefit of the change in relation to its CDF/LERF increase;
- The practicality of accomplishing the change with a smaller CDF/LERF impact; and
- The practicality of reducing CDF/LERF, in circumstances where there is reason to believe that the baseline CDF/LERF are above the guideline values (i.e., 10-4 and 10-5 per reactor year).

South Texas Project Experience with Allowed Outage Times

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

Example: 1-out-of-2 System

$$Q = \frac{1}{3}\lambda^2 T^2 + \lambda\tau + \frac{1}{2}\lambda_{CCF} T + \gamma_0 \gamma_1$$

λ	standby failure rate
T	Surveillance Test Interval
τ	Allowed Outage Time
λ_{CCF}	common-cause failure rate
γ_0	unconditional human error rate
γ_1	conditional human error rate

ΔCDF and Δ LERF can be calculated from the PRA.

Phased Approach to PRA Quality

- **In the 12/18/03 Staff Requirements Memorandum, the Commission approved the implementation of a phased approach to PRA quality.**
- **The phases are differentiated by the availability of standards.**
- **Phase 3 should be achieved by December 31, 2008. Guidance documents will be available to support all anticipated applications.**

- ***Standard for PRA for Nuclear Power Plant Applications*, ASME RA-S-2002.**
- ***"An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,"* RG 1.200, February 2004**

ACRS Interpretations of DiD

- ***<u>Structuralist</u>:*** **DiD is embodied in the structure of regulations and in the design of the facilities built to comply with those regulations. "What if this barrier or safety feature fails?"**

- ***<u>Rationalist</u>:*** **DiD is the aggregate of provisions made to compensate for uncertainty in our knowledge of accident initiation and progression.**

Sorensen, J.N., Apostolakis, G. E., Kress, T.S., and Powers, D.A., "On the Role of Defense in Depth in Risk-Informed Regulation," *Proceedings of PSA '99, International Topical Meeting on Probabilistic Safety Assessment*, pp. 408-413, Washington, DC, August 22 - 26, 1999, American Nuclear Society, La Grange Park, Illinois.

The Concerns

- Arbitrary appeals to the structuralist interpretation of defense-in-depth might diminish the benefits of risk-informed regulation.
- Strict implementation of risk-based regulation (the rationalist interpretation of defense-in-depth) without appropriate consideration of the structuralist defense-in-depth could undermine the historical benefits.

We continue to be surprised

- **Recent events have shaken our confidence in our assumptions.**
  - **"The NRC and DBNPS failed to adequately review, assess, and followup on relevant operating experience."**
  - **"DBNPS failed to assure that plant safety issues would receive appropriate attention."**
  - **"The NRC failed to integrate known or available information into its assessments of DBNPS's safety performance."**

[Davis Besse NPS Lessons-Learned Report, USNRC, September 30, 2002]

The ACRS Pragmatic Approach

- **Apply defense-in-depth (the structuralist approach) at a high level, e.g., the ROP cornerstones (e.g., IEs, Safety Functions).**
- **Implement the rationalist approach at lower levels, except when PSA models are incomplete. Revert to the structuralist approach in these cases.**

Risk-Informed Framework

Traditional "Deterministic" Approaches

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

Risk-Informed Approach

•Combination of traditional and risk-based approaches

Risk-Based Approach

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

Benefits (NRC)

- **Risk-informing regulatory activities have enhanced and extended the traditional, deterministic, by:**
  - **Allowing consideration of a broader set of potential challenges to safety,**
  - **Providing a logical means for prioritizing these challenges based on risk significance, and**
  - **Allowing consideration of a broader set of resources to defend against these challenges**

Remarks (NRC)

- **Risk-informed initiatives have enhanced every aspect of reactor regulations**
- **Steady progress is being made to continue the implementation of the Commission PRA policy and direction**
- **Enhanced public safety and a reduction of regulatory burden is resulting in redirection of resources to areas of greater benefit**

G. Holahan, RIODM Lecture, MIT, 2006

Special Treatment Requirements

- Requirements imposed on structures, systems, and components (SSCs) that go beyond industry-established requirements for commercial SSCs.
- *Safety-related* SSCs are subject to special treatment, including quality assurance, testing, inspection, condition monitoring, assessment, evaluation and resolution of deviations.
- *Non-safety-related* SSCs are not.
- The categorization of SSCs as *safety-related* and *non-safety-related* does not have a rational basis.
- These requirements are very expensive.
- The impact of special treatment on SSC performance is not known.

Traditional SSC Categorization

Safety-Related	Non-Safety Related

SSC Categorization (10 CFR 50.69

RISC - 1 **Safety-Related,** **Safety Significant** **FV>0.005 <u>and</u> RAW>2** ***Maintain Current Requirements*** **STP: 3,971 (6.0%)**	**RISC - 2** **Non-Safety Related,** **Safety Significant** **FV>0.005 <u>or</u> RAW>2** ***Impose Current Requirements*** **STP: 456 (0.7%)**
RISC - 3 **Safety-Related,** **Low Safety Significant** **FV<0.005 <u>and</u> RAW<2** ***Maintain Design Basis Requirements*** **STP: 13,755 (20.8%)**	**RISC - 4** **Non-Safety Related,** **Low Safety Significant** **FV<0.005 <u>and</u> RAW<2** **No Special Treatment** **STP: 47,876 (72.5%)**

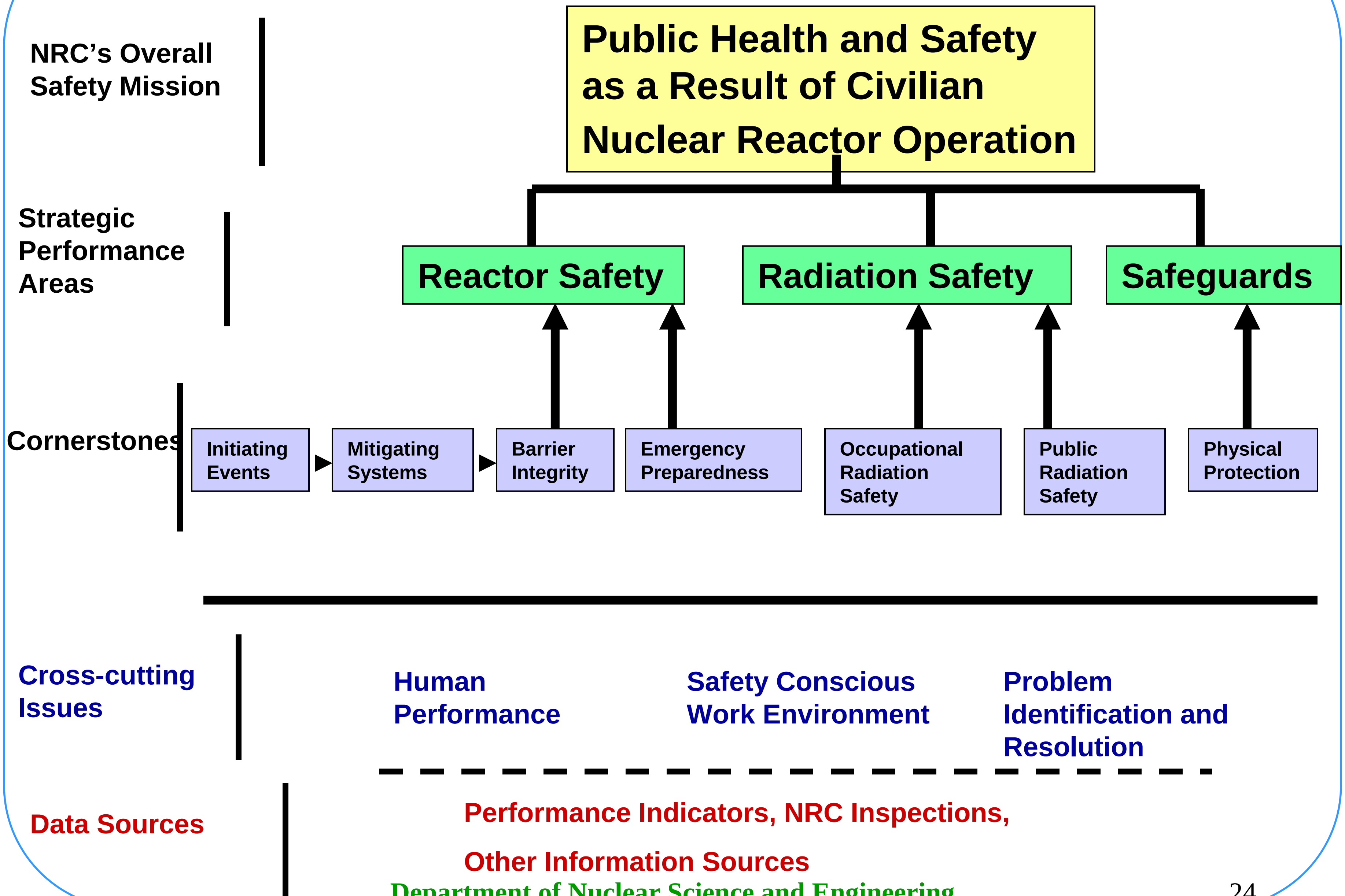
Importance Measures ↑

Traditional →

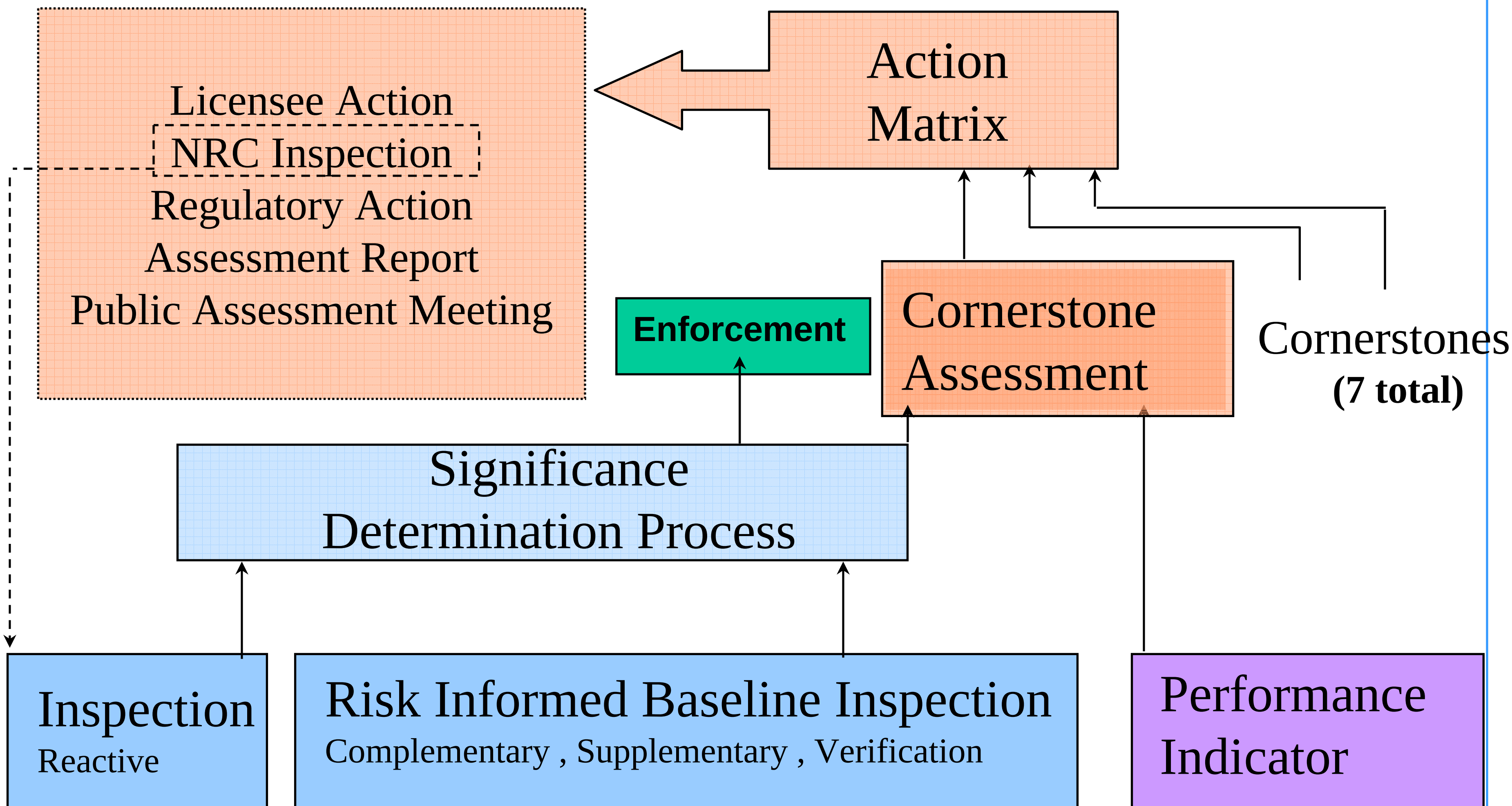
Reactor Oversight Process: Objectives

- Make the oversight process more objective, predictable, consistent, and risk-informed.
- Reduce unnecessary regulatory burden.
- Integrate inspection, assessment, and enforcement processes.
- Utilize objective indicators of performance.
- Utilize inspections focused on key safety areas.
- Apply greater regulatory attention to facilities with performance problems while maintaining a base level of regulatory attention on plants that perform well.
- Respond to violations in a predictable and consistent manner that reflects the safety significance of the violations.

Regulatory Framework



Plant Assessment Process



Levels of Significance Associated with Performance Indicators and Inspection Findings

- Green - very low risk significance (for PIs: Within peer performance)
- White - low to moderate risk significance
- Yellow - substantive risk significance
- Red - high risk significance

$\Delta CDF < 1E\text{-}6$
$1E\text{-}6 < \Delta CDF < 1E\text{-}5$
$1E\text{-}5 < \Delta CDF < 1E\text{-}4$
$\Delta CDF > 1E\text{-}4$

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

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

Performance Indicators (1)

- Initiating Events
  - Unplanned Scrams
  - Scrams with Loss of Normal Heat Removal
  - Unplanned Power Changes
- Mitigating Systems
  - Safety System Unavailability
  - Safety System Functional Failures
- Barriers
  - Fuel Cladding (Reactor Coolant System)
  - Reactor Coolant System (Leak Rate)

Performance Indicators (2)

- Emergency Preparedness
  - Drill/Exercise Performance
  - Emergency Response Organization Drill Participation
  - Alert and Notification System Reliability
- Occupational Radiation Safety
  - Occupational Exposure Control Effectiveness
- Public Radiation Safety
  - Radiological Effluent Occurrences
- Physical Protection
  - Protected Areas Security Equipment Performance Index
  - Personnel Screening Program Performance
  - Fitness-for-Duty/Personnel Reliability Program Performance

Performance Indicators (3)

- Emergency Preparedness
  - Drill/Exercise Performance
  - ERO Drill Participation
  - Alert and Notification System Reliability
- Occupational Radiation Safety
  - Occupational Exposure Control Effectiveness
- Public Radiation Safety
  - RETS/ODCM Radiological Effluent Occurrence
- Physical Protection
  - Protected Areas Security Equipment Performance Index
  - Personnel Screening Program Performance
  - Fitness-for-Duty/Personnel Reliability Program Performance

Examples of Thresholds for PIs

	G/W	W/Y	Y/R
Reactor Safety			
Unplanned Scrams	3	6	25
AFW Unavailability	0.02	0.06	0.12
Public Radiation Safety			
Radiological Effluent Occurrences	7 or more events in 3 yrs (rolling average); 4 or more in 1 yr	14 or more events in 3 yrs (rolling average); 8 or more in 1 yr	N/A

Objectives of the Significance Determination Process

- Characterize the significance of inspection findings using risk insights
- Provide a framework for communicating potential safety significant findings
- Provide a basis for assessment and/or enforcement actions

Significance Determination Process