Risk Applications in USNRC Inspection & Licensing Activities

Mike Franovich
Senior Reliability & Risk Analyst
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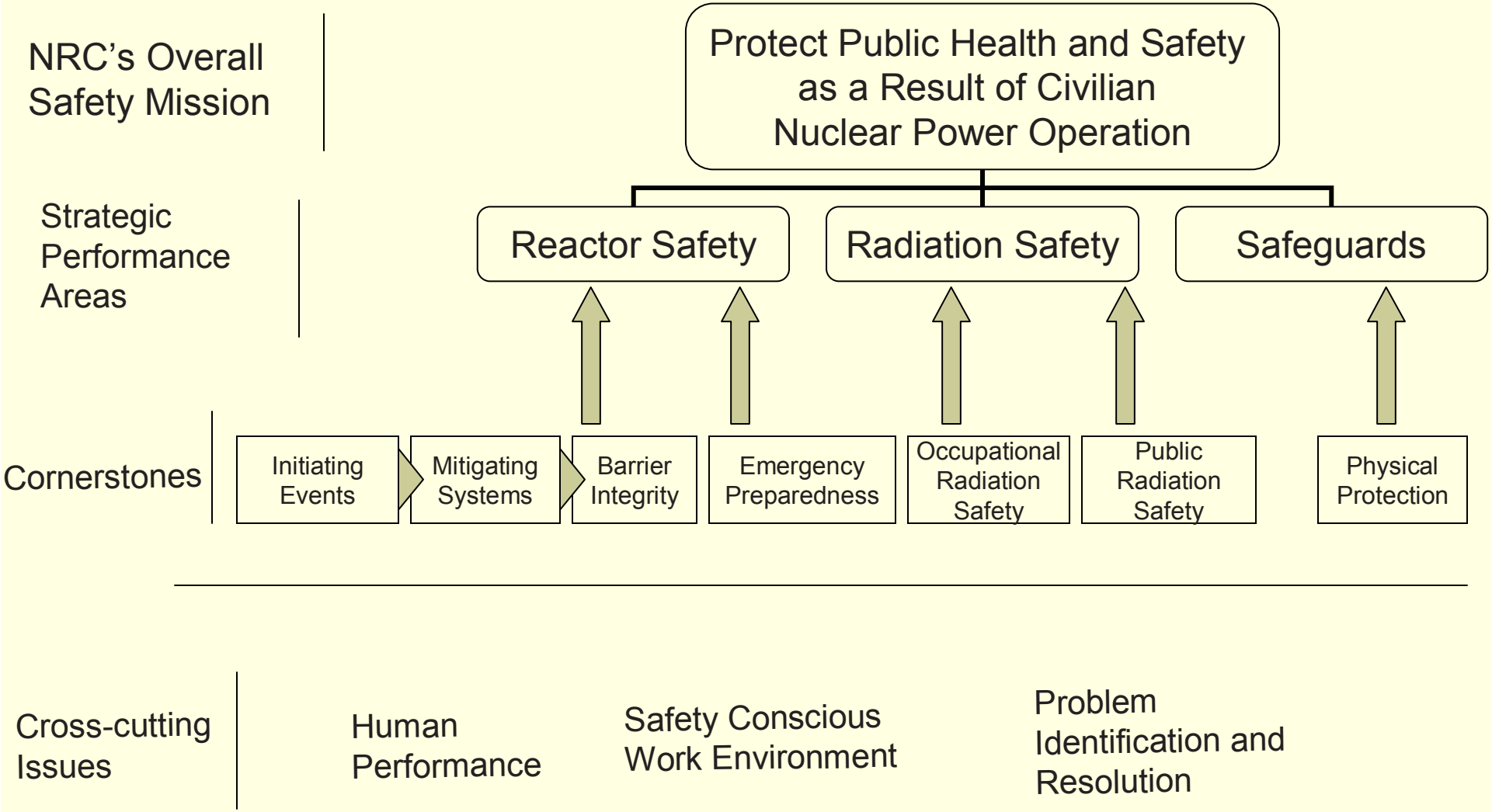
Presentation Outline

- Illustration of risk applications in selected areas of inspection and licensing
  - Reactor Oversight Process (ROP)
  - Significance Determination Process (SDP)
  - Incident Response
  - Maintenance Rule
  - Risk Informed Technical Specifications

Features of the Reactor Oversight Process

- Focuses Inspections on Activities Where Potential Risks Are Greater.
- Applies Greater Regulatory Attention to Facilities with Performance Problems While Maintaining a Base Level of Regulatory Attention on Plants That Perform Well.
- Makes Greater Use of Objective Measures of Plant Performance.
- Gives the Industry and Public Timely and Understandable Assessments of Plant Performance.
- Avoids Unnecessary Regulatory Burden.
- Responds to Violations in a Predictable and Consistent Manner That Reflects the Safety Impact of the Violations.

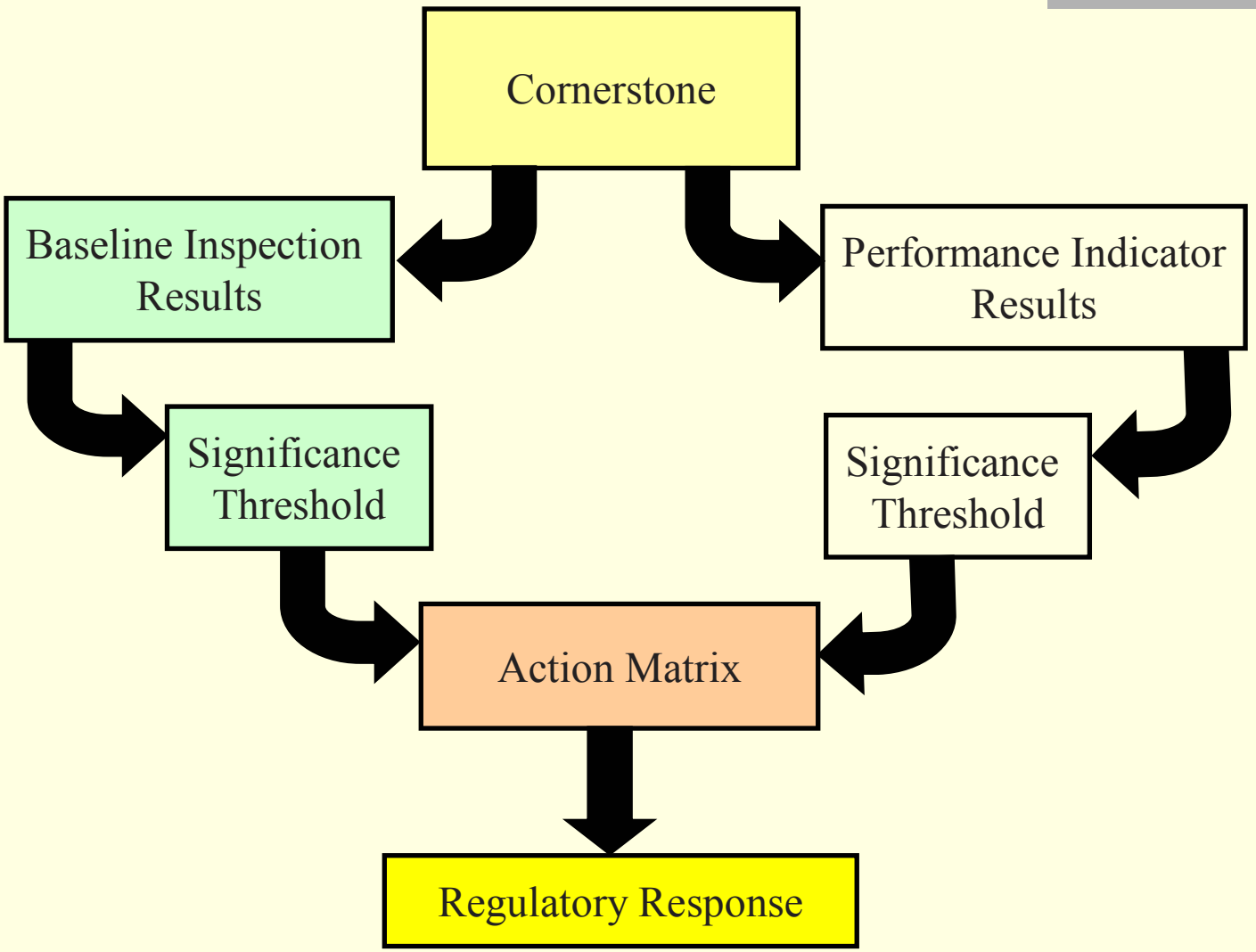
REGULATORY FRAMEWORK



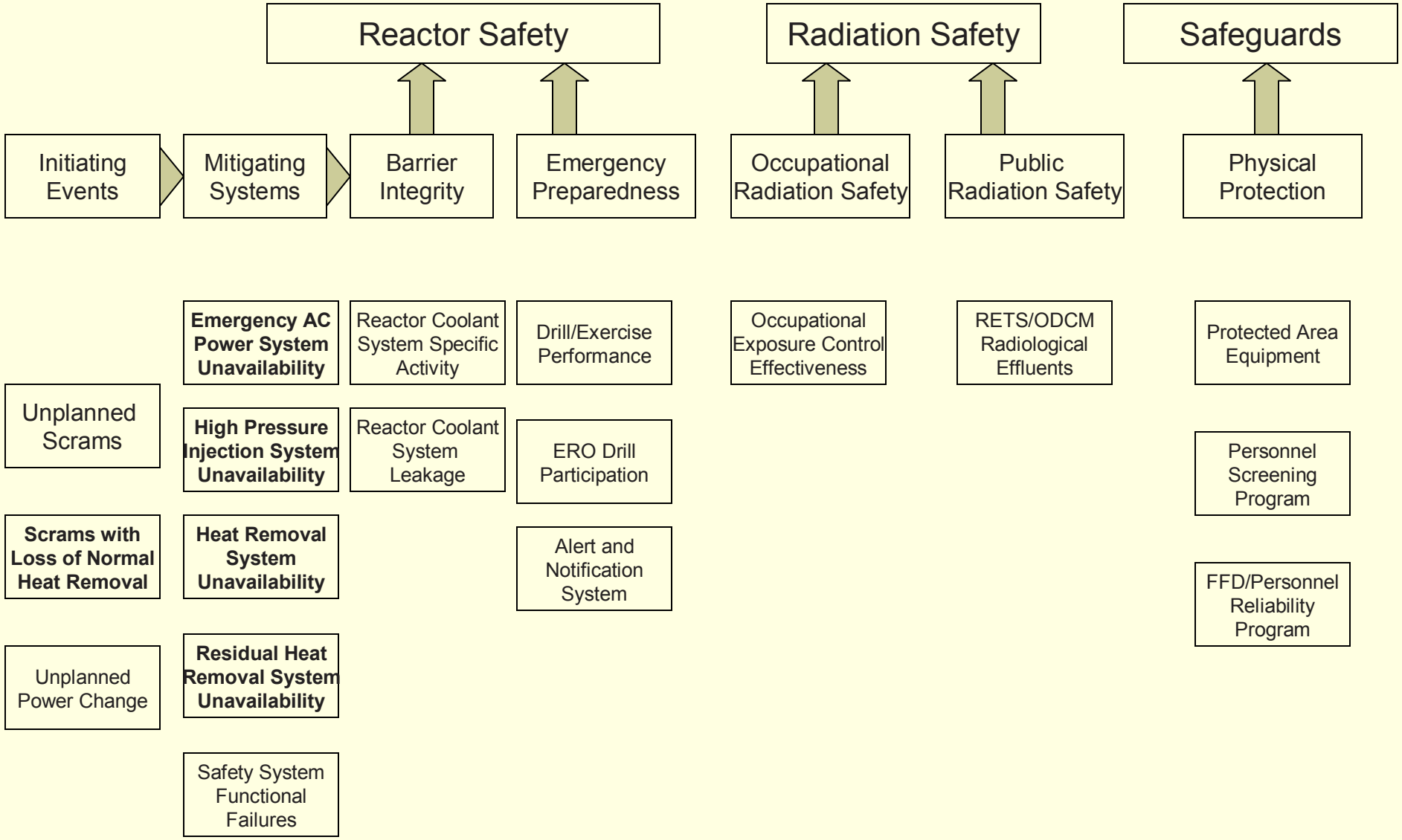
Use of Risk Information

- ROP based on a dual system-performance indicators (PIs) using objective data, and focused inspection that complements the PIs
- Used to identify PIs, and to establish thresholds for regulatory action commensurate with safety significance
- Used to focus the inspection program on those issues important to safety
- A Significance Determination Process (SDP) was developed to assess the safety significance of inspection findings to determine the appropriate regulatory response
- Used for assessing significance of plant events and degraded conditions to help determine appropriate NRC response

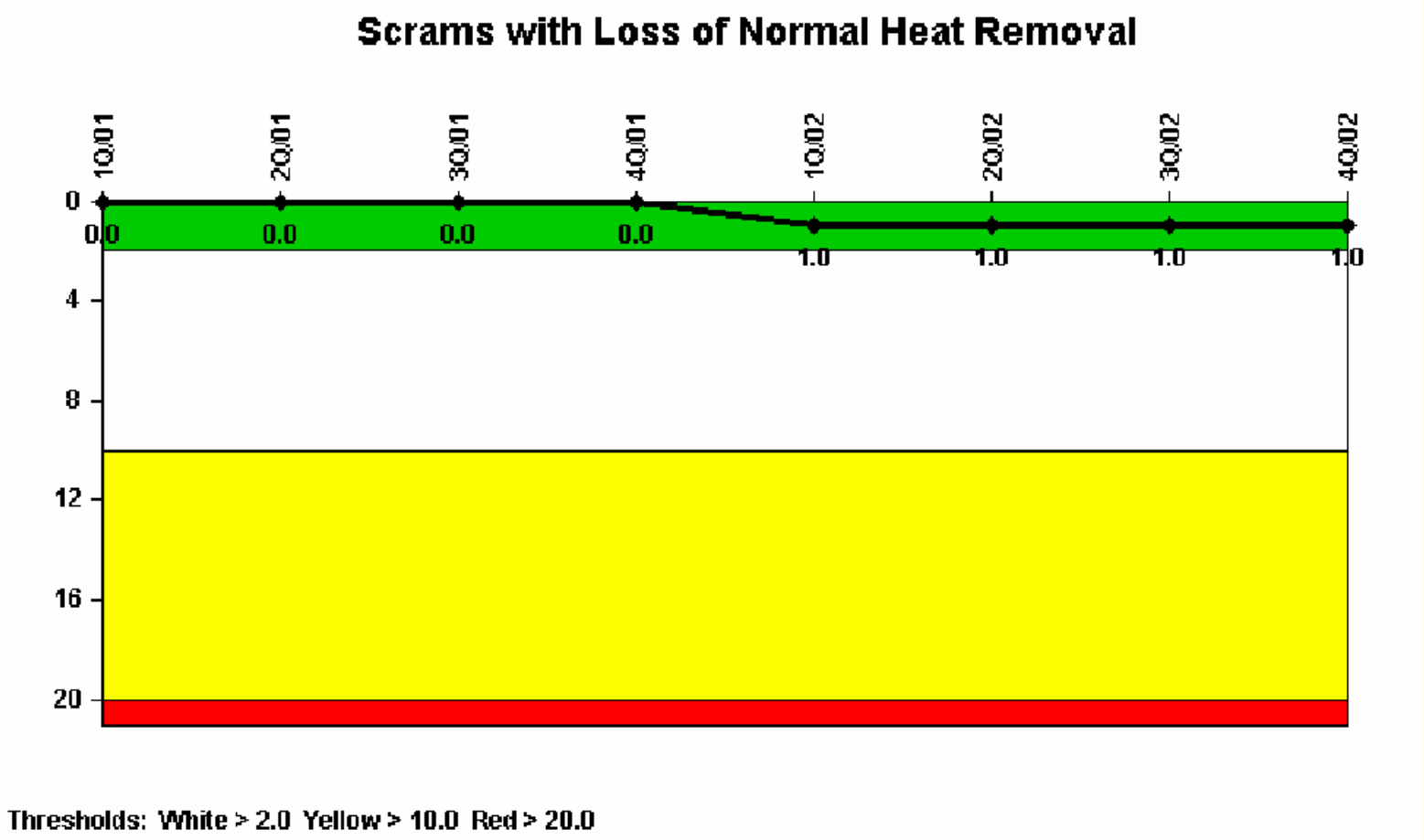
REACTOR OVERSIGHT PROCESS



Performance Indicators in the Seven Cornerstones



TYPICAL PERFORMANCE INDICATOR



SIGNIFICANCE DETERMINATION PROCESS (SDP)

- To characterize the significance of an inspection finding consistent with the NRC regulatory response thresholds used for performance indicators (PIs). SDP philosophy espoused in NRC Inspection Manual Chapter 308

- To provide a framework for discussing and communicating the potential significance of inspection findings.

- To provide a basis for assessment of licensee performance and enforcement actions associated with an inspection finding.

SDP is a Resource Allocation Management Tool

Significance Determination Process Overview

- Reactor Safety SDPs (mostly risk-informed)
  - At-power and shutdown findings
  - Special SDPs for emergency preparedness, fire protection, containment integrity, licensed operator requalification, steam generator tube integrity, and plant configuration control
  - Spent fuel pool/dry cask storage SDP scheduled to be issued in 2005

- Radiation Safety (deterministic)
  - Occupational Radiation Safety
  - Public Radiation Safety

- Safeguards (risk-informed)
  - Physical Protection

LEVEL OF SIGNIFICANCE ASSOCIATED WITH PERFORMANCE INDICATORS AND INSPECTION FINDINGS

- Green - very low risk significance – baseline inspection
- White - low to moderate risk significance – supplemental inspection (IP 95001)
- Yellow - substantive risk significance – supplemental inspection (IP 95002)
- Red - high risk significance – supplemental inspection (IP 95003)

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

Reactor Safety Significance Determination Process

- Three phase process
  - Phase 1 screens issues to Green, Phase 2, and/or Phase 3
  - Phase 2 evaluates issues using plant-specific risk-informed inspection notebooks (simplified event-tree approach) that are typically conservative. Notebooks incorporate licensee PRA information when appropriate. All 71 notebooks have been "benchmarked" against utility PRA models.
  - Phase 3 is a more detailed review using independent risk tools (e.g., SPAR models)

- Phases 1 and 2 are generally performed by inspection staff, with assistance of a Senior Reactor Analyst (SRA), where necessary.

- Phase 3 is performed by a Senior Reactor Analyst or a risk analyst.

ASSESSMENT

- Performance Indicators and Inspection Findings Are Combined for an Overall Assessment of Plant Performance.
- Action Matrix Is Used to Assess Performance and Determine Regulatory Actions.
- Quarterly, Mid-Cycle and End of Cycle Assessments Are Performed for Each Licensee

ACTION MATRIX CONCEPT

Licensee Response	Regulatory Response	Degraded Cornerstone	Multiple/Rep. Degraded Cornerstone	Unacceptable Performance

Increasing Safety Significance

Increasing NRC Inspection Efforts

Increasing NRC/Licensee Management Involvement

Increasing Regulatory Actions

SDP Challenges

- Level of analysis required for decision-making
- Ability to reasonably evaluate equipment importance and contribution to mitigate external event initiators (e.g., internal & external flooding, fire, earthquakes)
- Timeliness of SDP analyses

ROP Challenges

- Assessment of Safety Culture
  - Currently reviewing practices to incorporate into ROP
- Proper Number and Scope of Inspections
  - Add/Shed process being developed
- Cross-Cutting Issues
  - Human performance, problem identification and resolution, safety-conscious work environment

ROP References
Inspection Manual Chapters

- IMC 305, Operating Reactor Assessment Program
- IMC 307, Reactor Oversight Process Self-Assessment Program
- IMC 308, ROP Basis Document
- IMC 350, Oversight of Operating Reactor Facilities in a Shutdown Condition with Performance Problems
- IMC 608, Performance Indicator Program
- IMC 609, Significance Determination Process
- IMC 612, Power Reactor Inspection Reports
- IMC 1245, Qualification Program for the Office of Nuclear Reactor Regulation Programs
- IMC 2515, Light-Water Reactor Inspection Program -- Operations Phase
- Website address - http://www.nrc.gov/reading-rm/doc-collections/insp-manual/manual-chapter/index.html

U.S. NRC Policy
Incident Investigations

- Ensure that significant operational events are investigated in a timely, objective, systematic, and technically sound manner; factual information is documented and the cause or causes are determined.

Significant Operational Event

- Any radiological, safeguards, or other safety-related operational event that poses an actual or a potential hazard to public health and safety, property, or the environment.

- Examples include:
  - Declaration of a site area emergency
  - Exceeded a Safety Limit
  - Led to a significant (i.e., 5 times regulatory limit) radiological exposure

Event/Condition Response NRC Programs

- Event Response/Followup – Management Directive 8.3
- Significance Determination Process in the Reactor Oversight Process
- Accident Sequence Precursor (ASP) Program
- Operating Experience Program

Event Followup Inspection Procedure 71153

- Objective – Evaluate events and degraded conditions to provide input for determining need for Incident Investigation Team, Augmented Inspection Team, or Special Inspection
- Resident Inspectors communicate input to Regional Senior Reactor Analyst (SRA) and NRC Management
- Key role of inspector is to communicate plant configuration at time of event noting any complicating factors (e.g., human performance problems, unexpected plant responses, etc.)

Event Response Management Directive 8.3

- Quickly Determine Risk Significance of Actual Events that Occurred and Provide a Risk Input for Appropriate NRC Response

- Tools Used
  - Standardized Plant Analysis Risk (SPAR) Models
  - Quantitative and Qualitative Risk Insights from Previous Studies (e.g., Station Blackout, ATWS, etc.)
  - Licensee's (utility) Plant-Specific PRA Results, if Available

Event Response Management Directive 8.3 – cont.

- Risk Metrics Used:
  - Conditional Core Damage Probability (CCDP for events, Incremental CCDP (ICCDP) for conditions)
  - Conditional Large Early Release Probability (CLERP for events, ICLERP for conditions), if necessary

- Risk Analysts from NRC Headquarters and the Regional Office Involved Cooperatively for Consensus ("One Voice") Risk Input

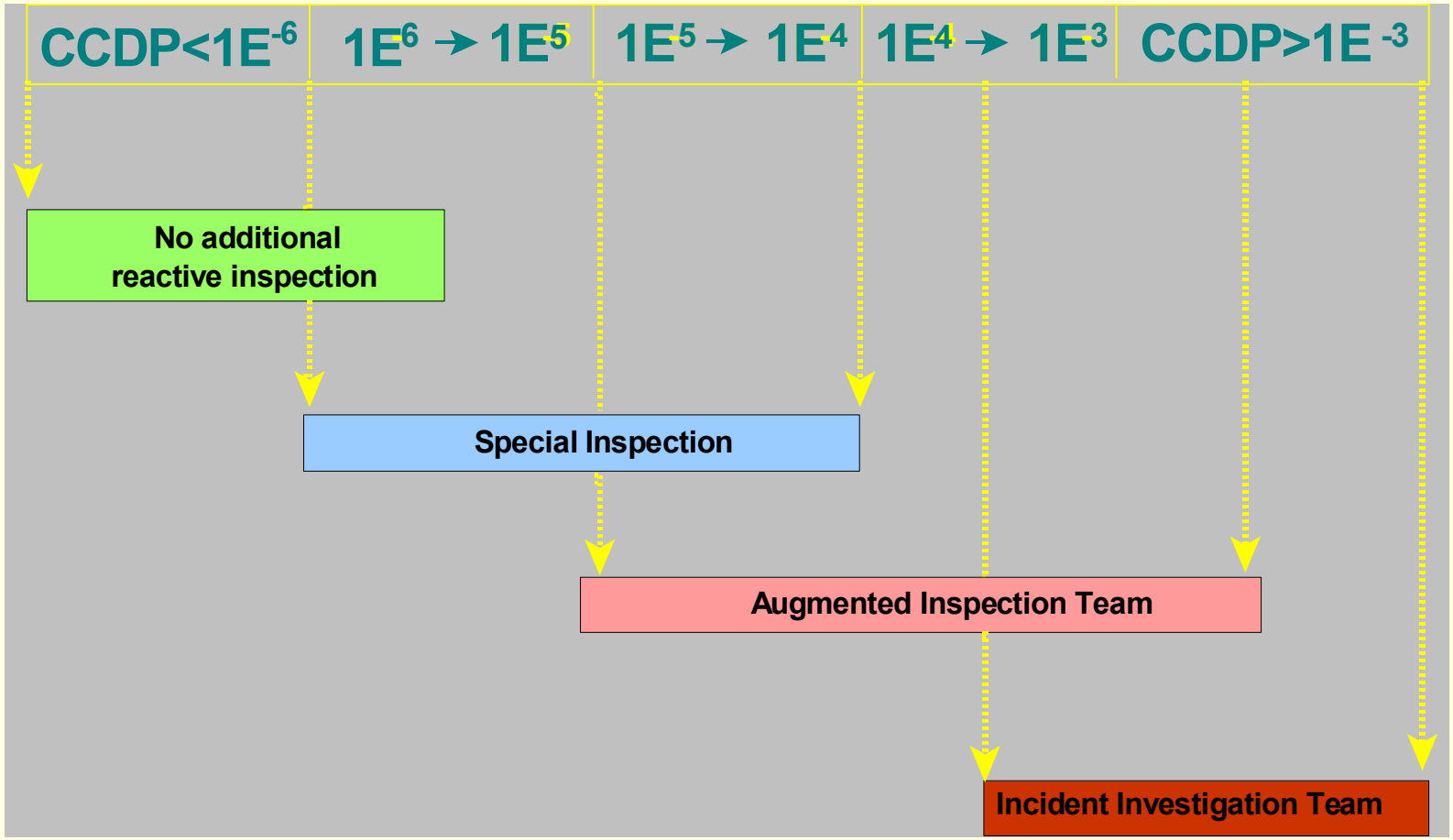
Deterministic Criteria for Event/Condition Response

- Involved plant operations that exceeded or not included in design bases
- Major deficiency in design, construction or operation having potential generic safety implications
- Significant loss of fuel integrity, primary coolant pressure boundary, or primary containment boundary
- Loss of safety function or multiple failures in systems to mitigate event

Deterministic Criteria for Event/Condition Response – cont.

- Possible adverse generic implications
- Significant unexpected system interactions
- Repetitive failures/events involving safety-related equipment or operations deficiencies
- Concerns with licensee operational performance

Graded Event Response vs. Conditional Core Damage Probability (CCDP)



USNRC Maintenance Rule (MR)

- 10CFR50.65 issued July, 1991 “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants”
- MR effective July 1996
- Establishes expectations for balancing structures, systems, & components (SSCs) unavailability with SSCs reliability (Rule sections a(1-3))
- Establishes expectations for assessing configuration risk management when removing equipment from service (Rule section a(4))

Maintenance Rule Overview

- Utilities implemented MR sections a(1 through 4) via guidance documents:
  - NUMARC 93-01
  - NRC Regulatory Guide 1.160
- PRA not specifically required by MR; however, utilities generally used PRA with deterministic criteria to scope SSCs into the MR program and establish performance criteria
- MR a(4) recently made a regulatory requirement

Maintenance Rule a(4)

- Licensees ‘required’ to perform MRa(4) risk assessments
- Risk thresholds and consequential actions established by Section 11 of NUMARC 93-01, Revision 3 (also refer to NRC Regulatory Guide 1.182)
- Current challenges are treatment of external event initiators for configuration risk management models (which are based on licensee PRAs)

Maintenance Rule a(4)
Risk thresholds for equipment out of service

ICDP		ILERP
$> 10^{-5}$	- Plant configuration should not normally be entered voluntarily	$> 10^{-6}$
10^{-6} - 10^{-5}	- Assess non-quantifiable factors - Establish *risk management actions*	10^{-7} - 10^{-6}
$< 10^{-6}$	- Normal work controls	$< 10^{-7}$

Risk Informed Technical Specifications and License Amendments

- PRA application in Technical Specifications has been around since the late 1980s in current reactor and new-advanced reactor design certification
- Regulatory Guide 1.174 and 1.177 provide framework for risk-informed license amendments (a common application is EDG TS allowed outage time extensions)
- Current challenge in Technical Specification Initiative 4b (flexible allowed outage times)

The End

Questions & Answers…..