SESSION 1-4

Risk-Informed Regulatory Programs

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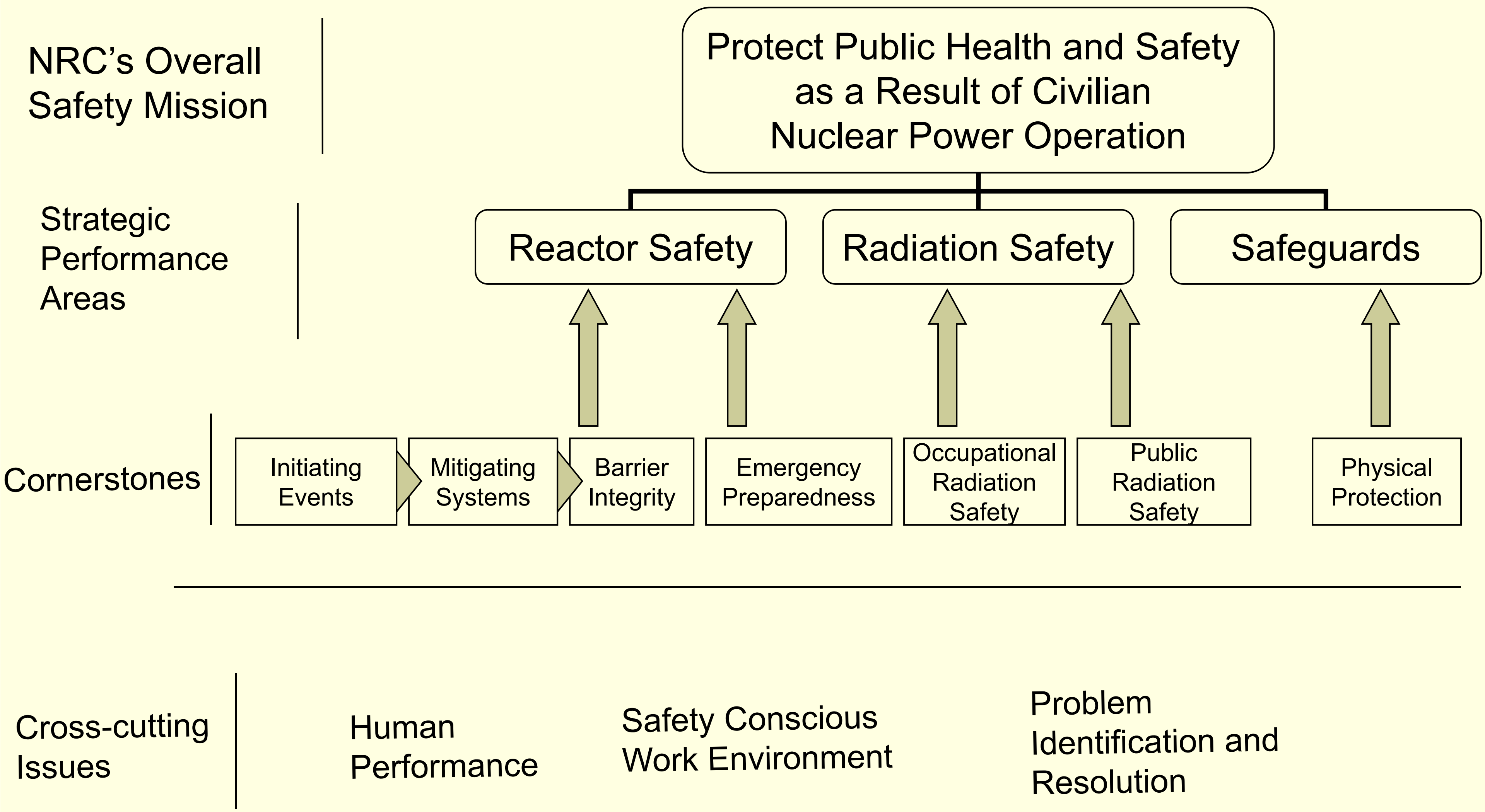
Presentation Outline

- Risk-informed regulatory programs in selected areas of USNRC regulatory activities
  - Reactor Oversight Process (ROP)
  - Significance Determination Process (SDP)
  - Incident Response
  - Maintenance Rule
  - Mitigating Systems Performance Index

USNRC Reactor Oversight Process

- Provides Risk-informed Focus in Inspections and Safety Management Activities
- Applies Greater Regulatory Attention to Facilities with Performance Issues While Maintaining a Base Level of Regulatory Attention on Plants That Perform Well
- Allows Greater Use of Objective Measures of Plant Performance
- Gives Timely and Understandable Assessments of Plant Performance to Industry and Public Stakeholders
- Avoids Unnecessary Regulatory Burden
- Responds to Violations in a Predictable and Consistent Manner That Reflects the Safety Significance of the Violations

REGULATORY FRAMEWORK



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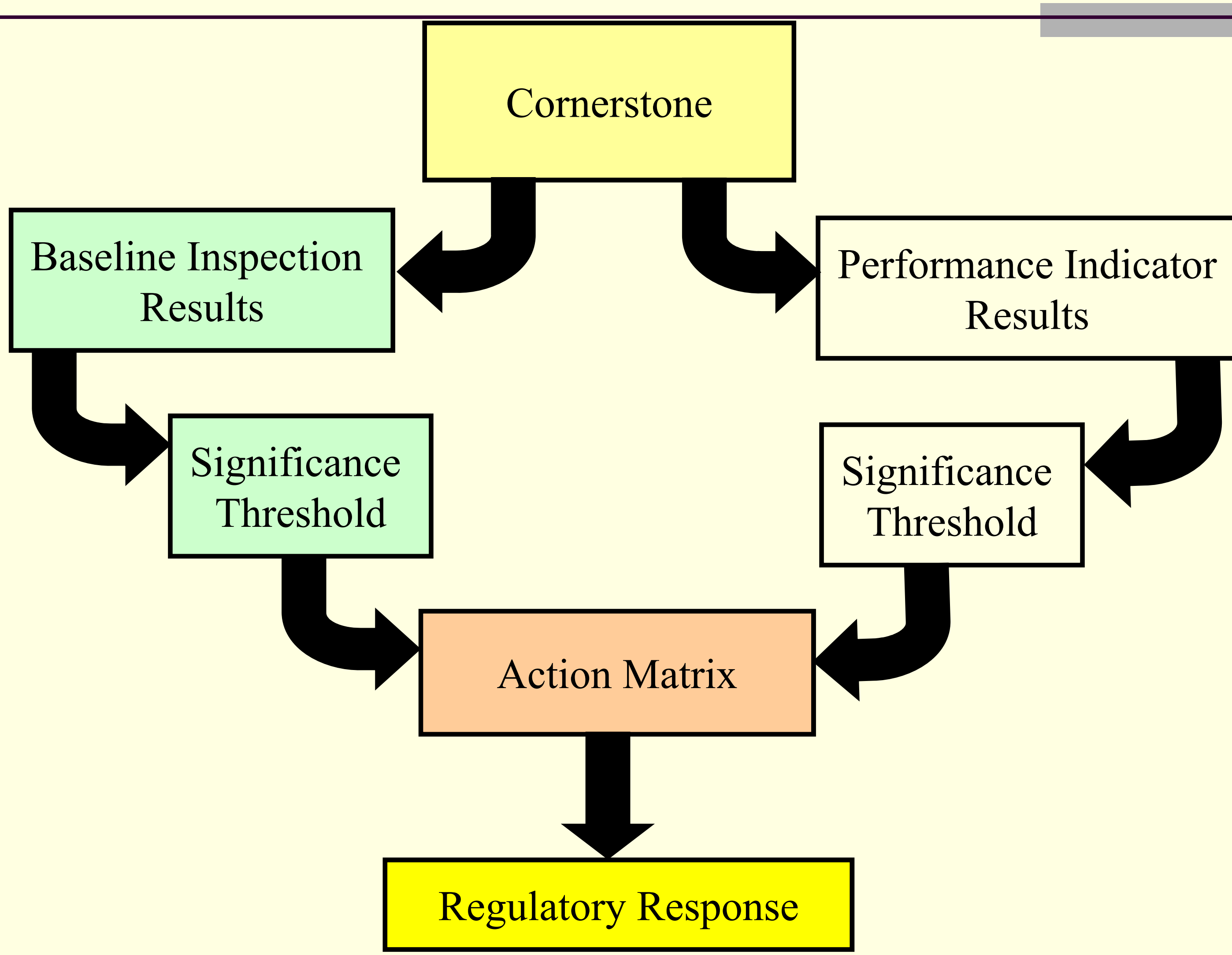
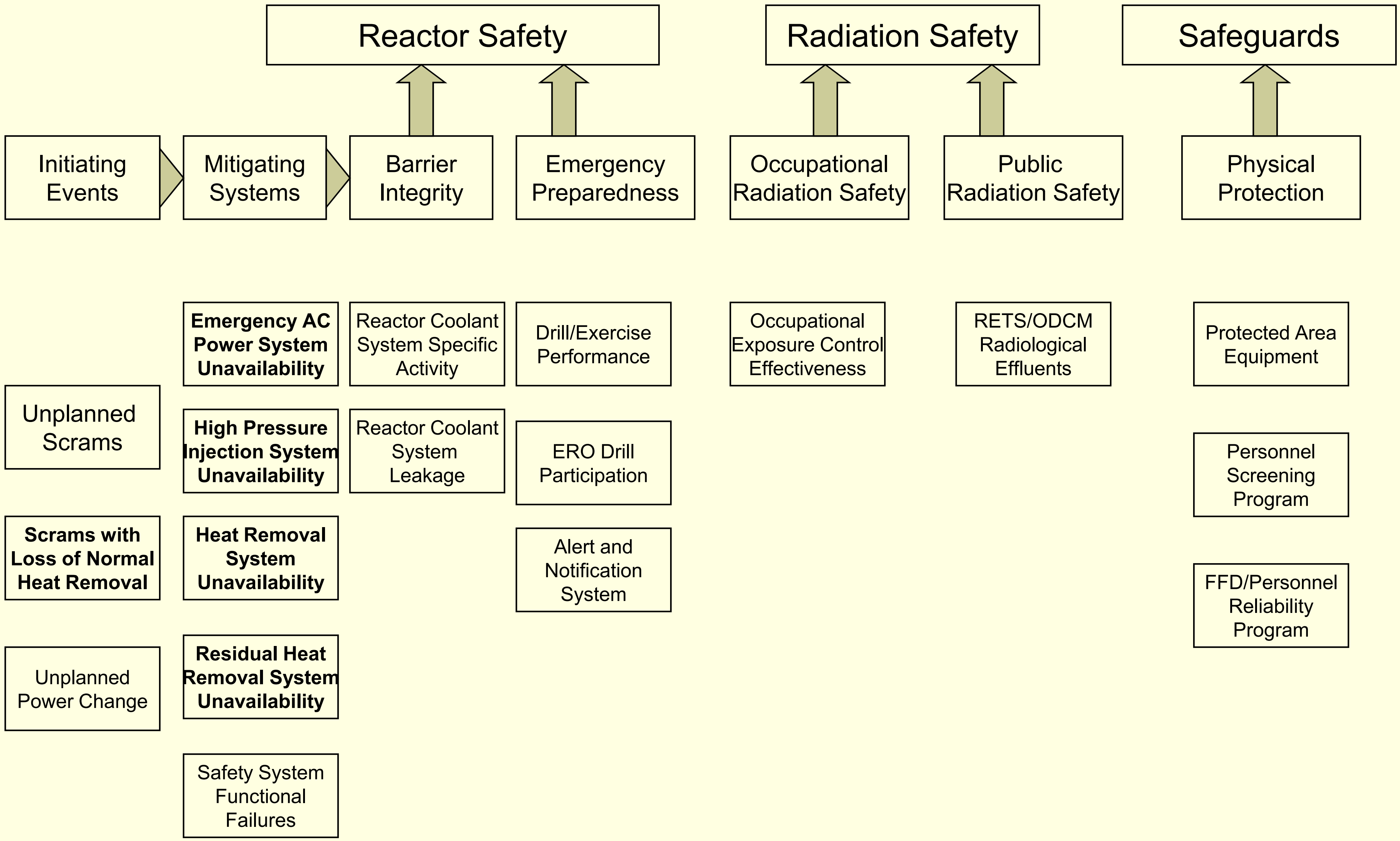
Use of Risk Information

- 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
- 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

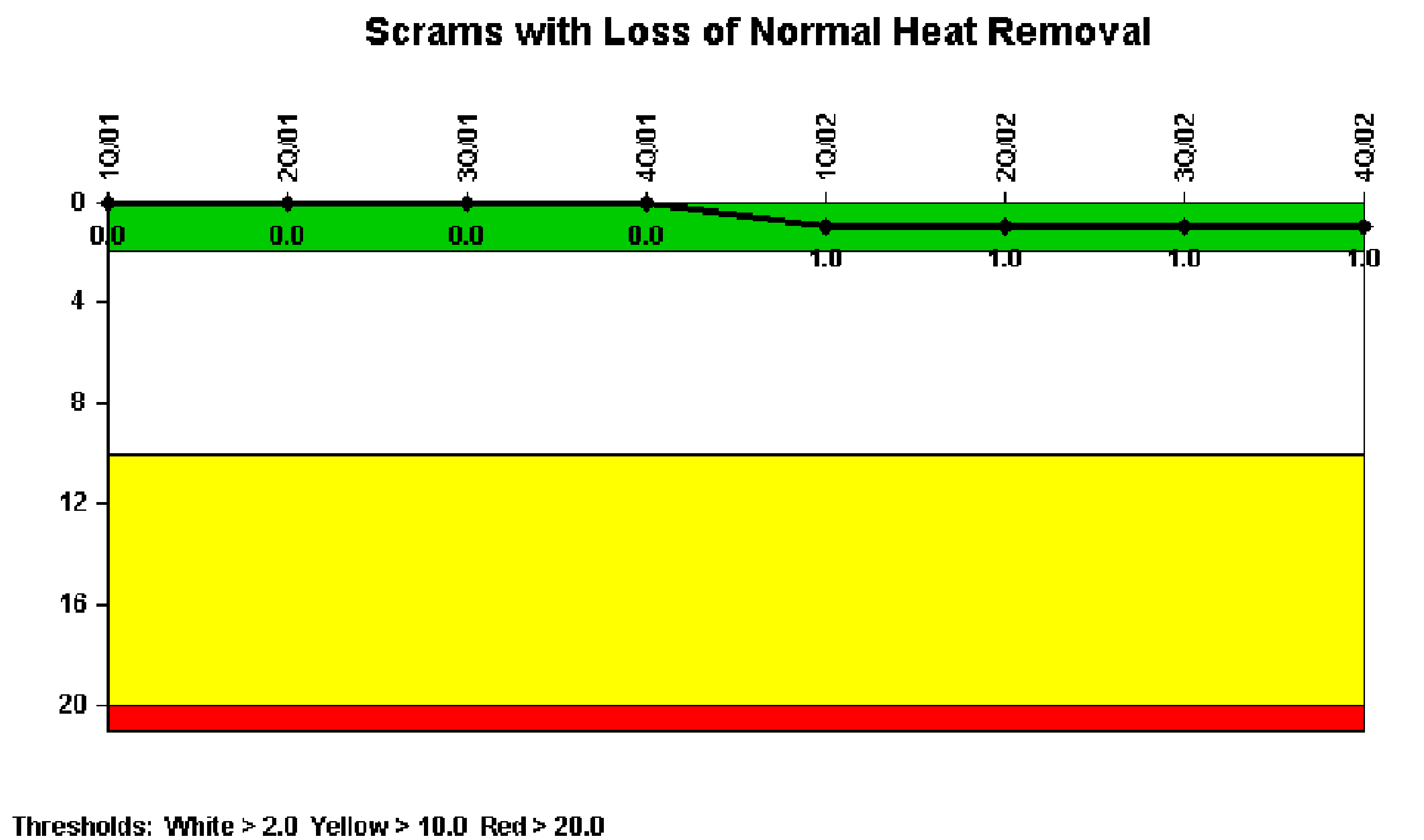
REACTOR OVERSIGHT PROCESS



Performance Indicators in the Seven Cornerstones



TYPICAL PERFORMANCE INDICATOR



SIGNIFICANCE DETERMINATION PROCESS (SDP)

- Characterize the significance of inspection findings in support of the Reactor Oversight Process
- Provide stakeholders with an objective and common framework for communicating the potential significance of inspection findings
- Provide a basis for assessment and enforcement actions associated with inspection findings, thereby reducing uncertainty
- Provide the staff with plant specific risk information for use in risk-informing the inspection program

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
- 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 (normal range)
- **very low risk significance**
- baseline inspection

White (outside normal range)
- **low to moderate risk significance**
- supplemental inspection (IP 95001)

Yellow (significant reduction in safety margin)
- **substantive risk significance**
- supplemental inspection (IP 95002)

Red (significantly outside design basis)
- **high risk significance**
- supplemental inspection (IP 95003)

$\Delta CDF < 1E\text{-}6$	Green
$1E\text{-}6 < \Delta CDF < 1E\text{-}5$	White
$1E\text{-}5 < \Delta CDF < 1E\text{-}4$	Yellow
$\Delta CDF > 1E\text{-}4$	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 that are typically conservative, yet representative of licensee 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 other risk analyst

Phase 1 SDP for At-Power Inspection Findings

- Prior to conducting a Phase 1 Screening, the performance deficiency must be of greater than minor significance

- The Phase 1 Screening Worksheet contains decision logic to determine if the deficiency can be characterized as Green without further analysis

- Deficiencies generally screen to Green if initiating event frequencies and total function of mitigating and containment systems are not lost

- Some deficiencies immediately screen to Green based on their low impact to overall plant risk (e.g., radiological barrier systems such as building ventilation)

Phase 2 SDP for At-Power Inspection Findings

- Step 1- Select Initiating Event Scenarios
- Step 2 - Estimate the Initiating Event Likelihood
- Step 3 – Determine the Remaining Mitigation Capability
- Step 4 - Estimate Risk Significance of Inspection Finding
- Step 5 – Screen for External Event Contribution
- Step 6 – Screen for Large Early Release Frequency (LERF) significance

Phase 3 SDP

- **Risk Significance Estimation Using Risk Basis That Departs from the Phase 1 or 2 Process**

  -If necessary, Phase 3 will refine or modify, with sufficient justification, the earlier screening results from Phases 1 and 2

  -Phase 3 also addresses findings that cannot be evaluated using the Phase 2 process (e.g., external event contributors)

  -Phase 3 analysis uses appropriate PRA techniques and rely on the expertise of NRC risk analysts

Phase 3 SDP

Phase 3 analysis package includes the following:

- Phase 1 and 2 results
- PRA tools used for the Phase 3 assessment
- Affected accident sequences
- Influential assumptions
- Sensitivity of results to each assumption
- Contributions of greatest uncertainty factors

Risk effects of Large Early Release Frequency, internal flooding and external events are also evaluated

Phase 3 analysis is documented in a Significance and Enforcement Review Panel (SERP) package and presented to SERP members for a preliminary decision

SERP Process

Preliminary SERP decision presented to licensee in a “Choice” letter

-Licensee has choice to respond by letter or attend a Regulatory Conference

-Licensee may accept preliminary result

If preliminary result is changed due to new information or insights, SERP reconvenes and determines final significance of finding

- final significance letter sent to licensee describing finding and regulatory significance

SDP Challenges

- Timeliness goal of < 90 days – use of <u>best available information</u> for decision-making
- Level of risk knowledge needed for risk-informed inspectors
- Enhancement of Phase 3 SDP risk analysis tools and guidance – Risk Assessment Standardization Project (RASP)

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

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 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

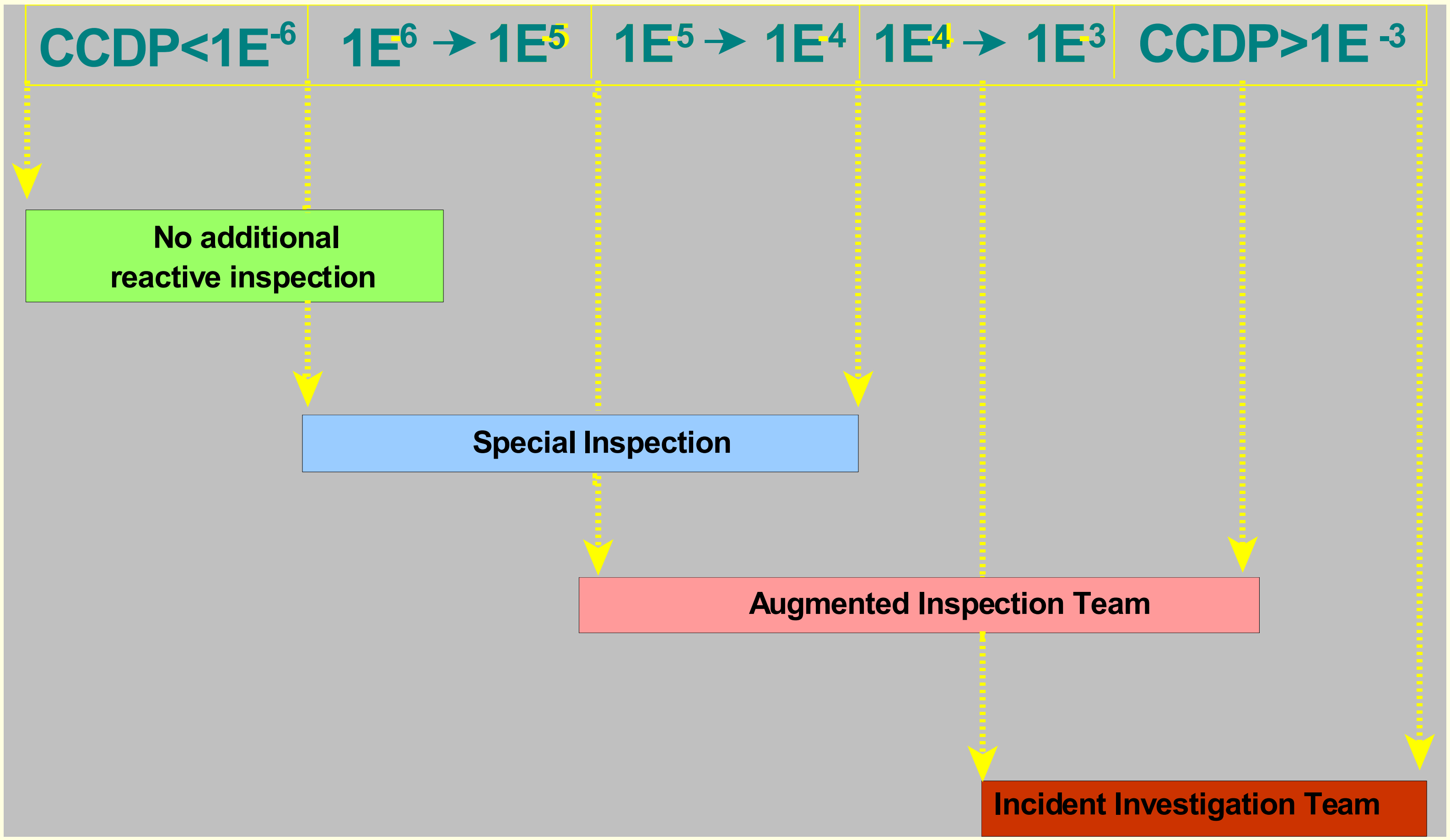
Deterministic Criteria for Event/Condition Response

- 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

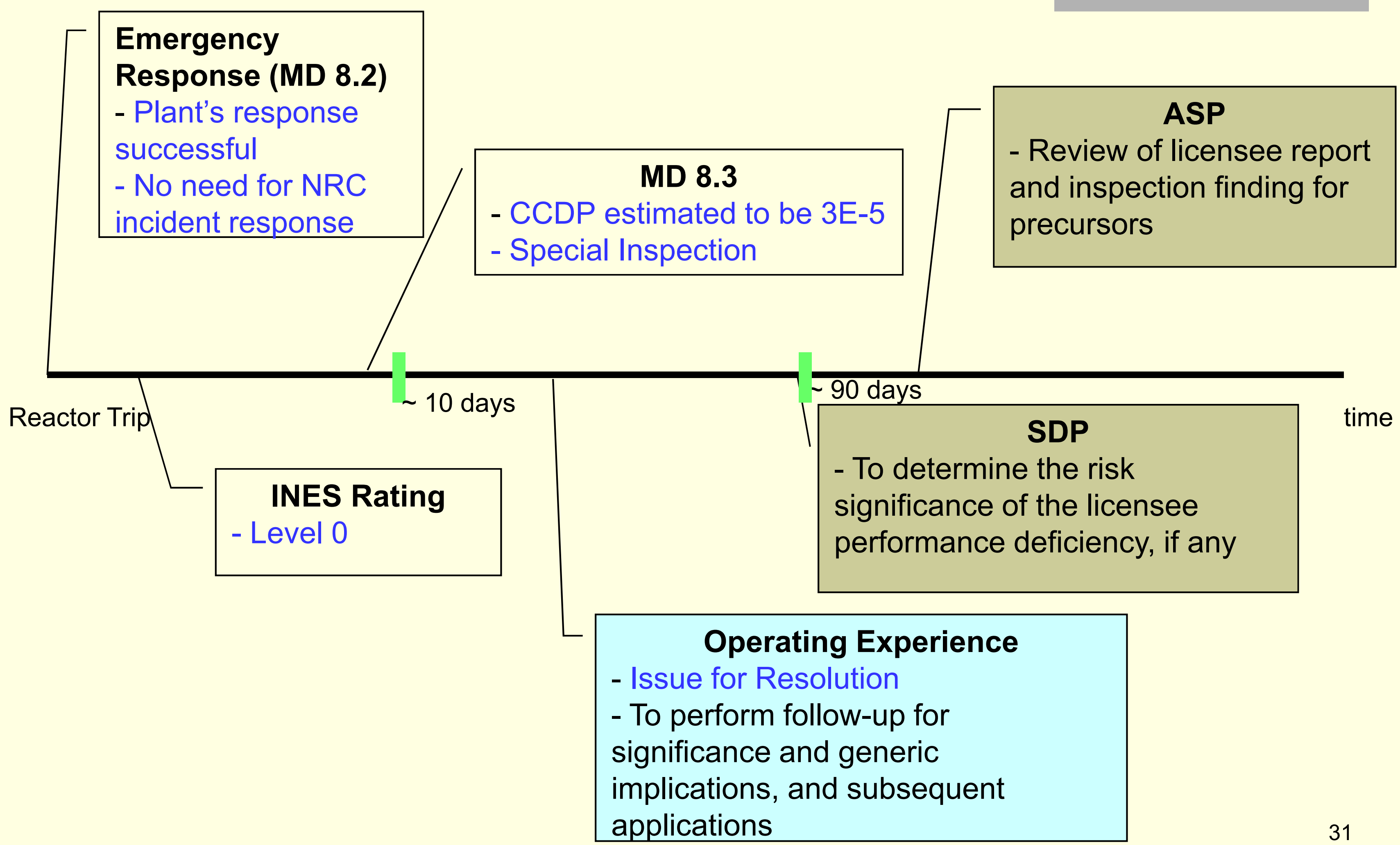
Graded Event Response vs. Conditional Core Damage Probability (CCDP)



Example: Plant X Automatic Reactor Trip

- Summary
  - A routine test of the alternate power source for the Control Rod Drive (CRD) System was in progress when power to the CRD system was interrupted
  - Reactor trip and AC power transferred to the Start-up source (switchyard)
  - The Main Steam Header pressure control setpoint did not automatically increase for post-trip RCS temperature control
  - The RCS cooled down to approximately 536F (versus a normal post-trip temperature of approximately 555F), reducing RCS pressure to the actuation setpoint for Engineered Safeguards Channels 1 and 2. This started the High Pressure Injection pumps in ECCS mode, caused partial containment isolation and initiated start-up of both Plant X Hydro Units (emergency power)
  - Pressurizer level decreased off-scale low and was recovered prior to securing the High Pressure Injection pumps

Example: Plant X Automatic Reactor Trip (Continued)



USNRC Maintenance Rule (MR)

- 10 CFR 50.65 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) regulatory requirement effective as of November 28, 2000

Maintenance Rule a(4)

- Licensees ‘required’ to perform MR a(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	$< 10^{-7}$

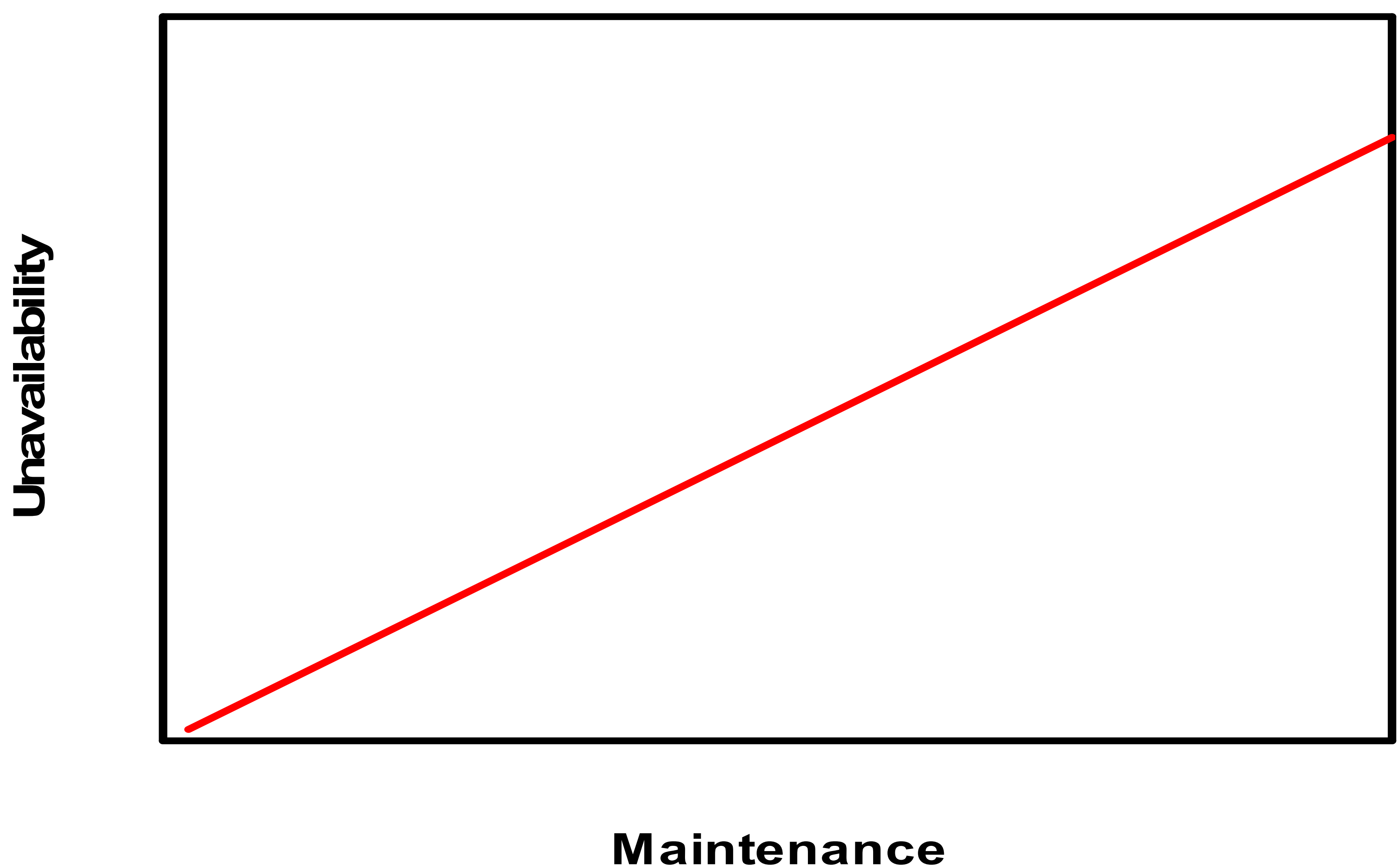
What is MSPI ?

A measure of the deviation of plant system unavailability and component unreliabilities from baseline values, weighted by plant-specific risk importance measures.

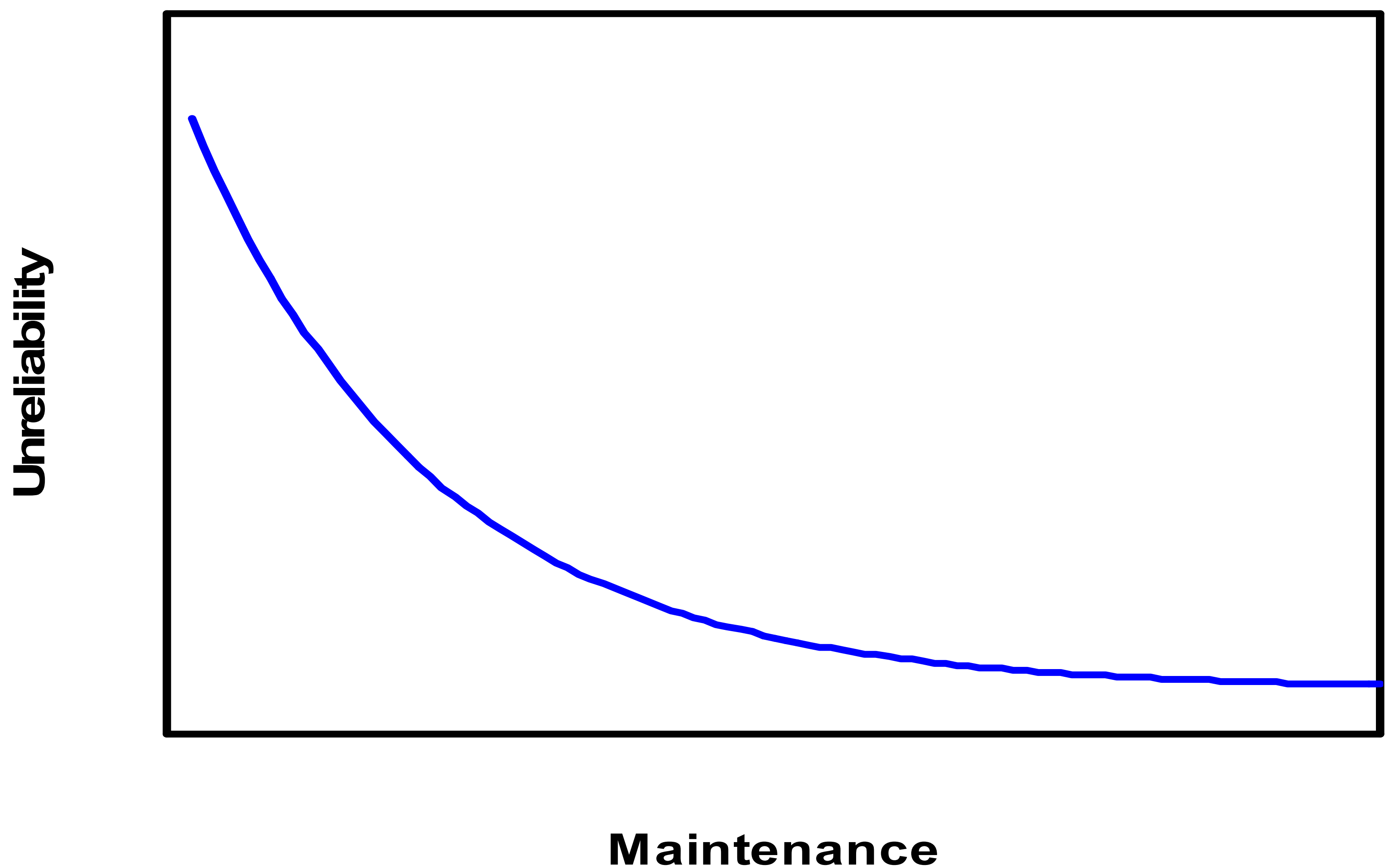
Reason for Change

- Problems with current Safety System Unavailability Performance Indicator
- Balances need for maintenance (availability) with equipment performance (reliability)
- Eliminates cascading cooling water system problems

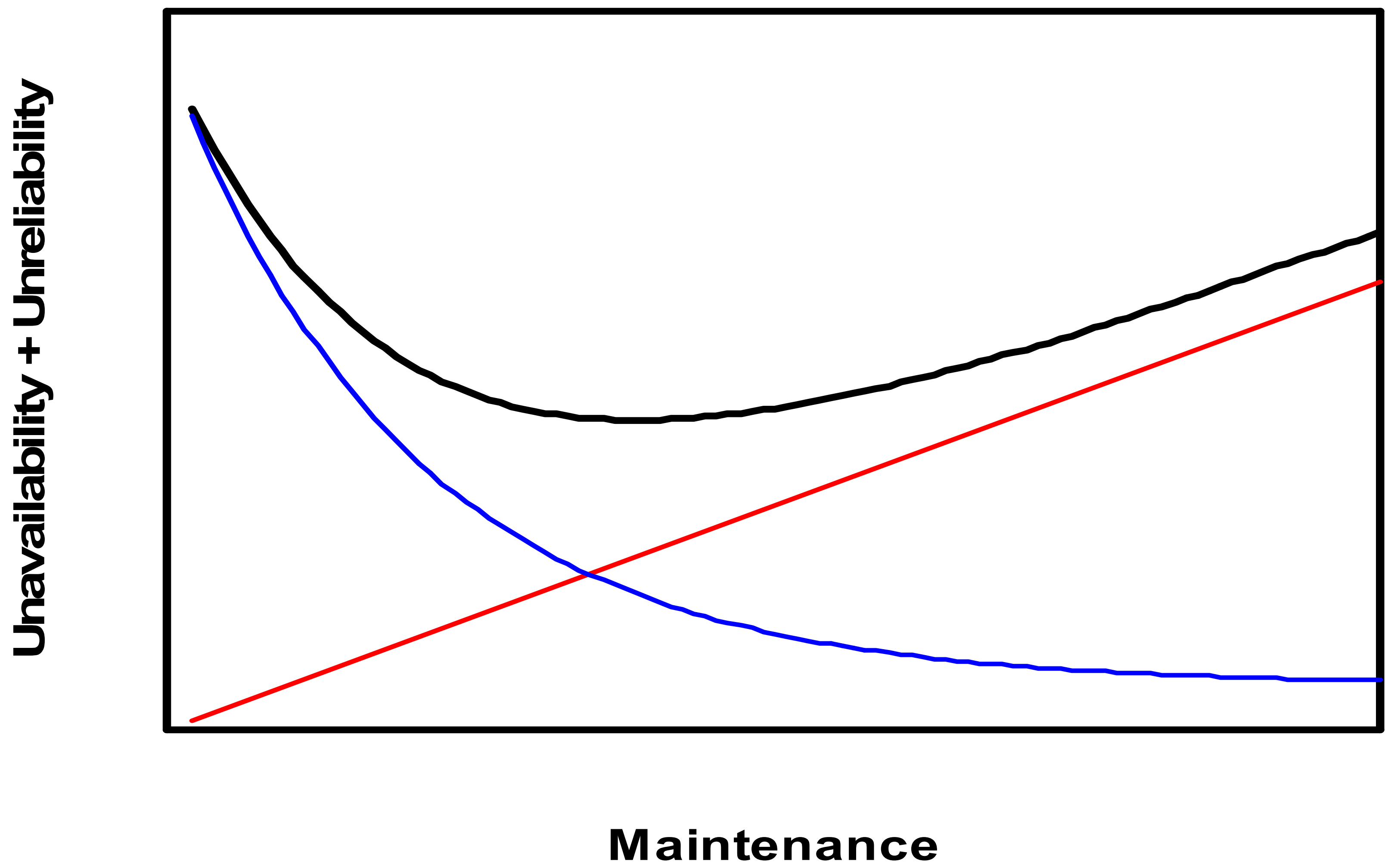
More maintenance increases Unavailability...



...but it decreases Unreliability…



...and there is an optimum!



Attributes of MSPI

- Risk-informed, performance-based.
- Allows trade-offs between unreliability and unavailability to optimize system performance.
- No penalty for on-line preventive maintenance hours up to pre-planned baseline.
- Reflects plant-specific design and operation.
- Addresses issues such as treatment of fault exposure time in Safety System Unavailability Performance Indicator.

Monitored Systems

BWRs	PWRs
High Pressure Coolant Injection/Core Spray	High Pressure Safety Injection
RCIC or Isolation Condenser	Auxiliary or Emergency Feedwater
Residual Heat Removal	Residual Heat Removal (may include containment spray)
Emergency AC power	Emergency AC power
Cooling Water Support Systems	Cooling Water Support Systems

The End

Questions & Answers…..