IAEA/USA Workshop

David Passehl
Senior Reactor Analyst
USNRC, Region III

Overview

- Introduction to Region III
  - Region III Organization
- Region III Reactor Safety Program
  - Reactor Oversight Process
  - Reactor Inspections
  - Incident Response
  - Incident Investigation
- Inspection Planning
- Component Design Basis Inspections
- Fire Protection Inspections
- Other Inspections

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Region III Reactor Program Arenas

General

Region III conducts inspection and licensing activities for 16 nuclear power plants in the Midwestern states of Illinois, Iowa, Michigan, Minnesota, Ohio and Wisconsin.

Functional guidance and direction for the reactor oversight process is provided by the Office of Nuclear Reactor Regulation.

Organizationally, in Region III, these functions are implemented by the Division of Reactor Projects and the Division of Reactor Safety, and are carried out by the region's resident and region-based inspection staff, and operator licensing examiner staff.

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Region III Reactor Program Arenas

Reactor Safety Program Elements

Region III implements the NRC's reactor safety program by utilizing the following program elements:

Baseline Inspections
- **Minimum inspection level received by all facilities**
- **Conducted by resident and region-based inspectors**

Supplemental Inspections
- **Based on licensee performance**
- **Focused inspections of problems and issues**
- **Conducted by resident and/or region-based inspectors**
- **Prescribed by the Action Matrix**

Temporary Instruction (TI) Inspections
- **For generic safety issues; one time inspections**

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Region III Reactor Program Arenas

Reactor Safety Program Elements (cont):

Event Follow-Up Inspections
- Special Inspections
- Augmented Inspections
- Incident Investigation Inspections

Allegation Review and Follow-Up

Enforcement/Significance Determination Process

Plant Performance Assessment
- Performance Indicators
- Inspection Findings

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Region III Reactor Program Arenas

Reactor Safety Program Elements (cont):

License Renewal Inspections

New Construction Program Development

Reactor Operator Licensing

Office of Investigations (OI) – Technical Support

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Region III Reactor Safety Program

Reactor Program Inspectors – Resident Inspectors

Full-time resident inspectors are stationed at each nuclear plant site to conduct inspections of equipment and to provide close surveillance of plant operations.

Typically, one senior and one resident inspector are assigned to each site.

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Region III Reactor Safety Program

Reactor Resident Inspectors (cont)

The resident inspectors perform three primary functions:

- direct inspection
- early response to events
- knowledge of plant status

The resident inspectors arrange their schedules to perform inspections during all hours of the day; including weekends and backshifts as necessary.

The resident inspectors are knowledgeable about a multitude of engineering and science-related applications that are associated with plant operations. They are also familiar with the areas evaluated by region-based inspectors in order to identify when a potential problem exists.

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Region III Reactor Safety Program

Reactor Program (Region-based) Inspectors

Region III based inspectors review plant security, emergency planning, radiation protection, environmental monitoring, inservice inspection of mechanical components, design engineering inspections of systems and components, fire protection, construction activities, and other specialized areas. They conduct about 150-175 routine inspections a year.

In addition, Region III conducts initial operator licensing examinations for candidates put forth by facilities to be reactor and senior reactor operators and conducts reviews of facility administered licensed operator requalification programs.

Region III inspectors, in conjunction with staff from headquarters and other regional offices, conduct special team inspections. These inspections focus on a specific plant activity, like maintenance or security, an operating problem or event.

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Region III Senior Reactor Analysts

The NRC Senior Reactor Analysts (SRAs) are trained to help achieve specific expectations in support of the NRC Reactor Oversight Process (ROP).

- Support implementation of NRC's risk-informed regulatory activities.
- Evaluate the potential risk significance of plant events.
- Provide effective communication about risk with internal and external stakeholders.
- Maintain open communication channels with licensee PRA staff and with other NRC offices performing PRA or significance determination process (SDP) related functions.

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Region III Senior Reactor Analysts

- Maintain awareness of the risk assessment capabilities, licensee-generated risk insights, and NRC-generated risk insights for those licensees specifically assigned.

- Maintain regional management awareness of significant PRA or significance determination process issues and changes.

- Support risk-informed inspection planning activities, and provide leadership and assistance in various risk-informed inspection activities.

- Support inspection activities by providing advice on regulatory review of risk issues, peer review of risk assessments, and performing detailed assessment of significance of inspection findings.

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Reactor Oversight Process (ROP)

- Process built on set of safety cornerstones that embodies concept of defense-in-depth

- Licensee performance assessed through performance indicators and inspections; both focus on plant features having greatest impact on safety and overall risk

- Different NRC response taken depending on risk significance

- Responses not ad hoc; established in response matrix

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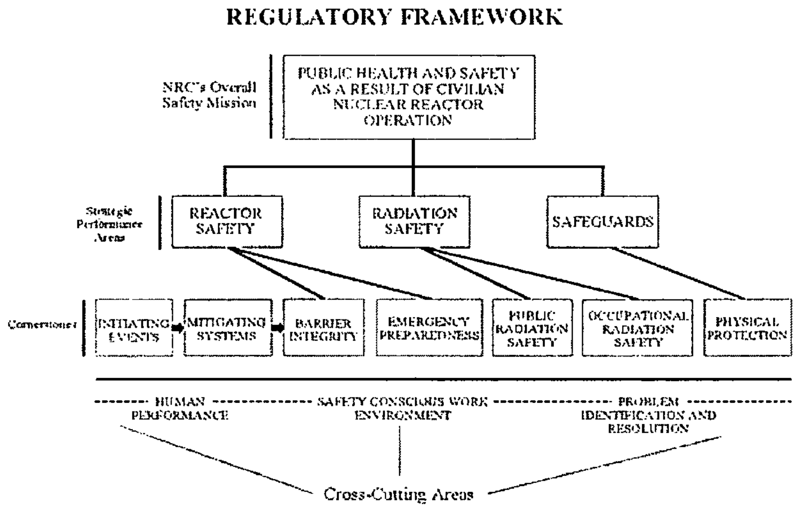
ROP Regulatory Framework



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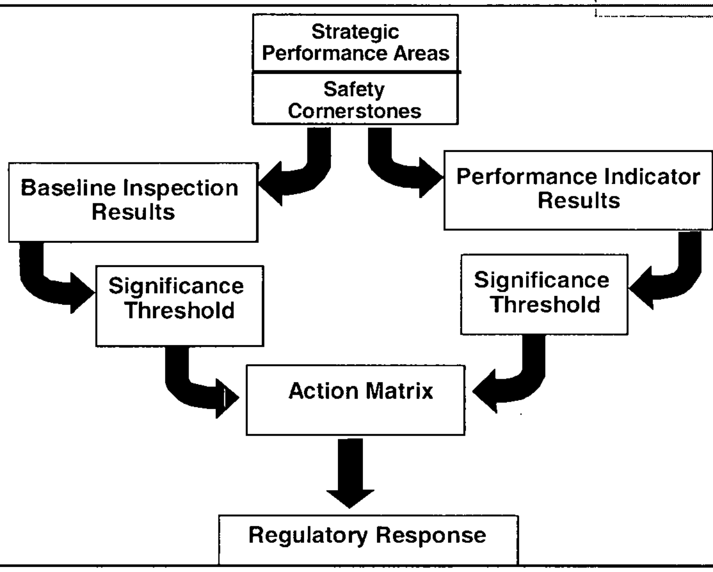
ROP Regulatory Framework



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Region III Reactor Program Arenas

- **Reactor Inspections**
  - Significance Determination Process (SDP)
  - NRC Inspection Manual Chapter 0609
  - Notice of Enforcement Discretion
  - Manual Chapter, Part 9900
- **Incident Response**
  - Ongoing Emergencies
  - Management Directive 8.2
- **Incident Investigation**
  - Events and Degraded Conditions
  - Management Directive 8.3

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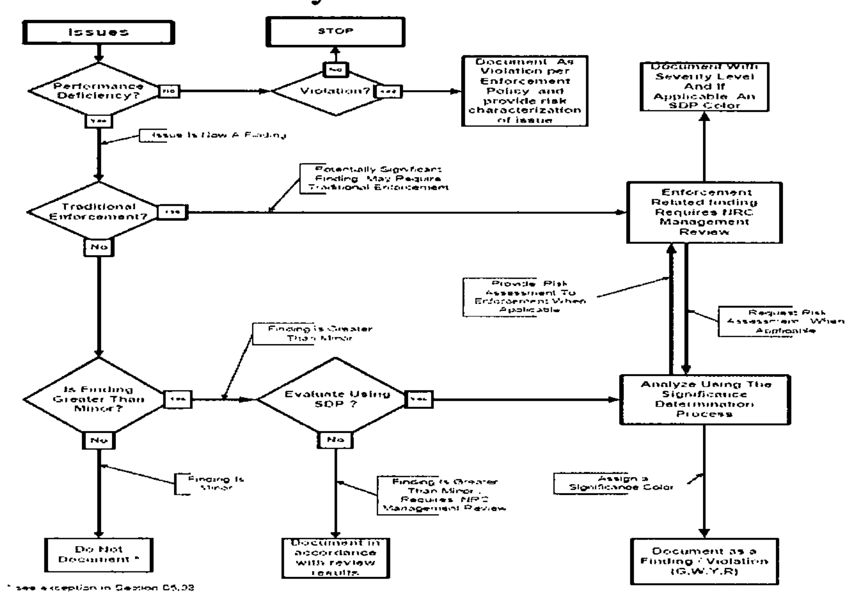
Entry into the SDP



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Performance Deficiency

- Performance Deficiency:

  An issue that is the result of a licensee not meeting a requirement or standard where the cause was reasonably within the licensee's ability to foresee and correct, and that should have been prevented.

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Performance Deficiency and Minor Finding Determination

- NRC Inspection Manual Chapter (IMC) 0612 clarifies:
  - If the issue is a PD
  - If Traditional Enforcement is appropriate, if so the Enforcement Process is followed.
  - If the issues is more than minor. (Similarity to examples in IMC 0612 Appendix E or IMC 0612 Appendix B questions)
  - If the issue can be evaluated within the SDP, if so it is transferred to the appropriate section of IMC 0609.
  - How to review and document issues that can not be evaluated in the SDP

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Reactor Significance Determination Process (at-power)

- Three Phase Process:
  - Phase 1 Screen Issues
  - Phase 2 Estimate Risk Using Plant Specific Risk-Informed Inspection Notebooks
  - Phase 3 Evaluate Risk Using Modification of Phase 2 and/or Independent Risk Tools
- Phases 1 and 2 are Generally Performed by Inspection Staff, with Assistance of a Senior Reactor Analyst (SRA), When Necessary.
- Phase 3 is Defined as ANY Departure from the Phase 2 Process, and are Performed by Risk Analysts.

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Minor Determination and Phase 1 At-Power Inspection Findings

- Minor Findings are not Normally Documented.
- Minor Determinations are Made in Accordance with NRC Inspection Manual Chapter 0612, Appendices E and B.
- Greater than Minor Findings are Processed Using the Phase 1 Screening Worksheet.
- The Screening Process is Designed to:
  - Reduce the Number of Findings Processed in Phase 2.
  - Decrease Inspection of Very Low Risk Significant Items.
  - Screen Some Deficiencies Immediately Based on Low Impact.

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Phase 2 Estimation At-Power Inspection Findings

- Findings are Evaluated Using the Risk-Informed Inspection Notebooks.
  - Rev 1 was benchmarked against the Licensee's PRA between 2001 and 2003
  - Rev 2 recently issued and was based on updated risk information
  - Trip Reports provide comparison of Notebook results and Licensee PRA calculated Risk Achievement Worth (RAW) values.

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Phase 2 Process

- Notebooks Assist the Inspectors in Identifying:
  - The Initiating Events Impacted by the Finding
  - The Accident Sequences Affected
  - The Systems Available to Perform Risk-Significant Functions
  - An Estimated Increase in Core Damage Frequency
  - Exposure Time assumed(>30 days assumes a year; 3- 30 days assumes a 10^{th} of a year and < 3 days assumes a 100^{th} of a year)

The SDP then spreads the delta-CDP over a year to get delta-CDF (per year) (same numerical value)

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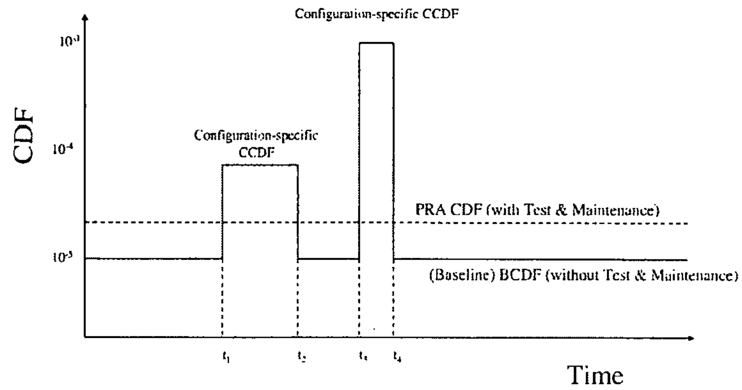
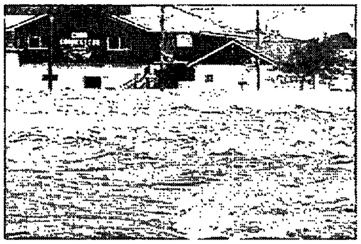
CDF Profile

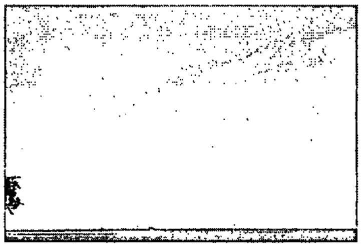


Phase 2 Process (Cont'd)

- Phase 2 results may be conservative if the actual exposure time is in the low end of the exposure band (e.g., 45 day finding will assume a years worth of exposure)
- If Green (i.e., very low risk significance), SRA prepares the analysis section writeup
- If Greater than Green, usually proceed to Phase 3 – unless agreement (SERP and Licensee) can be reached on the suitability of the Phase 2 result.

External Initiator Contribution Phases 2 and 3





- External Risk Contribution may be 10 times greater than Internal Alone
- Required in NRC Inspection Manual Chapter 0609, Appendix A, Attachment 1
- Performed for all internal results greater than 1×10^{-7}
- Predominately Fire, Flooding, and Seismic (Except High Winds Season)

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External Initiator Contribution

- The SRA will try to gather Individual Plant Examination of External Events (IPEEE) information and discuss it with the PRA staff
- The PRA staff may have more current information, including possibly a fully internal and external initiating events PRA
- External delta-CDF contributions are added to the Internal to get an estimate of the total delta-CDF.

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Large-Early Release Frequency

- Large-Early Release Frequency is a Separate Metric for Findings
- Required in NRC Inspection Manual Chapter 0609, Appendix A, Attachment 1
- Performed for all internal events sequences greater than 1 x 10^{-7}
- Currently Evacuation Time Versus Time of Release is Evaluated

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Large Early Release Frequency

- Increase in Large Early Release Frequency (delta-LERF) is a separate metric for inspection findings, as in the MD 8.3 the criteria are one order of magnitude lower than the delta CDF.
- The SRA will perform a delta-LERF review, as required per NRC Inspection Manual Chapter 0609, Appendix A, Attachment 1, for all sequences (internal or external) that have a delta CDF of greater than or equal to 1E-7 per year
- An initial screening is performed using IMC -0609 Appendix H, this is dependent on the cores damage sequence and the type of containment
- The PRA staff may be contacted to review the sequences and provide information from the Level 2 PRA for Phase 3 evaluations.

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Licensee Input to Phase 2 Process

- Analysts May be Asked for:
  - Assessment of Assumption Validity
  - Comments on Phase 2 Applicability
  - Validation of Phase 2 Using Licensee's PRA
  - Input to External Events and/or LERF Assessments
- Licensee May Also be Asked for:
  - Design Documents Related to Deficiency
  - Procedures to Support Recovery Credit
- It is Always in the Licensee's Best Interest to Provide and/or Comment on Conclusions for Completed Phase 2.
- Greater than Green Phase 2 Estimations Usually Proceed to Phase 3.

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Phase 3 Evaluation At-Power Inspection Findings

- **Phase 3 is a Risk Significance Evaluation Using a Risk Basis That Departs from the Phase 2 Process**
  - In Phase 3, SRAs will Refine, Modify, or Supercede the Phase 2 Result.
  - In Addition, Phase 3 Addresses Findings that Cannot be Evaluated Using the Phase 2 Process.
  - While Performing a Phase 3 Evaluation, the SRAs will Use Appropriate PRA or Other Techniques.
  - Specialty Risk Analysts May be Consulted.

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Phase 3 Methods

- A Phase 3 Evaluation May Include the Following:
  - Portions of the Phase 2 Result
  - A Statement of the Influential Assumptions
  - A Discussion of the Tools Used for the Evaluation
  - The Affected Accident Sequences
  - A Sensitivity Study of the Results for each Major Assumption
- The Risk Tools Used May Include:
  - **Standardized Plant Analysis Risk Models**
  - Draft SDP Tools
  - Portions of the Licensee's PRA
  - Hand Calculations
  - Bounding Analyses

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Phase 3 Process (Cont'd)

- If green, the SRA will discuss the outcome with the PRA staff and prepare the analysis section of the Inspection Report
- If the initial Phase 3 work indicates a greater that green issue, the SRA will continue the dialog with the PRA staff on the influential assumptions and dominant results with the PRA staff. Further, the inspector and the SRA will start to prepare the SERP package.

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Licensee Input to Phase 3 Process

- Licensee is Encouraged to Provide a Complete Phase 3 Evaluation Including:
  - All Assumptions Made
  - The Revision of the PRA Model Used in the Analysis
  - Any Changes Made to the Model of Record
  - The Top Sequence and Event Cutsets
  - External Events Evaluated and Outcome
  - The Methods Used to Evaluate LERF
  - Documentation to Support Recovery and Human Reliability Analyses
- Routine Discussions Between the NRC and the Licensee are Encouraged Throughout the Process.

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SRA - Licensee PRA Staff Interactions

- The SRAs routinely speak with the inspectors about pending issues and how to proceed in the SDP.
- Likewise as issues come up the SRAs routinely call the licensee's PRA staffs and vice versa. We have very good working relationships will all the licensee PRA staff
- In Phase 2 the PRA staff will be contacted if:
  - There are plant assumption questions based on the Phase 2 Notebook
  - A quick look at Phase 2 indicates it may be of higher than very low risk significance.

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SRA - PRA Staff Interactions

- Additional PRA types of information may be requested or exchanged to allow more detailed understanding and modeling of the plant in a modified Phase 2 or in the Phase 3 SPAR model. This may include
  - Design documents related to deficiency
  - Procedures to support recovery credit
  - PRA modeling information

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SRA - PRA Staff Interactions

- The SRAs interact with site PRA staff to ensure a common understanding and comparison of assumptions and results, all the way through development of the Significance and Enforcement Review Panel (SERP) package and review of additional information provided following a greater than green preliminary finding and the issuance of the Final Risk Determination.

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Reaching a Preliminary Determination

- Differences Between SRAs and Licensee's Evaluation Must First Be Understood
- Differences are Quantified to Ensure Understanding
- Critical Differences are Assessed to Determine the Best Approach to Modeling and/or Best Assumption to be Used
- NRC SERP Members have Final Decision

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Reactor Oversight Process

Risk Significance Based on ΔCDF vs. ΔLERF

Frequency Range/RY	Significance Based on ΔCDF	Significance Based on ΔLERF
$\geq 10^{-4}$	Red	Red
$< 10^{-4} - 10^{-5}$	Yellow	Red
$< 10^{-5} - 10^{-6}$	White	Yellow
$< 10^{-6} - 10^{-7}$	Green	White
$< 10^{-7}$	Green	Green

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Significance Determination Process
and
Enforcement Review Panel

- Phase 3 Result is Provided to SERP as the Recommended NRC Preliminary Determination
- If Preliminary SERP Decision is Greater than Green:
  - Licensee is Sent a "Choice Letter."
  - Licensee Must Respond by Letter or Attend a Regulatory Conference
  - Licensee May Accept Preliminary Result
- If Preliminary Result is Changed:
  - SERP Reconvenes
  - SERP Evaluates New Information or Insights
  - SERP Makes Final Significance Determination of Finding
- Final Significance Letter is Issued

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Licensee Input to SERP Process

- Licensees Do Not have Direct Input to the SERP Process.
- Written Responses and/or Regulatory Conference Presentations Should Completely Explain Licensee Positions
- Licensee May be Asked to Provide Additional Information in a Short Period of Time
- Final Significance Determination is the Responsibility of the NRC
- Licensee May Decide to Appeal Final Determination upon Meeting Certain Criteria

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SDP References
Inspection Manual Chapters

- IMC 308, Attachment 3 and Associated Appendices A thru J, Significance Determination Process Basis Document
- IMC 609, Significance Determination Process
- IMC 60901, Significance and Enforcement Review Process
- IMC 60902, Process for Appealing NRC Characterization of Inspection Findings (SDP Appeal Process)
- IMC 60903, Senior Reactor Analyst Support Objectives
- IMC 609A, Determining the Significance of Reactor Inspection Findings for At-Power Situations
- IMC 609, Appendix B, Emergency Preparedness SDP
- IMC 609, Appendix C, Occupational Radiation Safety SDP
- IMC 609, Appendix D, Public Radiation Safety SDP
- IMC 609, Appendix E, Physical Security SDP (withheld from public)
- IMC 609, Appendix F, Fire Protection SDP
- IMC 609, Appendix G, Shutdown Operations SDP
- IMC 609, Appendix H, Containment Integrity SDP
- IMC 609, Appendix I, Operator Requalification Human Performance SDP
- IMC 609, Appendix J, Steam Generator Tube Integrity Findings Significance Determination Process
- Web address - http://www.nrc.gov/reading-rm/doc-collections/insp-manual/manual-chapter/index.html

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Notice of Enforcement Discretion

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Reactor Inspections
Notice of Enforcement Discretion

- Circumstances may occasionally arise where a power reactor licensee's compliance with a license condition is inappropriate with protecting the public health and safety.
- In these circumstances, the NRC staff may choose to not enforce the license condition. This enforcement discretion, designated as an NOED, is exercised only if the NRC staff is clearly satisfied that the action is consistent with public health and safety.

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Notice of Enforcement Discretion
Examples

- NOEDs may involve circumstances involving:
  - Plant Transients
  - Performance Testing
  - Inspection
  - System Realignments
  - Weather or other external factors

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Notice of Enforcement Discretion

The license amendment process is to be used in preference to NOEDs whenever possible.

Normally, the NRC staff considers NOED requests only if there is not enough time to process an emergency amendment request and the licensee can demonstrate that they contacted the staff immediately after identifying the problem.

Generally, an NOED request will not be considered if at least 72 hours of Completion Time remain for the affected license condition at the time the problem is identified. The staff can often disposition an emergency amendment request in less than 72 hours.

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Notice of Enforcement Discretion Risk Aspects

The current NOED policy and guidance require that licensees demonstrate to the staff's satisfaction, that a proposed NOED does not result in any net increase in radiological risk to the public.

A risk-informed basis demonstrates that continued operation is essentially within the plant's normal work control levels and, therefore, there is no net increase in radiological risk to the public at those levels. Normal work control levels, expressed in terms of incremental core damage probability and large early release probability, are specified in industry and NRC guidance on configuration risk management (e.g., Regulatory Guide 1.182).

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Notice of Enforcement Discretion Risk Aspects

Incremental Core Damage Probability (ICDP). The ICDP is the product of the incremental CDF and the annual fraction of the duration of the configuration [i.e., ICDP = ICDF x (duration in hours)÷(8760 hours per reactor year)].

Incremental Large Early Release Probability (ILERP). The ILERP is the product of the incremental large early release frequency (ILERF) and the annual fraction of the duration of the configuration. The ILERP=(ILERF x duration in hours)÷(8760 hours per reactor-year).

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Notice of Enforcement Discretion Risk Aspects

$$ICDP = \frac{ICDF \times Duration}{8760\ hrs/Rx\text{-}year}$$

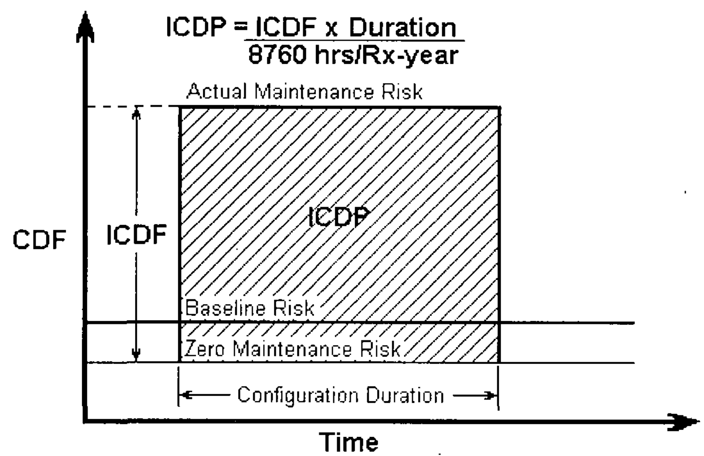

Figure 1 - Relationship of ICDF to ICDP

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Notice of Enforcement Discretion Risk Aspects

Use the zero maintenance PRA model to establish the plant's baseline risk and the estimated risk increase associated with the period of enforcement discretion.

For the plant-specific configuration the plant intends to operate in during the period of enforcement discretion, the incremental core damage probability (ICDP) and incremental large early release probability (ILERP) should be quantified and compared with guidance thresholds of less than or equal to an ICDP of 5E-7 and an ILERP of 5E-8.

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Notice of Enforcement Discretion Risk Aspects

Discuss the dominant risk contributors (cut sets/sequences) and summarize the risk insights for the plant-specific configuration the plant intends to operate in during the period of enforcement discretion.

This discussion should focus primarily on risk contributors that have changed (increased or decreased) from the baseline model as a result of the degraded condition and resultant compensatory measures.

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Notice of Enforcement Discretion Risk Aspects

Explain compensatory measures that will be taken to reduce the risk associated with the specified configuration. Compensatory measures to reduce plant vulnerabilities should focus on both event mitigation and initiating event likelihood. The objectives are to:

- reduce the likelihood of initiating events;
- reduce the likelihood of unavailability of trains redundant to the equipment that is out-of-service during the period of enforcement discretion; and
- increase the likelihood of successful operator recovery actions in response to initiating events.

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Notice of Enforcement Discretion Risk Aspects

Discuss how the proposed compensatory measures are accounted for in the PRA.

These modeled compensatory measures should be correlated, as applicable, to the dominant PRA sequences.

Other measures not directly related to the equipment out-of-service may also be implemented to reduce overall plant risk and, as such, should be explained.

Compensatory measures that cannot be modeled in the PRA should be assessed qualitatively.

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Notice of Enforcement Discretion Risk Aspects

Discuss the extent of condition of the failed or unavailable component(s) to other trains/divisions of equipment and what adjustments, if any, to the related PRA common cause factors have been made to account for potential increases in their failure probabilities.

Discuss external event risks for the specified plant configuration.

Discuss forecasted weather conditions for the NOED period and any plant vulnerabilities related to weather conditions.

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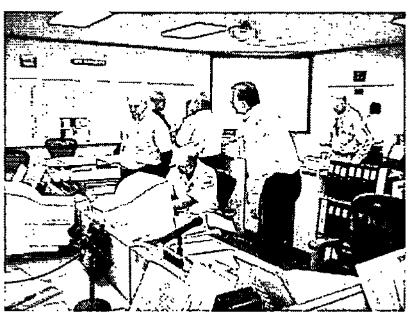
Incident Response



- NRC's Response to an Ongoing Event
- Activated in Region at ALERT or Higher
- NRC to Assist Licensee in Protecting Public
- Risk Information used to support Reactor Safety Team
- Risk Analysts would be Counterparts on Reactor Safety Team Bridge

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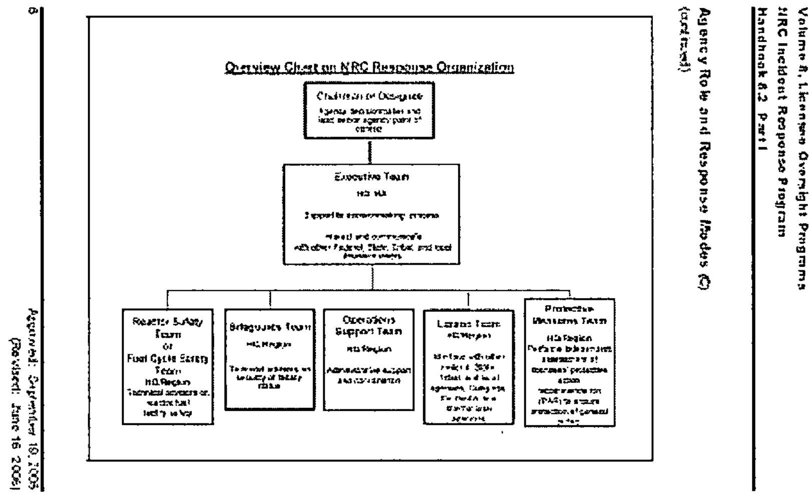

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NRC Response Modes

Response Mode	Team Leading the Response	Lead NRC Position	Description of Mode
NORMAL	N/A	N/A	Routine state of operations includes all activities for incident response readiness (including HQ Ops Officers, Regional Duty Officers, & Resident Inspectors)
MONITORING	Regional Base Team	Base Team Manager	Events are well understood and do not present an imminent danger to the health and safety of the public
ACTIVATION	HQ Executive Team	Executive Team Director	Incident sufficiently complex or uncertain: (1) Warrants extensive analysis and evaluation by the agency; (2) Warrants consideration for sending NRC site team; &/or (3) Involves terrorist activities
EXPANDED ACTIVATION	Site Team (if dispatched) or HQ Executive Team	Site Team Director (if dispatched) or ET Director	Incident severity and/or uncertainty warrants full agency response capabilities to support overall Federal response

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Incident Investigation

- Obtain Factual Information about Events and Conditions
- Decision to Initiate is Risk-Informed
- Risk Guidelines in Management Directive 8.3
- NRC Decides How to use Discretionary Inspection Resources
- Licensee MAY be asked for Assistance and/or Risk Characterization if Time Permits

FIRSTENERGY CORP.

Rust and dried boric acid are evident in this photo taken in April 2000 during an inspection of the Davis-Besse nuclear reactor lid. The company did not provide the photo to the Nuclear Regulatory Commission last fall, as it attempted to convince the agency that nozzles on the lid weren't leaking.

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Management Directive 8.3 - NRC Incident Investigation Program

- **Quickly** gather and analyze factual information about events and degraded conditions to determine if NRC needs to apply reactive inspection effort
- Inspection Staff gathers the information and passes it to their Branch Chief and the SRA
- The Branch Chief makes the determination if a deterministic criteria has been met
- Discretionary level of response based on both deterministic and risk criteria
- The SRA will contact the Licensee PRA staff to discuss the risk characterization assumptions and outcome. If time permits this will happen prior to an the NRC MD 8.3 decision.

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MD 8.3 Deterministic Criteria

- Operations outside design basis
- Major deficiency in design, construction, or operation
- Significant loss of integrity of fuel, primary boundary, or containment
- Loss of safety function or multiple failures
- Possible adverse generic implications
- Significant unexpected system interactions
- Repetitive failures or events
- Questions or concerns pertaining to licensee operational performance

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MD 8.3 - Risk Criteria

CCDP <1E-6	1E-6 - 1E-5	1E-5 - 1E-4	1E-4 - 1E-3	CCDP >1E-3
No additional inspection				
	Special Inspection			
		AIT		
				IIT

CLERP <1E-7	1E-7 - 1E-6	1E-6 - 1E-5	1E-5 - 1E-4	CLERP >1E-4
No additional inspection				
	Special Inspection			
		AIT		
				IIT

- Overlap areas allows some NRC management discretion.
- Risk Assessment includes the actual conditions that occurred. Depending on the event it will included known equipment problems and human performance issues.
- The plant Standardized Plant Analysis Risk (SPAR) model is used for the calculation.
- Risk Metrics
  - **For events - conditional core damage probability (CCDP) or conditional large early release probability (CLERP)**, given that the event has happened, this includes any associated equipment or human problems that actually happened
  - **For a degraded equipment condition – increase in core damage probability (delta-CDP) or increase in large early release probability (delta-LERP)**, given the equipment condition over an assumed exposure time. This is based on the actual plant configuration from the zero-test and maintenance baseline
  - **CLERP and delta-LERP bands are one order of magnitude lower than CCDP and delta-CDP**
- SI/AIT/IIT
  - Headquarters involvement in AIT and IIT decisions.
  - Inspection Charter will outline the areas of concern and the associated risk
  - Press release, unless security related
  - Inspection Report will discuss the MD8.3 risk assessment and a a final risk assessment given the information developed during the inspection
  - Findings are discrete issues and are evaluated based on the SDP
  - RES may conduct an ASP review and ask the licensee and SRA to provide comments.

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Inspection Planning

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Available Planning Tools

Risk-Based

- Plant Specific, Risk-Informed Inspection Notebooks
- Standardized Plant Analysis Risk (SPAR)
- Licensee Probabilistic Risk Analysis (PRA)

Traditional Defense-in-Depth

- Final Safety Analysis Report
- Equipment Performance History (CAP)
- Design Modifications/Changes
- Surveillance Testing
- Operating Experience

- Some of this risk material may be considered “SUNSI” (Sensitive Unclassified Non-Safeguards Information)

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Plant Specific, Risk-Informed Inspection Notebooks

- Readily available, desk-top reference for inspectors in the field.
- Standardized format, with good benchmarking by Brookhaven National Lab (BNL) to address unique plant design attributes and plant specific PRA applications.

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Notebook Information/Insights

- SDP Phase 2 tool
- Table 1 – Initiating Events of signficance
- Table 2 – Initiators and System Dependency
  - Risk significant systems
  - Major components and support systems
  - Unique plant characteristics
- Table 3 - Initiating Events and available mitigation systems/actions
- Event Tree Diagrams
- Table 4 – Quick reference (benchmarked events and PRA comparison)

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Standardized Plant Analysis Risk (SPAR)

- Standardized, partial-scope models
- Use event trees linked to component-level fault tree logic (fault tree linking method) to produce accident sequence cut sets
- Produced by Idaho National Lab (INL) for NRC use
- Ensures best available data application (up-to-date industry event and component reliability data) and standardized, repeatable risk assessments by NRC risk analysts.

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Licensee PRA

- Detailed, computer models with complete PRA or IPE data bases, use either large event tree approach or fault tree linking method.
- Although each licensee has an Individual Plant Examination (IPE) and Individual Plant Examination of External Events (IPEEE), only a few have full-scoped (internal and external events) computerized risk models
- Vary in format and quality from plant to plant

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Importance Measures

- A value derived from a Probabilistic Risk Assessment (PRA) model which quantifies the risk impact of a basic event relative to the total risk estimated by the model.
- SPAR models and licensee PRAs capable of generating Importance Measures (IMs).

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Importance Measures of Interest

- Risk Achievement Worth (RAW)
  - RAW = F(1) / F(X)
  - Measures the amount by which risk is increased if failure of Event X is certain.
- Risk Reduction Worth (RRW)
  - RRW = F(X) / F(0)
  - Measures the amount by which risk is reduced if failure of Event X can be eliminated.

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Component Design Basis Inspections

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Overview of CDBI

- Component, not system based
- Sample based on risk and margin, not just safety significance
- Operating Experience
- Emphasis on component/operations interface
  - Do operating procedures, training, etc. match actions credited in design or licensing bases?
  - Are operators trained, knowledgeable, capable of performing the risk-significant activities?

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Overview of CDBI

- NRC team
  - Lead inspector
  - SRA (as needed)
  - Operations inspector
  - 2 engineering inspectors (Mechanical/Electrical)
  - 2 contractors (Mechanical/Electrical Design)
  - Observer

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Overview of CDBI

- Request for Information Letter
- In office prep - Team Leader and SRA
- 1st: Onsite preparation – team plus SRA
- 2nd: In office (prep & non-inspection activities)
- 3rd: Onsite preparation and inspection
- 4th: In office (inspection & other activities)
- 5th: Onsite inspection
- 6th: In office (inspection & other activities)
- 7th: Onsite inspection and exit meeting

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Overview of CDBI

- Coordination
  - PRA: interface with SRA during preparation and for potentially risk significant findings
  - Operations: walking down procedures and mock performance of JPMs, may involve simulator time
  - System/Component engineering: address component-specific questions, system walk downs, operating experience, system/component health
  - Design engineering: address questions related to calculations, processes, etc.
  - Regulatory Assurance: Logistics, meeting schedule, licensing information, etc.

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CDBI
Types of Findings

- Vortexing
- Impact from Tornados
- TMI modification – ability to close RCIC suppression pool suction valves on containment isolation
- Diesel generator frequency
- Operator Actions
- Corrective Actions

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Fire Protection Inspections and Issues

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Topics

- NFPA 805 initiative
- Operator Manual Actions
- Post-Fire Safe Shutdown Circuit Analysis
- Fire Protection Inspection Issues

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Risk-Informed and Performance Based NFPA 805

- NFPA 805 is voluntary alternative to the fire protection rule, while maintaining safety and reducing unnecessary regulatory burden
- 11 nuclear licensees sent NRC letters of intent to transition 40 nuclear plants to NFPA 805
- Region III sites:
  - FENOC – Perry, Davis Besse
  - NMC – Point Beach, Palisades, Prairie Island, Monticello
  - FPL – Duane Arnold
  - AEP – D.C. Cook

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Risk-Informed and Performance Based NFPA 805 - continued

- NFPA Transition requires of use of a Fire PRA and Fire modeling tools to evaluate plant ability to achieve safe shutdown conditions following a fire
- NFPA is alternative approach to deterministic compliance with Appendix R
- NRC providing Enforcement Discretion for existing non-willful, non-Red risk significance findings

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Operator Manual Actions

- Issue:
  - NRC reviewed and approved operator manual actions as acceptable methods to safely shutdown plant
  - Some licensees rely upon manual actions which were not reviewed and approved by NRC
- RIS 2006-10, Regulatory Expectations With Appendix R Paragraph III.G.2 Operator Manual Actions, issued in June 2006.
- RIS discusses pending issuance of Enforcement Guidance and need for compensatory actions, where unapproved manual actions are relied upon

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Post-Fire Safe Shutdown Circuit Analysis

- Issue:
  - Potential fire-induced electrical circuit failures could prevent operation of equipment necessary to achieve and maintain safe shutdown
- RIS 2005-030, "Clarification of Post-Fire Safe-Shutdown Circuit Regulatory Requirements" provides clarification of regulatory requirements related to post-fire safe-shutdown circuit analysis
- Generic Letter is expected to be issued in 2006 regarding circuit analysis and spurious actuations.

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Fire Protection Inspection Issues

- Region III continues to focus on Appendix R circuits and operator manual actions
- Current RIS documents and pending Generic Letter regarding circuits laying regulatory groundwork for issue resolution
- Operator manual actions must be feasible and reliable
- Plants in NFPA 805 transition are subject to Enforcement Discretion
- Risk evaluations of inspection and licensee-identified findings can be complex

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Other Baseline Inspections

- **PI&R** – focus is on selection of corrective action issues involving systems with some risk significance
- **Heat Sink** – focus on more risk significant heat exchangers.
- **Modifications/50.59** – focus on the more risk significant systems, where practical.
- **Maintenance** – focus on the 50.65 (a)(1) systems

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End

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