"Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications"

By:Duncan Brewer
PRA Section Manager
Duke Power Company

Why Develop a Standard?

- ASME has undertaken development of a consensus standard for commercial nuclear power plant PRAs
- Several, complementary motivations for developing this standard:
  - Industry consensus standard requested by NRC
  - ASME interested in standard to support recent risk-informed code cases (e.g., ISI, IST)
  - Industry (via NEI) interested in standard to complement PRA Peer Review Certification initiative

ENCLOSURE 3

NRC's Objective of Standard

"...have a PRA standard such that the level of confidence in the technical quality of the PRA would be sufficiently adequate to support the identified applications and such that only an audit or inspection of the PRA by the United States Nuclear Regulatory Commission (NRC) would be needed to ensure its quality to support those applications."

ASME Organization

- ASME Committee on Nuclear Risk Management
  - Chairman is Sid Bernsen
- Project Team
  - Chairman is Ron Simard, NEI

MAIN COMMITTEE

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ASME Organization

- Initial charter for CNRM:
  - To develop, revise, and maintain standards and guides on risk management techniques supporting PRA and performance-based applications within ASME nuclear codes and standards.
- Charter for the Project Team:
  - Prepare a standard that sets forth the criteria and methodology necessary to ensure an acceptable level of quality and confidence when a specific risk assessment methodology is applied to a commercial nuclear power plant.

Purpose of Comment Period

- Initial draft (Revision 10) was available for public review & comment February 1999
- Public Workshop held March 1999
- Comments were due May 1999
- Enabled Project Team to get broader input
- Comments required major revision effort
- Draft 12 is currently available for public review and comment
- Public workshop is scheduled June 27

Comments

- Major comment on Draft 10 was that there needed to be a closer tie to the industry's Peer Review and Certification Process
- Also, many were confused by the use of *Shall, Should and May* language

Outline

1 Introduction

2 Definitions

3 Risk Assessment Application Process

4 Risk Assessment Technical Requirements

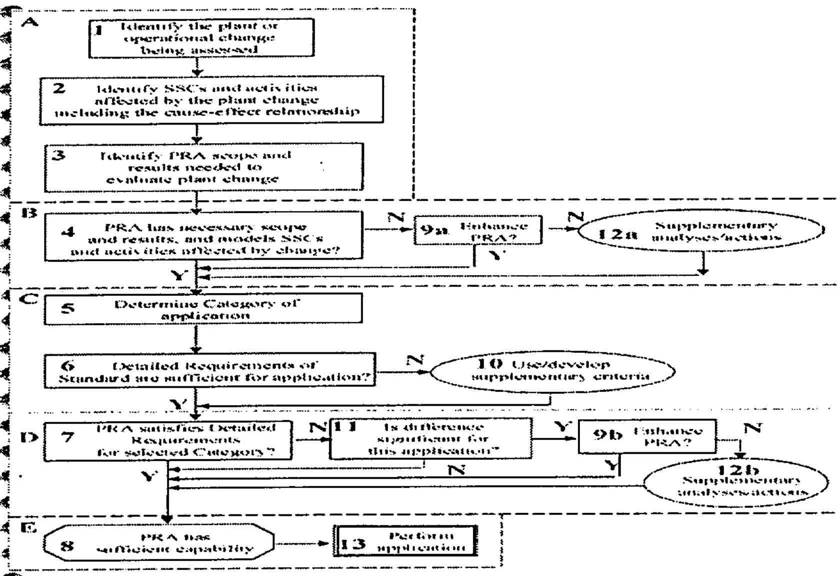
5 PRA Configuration Control

6 Peer Review

Section 3 Risk Assessment Application Process

- Identify and Define the Application
- Assess PRA Scope, Results and Models
- Determine Application Category/Level of Detail
- Compare PRA Model to the Standard

Section 3 Application Process



Section 4 Technical Requirements

- Initiating Events Analysis
- Accident Sequence Analysis
- Success Criteria
- Systems Analysis
- Human Reliability Analysis
- Data Analysis
- Internal Flooding
- Quantification
- Level 2 /LERF Analysis

Section 4 Technical Requirements

- High Level Requirements for Each Element
- Supporting Requirements for Each High Level Requirement
- Supporting Requirements Defined for Three Categories of Applications

Example High Level Requirement

- Dependencies: the systems analysis shall provide a reasonably complete treatment of common cause failures and intersystem and intra-system dependencies (HLR-SY-C)

Supporting Requirement Categories

- Category I Applications - Related to Dominant Accident Sequences
- Category II Applications - Related to Risk Significant Sequences
- Category III Applications - Related to All Modeled Sequences

Example Supporting Requirements

- Category I, II and III -
  - INCLUDE common-cause failures for identical components that provide redundancy. An acceptable method is represented in NUREG/CR-5485

Example Supporting Requirements

- Category I - For truncating cutsets, USE screening values <1E-4*base CDF and LERF
- Category II - For truncating cutsets, USE screening values <1E-5*base CDF and LERF
- Category III - For truncating cutsets, USE screening values <1E-6*base CDF and LERF

Section 5 - PRA Configuration Control

- Monitoring PRA Inputs and Collecting New Information
- PRA Maintenance and Upgrades
- Pending Changes
- Previous PRA Applications
- Use of Computer Codes
- Documentation

Section 6 - Peer Review

- At Least One Peer Review is Required
- Requires a Written Methodology
  (NEI-00-02, PRA Peer Review Process Guidance is an acceptable method)
- Minimum Qualifications are Specified
- PRA Elements Reviewed Against Section 4 Requirements
- Peer Review Should Include Use of Expert Judgement, Configuration Control and Documentation

Schedule

- Draft 12 for Public Review - 6/10/00
- DC Area Workshop - 6/27/00
- ACRS PRA Sub-committee - 6/28/00
- Full ACRS - July
- End of Comment Period - 8/10/00
- Revision 13 for ASME Committee Approval - September, 2000
- Pre-publication Copy Available for Use - 12/31/00

PSA Quality and Applications

Region II Risk Analyst information Meeting

June 20, 2000

Ching Guey, FPL (561-694-3137; ching_guey@fpl.com)

PSA Quality

- Programmatic Requirements vs Pragmatic Constraints

e.g., Self Assessment/Peer Review,

- Application-Orientation; Additional Layer vs Alternative
- Actionable vs Esoteric (HRA, CCF...)

Ongoing PSA Activities/Applications

- **Safety Significance Determination: SDP, Enforcement Conference support**
- **MR, CRMP**
- **PLA e.g., AOT Extension**
- **License Renewal**
- **PSA Maintenance & Update**
- **Procedure & Control**

Unrealistic Expectations & PSA Limitations

- Unwarranted QA requirements
- Limitations of Scope and Level of Detail
- Limitations of Knowledge or Information
- Unintentional Ratchetting from proactive users

Recommended Actions

- Comply with regulatory Requirements
- Ensure PSA quality
- Promote PSA process; impossible to have an all-comprehensive model
- Learn through applications

Promoting the PSA use

- Value-Added Activities
  - **PSA update/maintenance**
  - **Sharing the insights**
  - **Enhancing the realism of PSA model**
- Activities for further considerations
  - **More colors for At-Power OLRM????** (e.g. MR a(4))
  - **Managing plant performance by risk #s????**

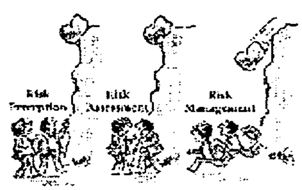
APPROPRIATE USE OF PRA/PSA???

Challenges

- Cultivate Realistic Expectations
- Obtain a balance between short-term and long-term needs
- "Earn" and Keep the benefits of PSA uses
- Develop and Deploy with tight resources

Conclusions

- Increasing Demands and Expectations
- Decreasing Resources and Time
- Simple, but not simpler
- Add value
- Meet plant needs



Questions to consider

- PSA model requirements for MR a(4)
- Is PSA model mature enough to:
  - have colors for RED, ORANGE, YELLOW
- How/When is PSA application initiated?
- How are PSA insights communicated?
- PSA resources full-time equivalent &$$$

MAINTENANCE RULE 10 CFR 50.65 (a)(4) INSPECTIONS

Peter Wilson, US NRC

Region II Risk Analysts Meeting

June 20, 2000

BACKGROUND

- Revised rule issued July 18, 1999
- Rule becomes effective on November 28, 2000
- Adds initial statement:

  "The requirements of this section are applicable during all conditions of plant operations, including normal shutdown operations."

BACKGROUND CONTINUED

- Adds new paragraph:

  "(a)(4) Before performing maintenance activities (including but not limited to surveillance, post-maintenance testing, and corrective and preventive maintenance), the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities. The scope of the assessment may be limited to structures, systems, and components that a risk-informed evaluation process has shown to be significant to public health and safety."

BACKGROUND CONTINUED

- RG 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants," dated May 2000, issued as guidance.
- RG 1.182 endorses revised NUMARC 93-01 Section 11, "Assessment of Risk Resulting from Performance of Maintenance Activities," dated February 22, 2000.
- NRC will be using NUMARC 93-01 Section 11 as basis for inspection procedures.

(a)(4) INSPECTION PLANS

- No baseline (programmatic) inspections are planned
- Routine Inspections
  - Part of the new revised reactor oversight process
  - To be performed by resident staff
- Supplemental Inspections
  - To be performed after risk-significant problems have been identified with a licensee's implementation of (a)(4).

ROUTINE (a)(4) INSPECTION PLANS

- Routine Inspections will be focused on:
  - Assessments
  - Management of risk when thresholds are exceeded
- Routine inspections will be performed during all modes of operation.

ROUTINE (a)(4) INSPECTION PLANS

- Assessments
  - Inspectors will verify that an assessment was performed when appropriate.
  - Inspectors will check if the assessment considered all SSCs within scope of the the rule
  - The inspectors will determine if the assessment considered:
    - External events
    - Internal flood hazards
    - Significant potential for trips/scrams
    - Significant complications to recovery efforts
    - Containment integrity

ROUTINE (a)(4) INSPECTION PLANS

- Assessments Continued
  - The inspectors will check if the capabilities of the risk assessment tools were exceeded and whether additional qualitative judgments were made due to tool limitations.
    - Qualitative assessment
    - Matrix
    - "cutset editors"
    - risk monitors

ROUTINE (a)(4) INSPECTION PLANS

- Scope of assessments:
  - May include full, original MRule scope
  - May be limited, at licensee's option, to SSCs:
    - included in the scope of the plant's level one, internal events PSA, plus
    - determined to be high safety-significant through the MRule risk-ranking process
    - Must also include other deviations from designed plant configuration, including:
      - temporary changes to support the maintenance activity

ROUTINE (a)(4) INSPECTION PLANS

- Management of risk
  - Inspectors will verify that licensee risk management actions have been implemented when thresholds have been exceeded.
    - Actions to provide increased risk awareness
    - Actions to reduce duration of maintenance activity
    - Implementation of contingency plans
    - Actions to minimize magnitude of risk increase

ROUTINE (a)(4) INSPECTION PLANS

Thresholds:

Configuration CDF < 10^{-3}

ICDP	[illegible]	ILERP
> 10^{-5}	-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}

SUPPLUMENTAL (a)(4) INSPECTION PLANS

- Will be triggered if risk significant problems are identified
  - Will be an extensive process review
  - Will focus on area(s) of concern
  - May look at PSA quality issues
  - Will look at extent of condition

INSPECTION GUIDANCE DELOPMENT PLAN

- (a)(4) Roll-out Plan Products:
  - Updated IP 71111.12
  - Updated IP 71111.13
  - New supplementary procedure for biennial regional review of site's (a)(4) activities
    - New (a)(4) SDP
    - Updated EGM

INSPECTION GUIDANCE DELOPMENT PLAN

- (a)(4) Roll-out Plan continued:
  - Visit some licensees with different assessment tools for information
    - Develop IP revisions
    - Visit one site per region for IP V&V and upgrade
    - Visits will include staff of IQMB and DSSA; regional staff are invited, welcome, and encouraged to participate
  - Orient NRC staff:
    - Participate in resident counterpart meetings
    - Visit regions, as possible

Hatch Unit 1
Partial Loss of Main Feedwater Flow
Risk Significance Analysis

Anees Farruk
PRA Supervisor

Southern Nuclear Operating Company

SOUTHERN COMPANY
Energy to Serve Your World

Risk Analysis - Approach

- Perform Risk Significance Evaluation of the Initiator and Degraded Condition, and Compare Results to Various Numerical Criteria Published by the NRC Noted Below:
- The Following Criteria Published in Draft NRC Management Directive 8.3, "NRC Incident Investigation Procedure", Part I Was Used for Comparing Results of Risk Analysis:
  - *Reactor Events: Conditional Core Damage Probability (CCDP)*
  - *Degraded Conditions: Instantaneous Core Damage Frequency (ICDF)*
- NRC Significance Determination Process Matrix
- New NRC Oversight Process Performance Indicator for Mitigating Systems:
  - *Delta Core Damage Frequency (ΔCDF)*

Risk Analysis Event Scenario

- Event Scenario:
  - **Automatic Reactor Scram Due Low water Level**
  - **HPCI Available with Potentially Reduced Reliability**
    - » Potential for Reduced Reliability Attributed to Existence of Continued High Water Level Condition
  - **RCIC Available with Reduced Reliability**
    - » Reduced Reliability Attributed to High Water Level Trip malfunction
  - **SRVs Available**
  - **Availability of Condenser Following Opening of MSIVs**

Risk Analysis Assumptions

- **Failed Equipment: Failure Event Set as 'TRUE'**
- **Recovery of Failed or Secured Equipment: Used a Random Non-recovery Probability**
- **Successful Equipment: Used a Random Failure Probability**
- **Occurrence of Initiator: Initiator Set as 'TRUE' or Used an Average Annual Frequency Appropriate for the Risk Measure Calculated**
- **Average Equipment Degradation Duration:** 78/2 Days (11/10/1999 - 1/26/2000)
- **PRA Model: Used the Post-IPE Hatch U1 Average Core Damage Frequency Model**

Numerical Criteria: Equations

- Conditional Core Damage Probability:

  CCDP = Average Annual CDF From U1 $|_{\text{LOFW}=1}$
- Instantaneous Core Damage Frequency:

  ICDF = Average Annual CDF From All Initiators $|_{\text{DEGRADED CONDITION}}$
- Delta Core Damage Frequency:

$$\Delta CDF = \sum_{I=1}^{I=N} \Delta ICDF * (\text{Degraded Condition Duration})$$

Where: ΔICDF = Average Annual CDF From All Initiators$_I$ $|_{\text{DEGRADED CONDITION}}$

- Average Annual CDF From All Initiators $|_{\text{BASE CASE}}$

N = Number of Discrete (Non-Overlapping) Degradation Condition Periods

PRA Results - Dominant Core Damage Sequence

- Loss of Partial Feedwater Initiating Event
- Loss of Power Conversion System (MSIV Closure)
- Loss of High Pressure Coolant Injection (Both HPCI and RCIC Degraded by Vessel Overfill & Not Recovered)
- Loss of Primary System Depressurization for Condenser Vessel Injection or Low Pressure Injection

PRA Results - CCDP

- This Risk Measure Provides an Estimate of Risk Significance of the Loss of Defense-in-Depth Caused Subsequent to the Occurrence an Initiating Event
- Conditional Core Damage Probability Assumes Occurrence of LOFW Initiating Event and Initial Unavailability of Failed Equipment
- CCDP Value for the Scenario Was Calculated As *8.2E-07*
- As Shown in Figure 1, the Hatch U1 LOFW Event Is Classified As *a Non-risk Significant Event*

PRA Results - CCDP

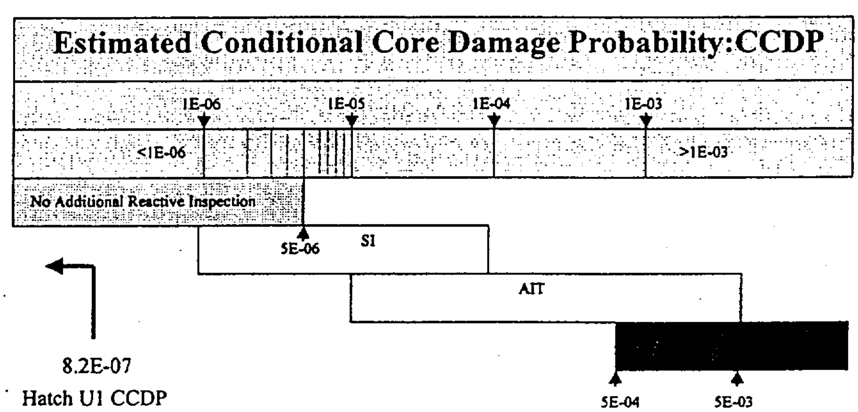


Figure 1

PRA Results - ICDF

- *Instantaneous Core Damage Frequency (ICDF)* Measure Provides an Estimate of the Core Damage Risk Assuming All Initiating Events Are Likely to Occur at a Random Frequency and the Failed Equipment Is Initially Unavailable when Demanded During an *Entire Year*
- A Value Greater than 1E-03/Year is considered *Risk Significant*
- ICDF Bounding Value Was Calculated As *7.26E-05/Year* (Base Case CDF = *1.65E-5/Year*)
- Based on these Results, the Hatch U2 LOFW Event Is Classified As *a Non-risk Significant Event*

PRA Results - SDP Evaluation

- This Risk Measure Provides a Estimate of the Incremental Risk Increase in Terms of Numerical Values Considered As Surrogate to ΔCDF Assuming All Initiating Events Are Likely to Occur at a Random Frequency and the Failed Equipment Is Initially Unavailable when Demanded During the Degradation Periods
- Revised Hatch SDP Sheets Reflecting Post-IPE Model Changes Were Used for the Risk Analysis
- Bounding SDP Sheet Evaluation: As Shown in Figure 3 the Hatch U1 LOFW Event + Degraded Condition Is Classified Under a Plant Performance Condition of *Yellow Region* Requiring Phase III PRA

PRA Results - SDP Evaluation

	Remaining Mitigating Capability Rating						
IE Likelihood	6	5	4	3	2	1	0
A			★				
B				★			
C							
D						Yellow	
E						White	
F							
G							
H							

Hatch U1 LOFW Performance Condition

Red

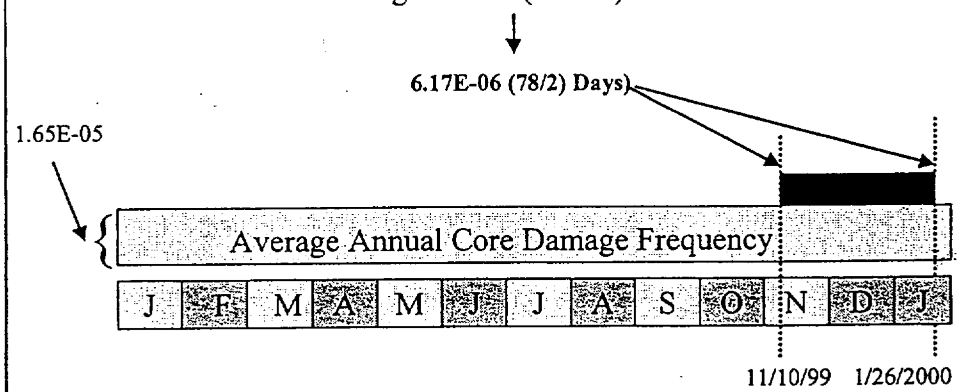
Figure 3

PRA Results - Delta CDF

- ***Delta Core Damage Frequency (ΔCDF)*** **Measure Provides an Estimate of the Incremental Core Damage Risk Increase Assuming All Initiating Events Are Likely to Occur at a Random Frequency and the Failed Equipment Is Initially Unavailable During the Degradation Periods**
- **ΔCDF Value Was Calculated As *6.17-06***
- **ΔLERF Value Was Calculated As *3.41E-07***
- **As Shown in Figure 4 the Hatch U1 LOFW Event Is Classified Under a Plant Performance Considered *Acceptable (White Region): Increased Regulatory Response Band***

PRA Results: ΔCDF

Total Incremental CDF
Attributable to Equipment
Degradation (ΔCDF)



PRA Results - ΔCDF

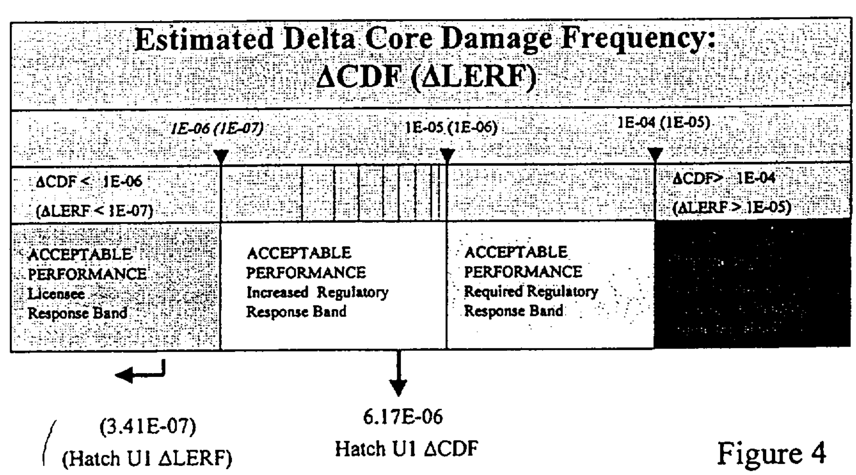


Figure 4

Risk Significance Analysis Conclusions

- *CCDP Risk Measure:* As Shown in Figure 1, the Hatch U1 LOFW Event Is Classified As a *Non-risk Significant Event*
- *ICDF Risk Measure:* The Hatch U1 LOFW Event Is Classified As *a Non-risk Significant Event*
- *Bounding SDP Sheet Evaluation:* As Shown in Figure 3 the Hatch U1 LOFW Event Is Classified Under a Plant Performance Condition of *Yellow Region* Requiring Phase III PRA
- *New PI Measure:* As Shown in Figure 4 the Hatch U1 LOFW Event Is Classified Under a Plant Performance Condition Considered *Acceptable (White Region): Increased Regulatory Response Band*

Significance Determination Process Update

Peter Wilson, US NRC
Region II Risk Analysts Meeting
June 21, 2000

Background

- The NRC developed the significance determination process (SDP) to provide a method of assigning a risk characterization to inspection findings.
  - align inspection finding with risk-informed performance indicators.
  - improve NRC objectivity.
  - more understandable and predictable process.
  - provide increased focus on aspects of performance that have the greatest impact on safe plant operation.
  - Improve communication.

Background Continued

- The NRC has developed or is in the process of developing SDPs for each of the 7 cornerstones of reactor safety
- This discussion will focus on the following cornerstones:
  - Initiating Events
  - Mitigating Systems
  - Barrier Integrity

Background Continued

- There are three phases in the SDP for these cornerstones
  - **Phase 1** - Definition and Initial Screening of Findings
    - Precise characterization of the finding and an initial screening-out of low-significance findings
  - **Phase 2** - Risk Significance Approximation and Basis
    - Initial approximation of the risk significance of the finding and development of the basis for this determination for those findings that pass through the Phase 1 screening
  - **Phase 3** - Risk Significance Finalization and Justification
    - As-needed refinement of the risk significance of Phase 2 findings by an NRC risk analyst

Background Continued

- The NRC identified the need to develop several phase 1and/or 2s:
  - Significance Determination of Reactor Inspection Finding for At-Power Situations
  - Fire Protection and Post-Fire Safe Shutdown SDP
  - Shutdown Safety SDP
  - Containment Integrity SDP

At Power SDP

- Phase 2 worksheets first draft sent to each site.
  - Developed from IPEs by BNL.
  - Early recognition that the information was dated.
- Benchmarking of pilot plant worksheets identified several issues.
  - Lack of special initiators.
  - Initiating event frequency differences.
  - Human error probability differences.

At Power SDP

- NRC analysts have visited most sites to:
  - update the phase 2 worksheets to reflect current plant design and operation,
  - Obtain information on special initiators
  - Obtain information to allow for future NRC benchmarking
  - Remaining sites to be visited
    - Crystal River 3
    - St Lucie
    - Turkey Point
    - North Anna
    - Surry
    - DC Cook

At Power SDP

- The current phase 2 SDP does not currently consider external events and internal floods.
  - BNL is conducting a review to determine which plants should have SDPs to address external initiators.
  - NRC decision add additional work sheets for external events and floods will follow receipt of BNL's report.
  - Expecting BNL report in the fall 2000.

At Power SDP

- The NRC intends to complete the revision 0 of the at power phase 2 worksheets in the fall 2000.
  - Current plans are to add the worksheets to the NRC inspection manual.
- As an interim measure, inspection finding.
- s that are not filter out by the phase 1 screen are being reviewed by NRC risk analysts and phase 3 evaluations are being performed when appropriate.

Other Reactor SDPs

- Fire Protection and Post-Fire Safe Shutdown SDP.
  - SDP is complete.
  - Minor changes in the treatment of control room fires are being considered for a future revision.

Other Reactor SDPs

- Shutdown Safety SDP
  - Phase 1 screening tool complete
  - Phase 2 tool is expected to be completed in fall 2000
  - Currently all shutdown finding are under going phase 3 analyses
- Containment Integrity SDP
  - Still under development
  - Currently all containment integrity finding are under going phase 3 analyses

Resources

- The NRC's SDPs are documented (minus the at-power phase 2 worksheets) in Inspection Manual Chapter 609
- Draft MC 609 is publicly available via the public document room or
- Via the internet at:
  - www.nrc.gov/NRR/OVERSIGHT/ROP/documents.htm l
- Expect several revisions over the next year

Conclusions

- The NRC's SDPs are works in progress
- The progress made to date would not have been possible without your input and cooperation