IAEA REGIONAL WORKSHOP
ON
DESIGN, EVALUATION, AND LICENSING
OF NPP MODIFICATIONS

Use of Risk Insights in Regional Operations

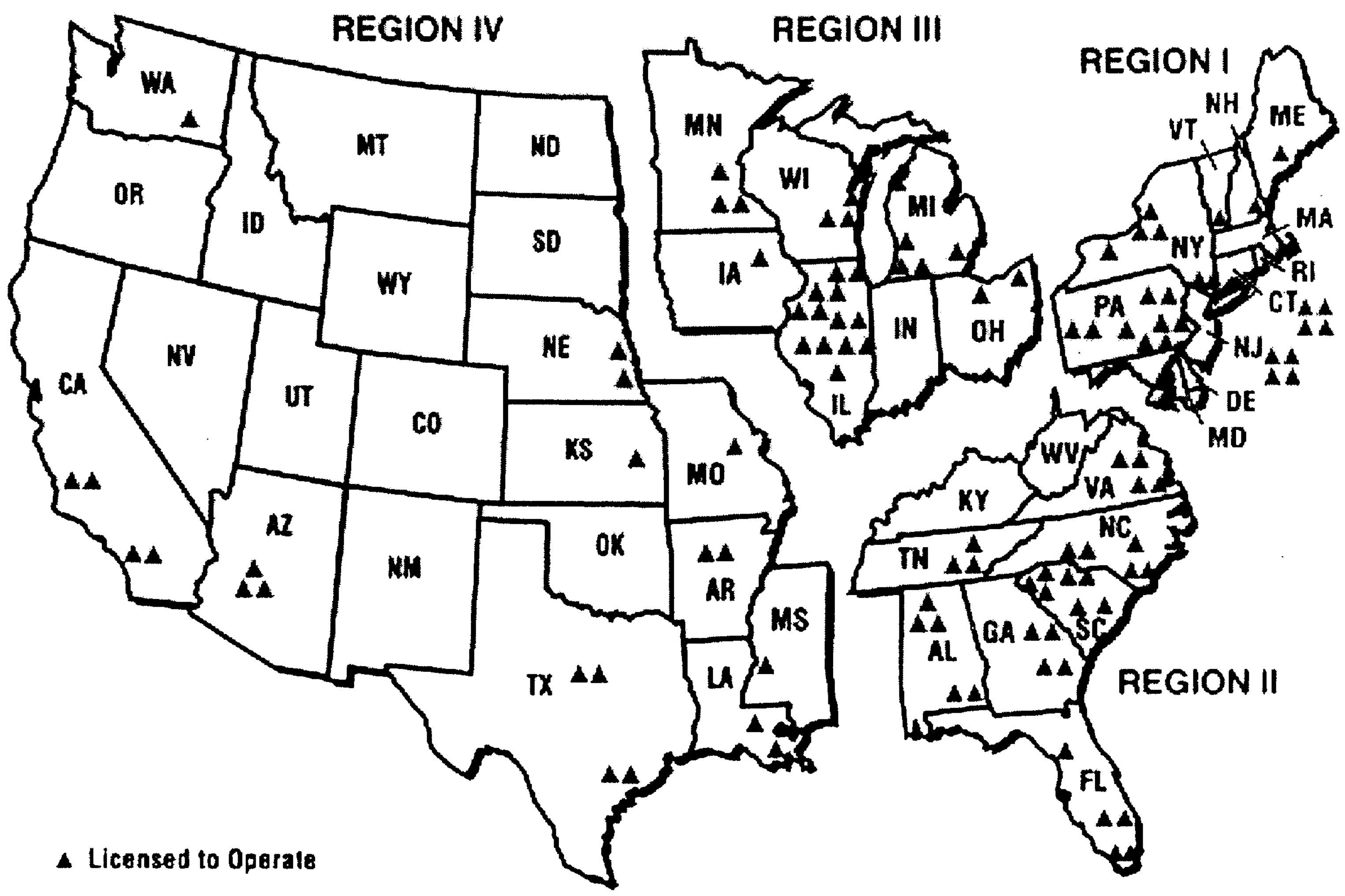
Jeffrey L. Shackelford

USNRC Regional Offices

- Four regional offices
  - Region I - King of Prussia, Pennsylvania (near Philadelphia)
    - Northeastern United States
  - Region II - Atlanta, Georgia
    - Southeastern United States
  - Region III - Lisle, Illinois (near Chicago)
    - Central/Midwestern United States
  - Region IV - Arlington, Texas (near Dallas-Ft. Worth)
    - Western United States

- The NRC's overall inspection and assessment program is developed by the Office of Nuclear Reactor Regulation (NRR); however, the program is implemented and managed at the regional office level.

United States Nuclear Regulatory Commission
Region Office Areas of Coverage



Note: There are no commercial reactors in Alaska or Hawaii.

Probabilistic Safety Assessment Assets in the Regional Offices

- Each regional office is staffed by two qualified senior reactor analysts (SRA)
  Two senior reactor analysts are assigned to the headquarters office

- The NRC inspector training and qualification program includes mandatory training and instruction in Probabilistic Risk Assessment (PRA) technology
  - PRA Technology and Regulatory Perspectives Course (2 weeks)

- NRC management training includes courses related to the uses, limitations, and interpretation of PSA results
  - PRA for Technical Managers Course

- Additional resources are available on a case basis from the Office of Nuclear Reactor Regulation and the Office of Nuclear Regulatory Research (NRC headquarters personnel)

Senior Reactor Analysts - Region IV
(SRAs)

- Senior qualified reactor inspector
  - Senior region-based inspector
  - Senior resident inspector

- Extensive formal training in PSA technology
  - System modeling
  - Reliability analysis
  - Statistical analysis
  - Human reliability analysis
  - Computer code usage

- On the job training assignments
  - Temporary assignments in NRC headquarters PSA organizations
  - Interface with National Laboratory organizations

- Formal Certification Process
  - Two year training and qualification program

Senior Reactor Analysts
(SRAs)

- Region based SRAs report to the Director, Division of Reactor Safety in each NRC Region

- Two headquarters SRAs report coordinate program activities within the Office of Nuclear Reactor Regulation

- SRA Counterpart Activities

  Biweekly counterpart teleconference to discuss recent issues of interest, plant events, inspection findings and other risk-related topics

  Biannual counterpart meeting to discuss program related issues and other important risk related topics

Use of PSA Technology in the NRC Regional Offices
Primary Program Areas

- Development and dissemination of risk related information
- Notice of enforcement discretion (NOED)
- Enforcement severity evaluations
- Inspection finding significance evaluations
- Event evaluations
- Outage risk reviews
- Inspection planning and prioritization
- Other activities

PSA Resources in the NRC Regional Offices

- Simplified Plant Analysis Risk (SPAR) models
  Simplified risk model of each US nuclear power plant

- Individual Plant Examination (IPE) information
  NRC Generic Letter 88-20 required each operating US nuclear power plant to submit a detailed examination of plant risk and potential vulnerabilities
  (http://www.nrc.gov/NRC/GENACT/GC/GL/1988/gl88020.txt)

- Updated plant risk information
  Most US nuclear power plants have performed updates to their original IPE submittals

  Region based collections or "libraries" of updated plant information

- Generic information
  NRC reliability studies for important systems and components
  Initiating event frequency studies
  Accident sequence precursor studies
  Other NRC PSA initiatives

NRC Region IV SRA Activities

Development and dissemination of risk related information

- Senior reactor analyst site interface visits
  - Site interface visits conducted at each facility
  - Professional/Technical contacts established
  - Enhanced working relationships with NRC resident inspection staff

- Probabilistic Safety Assessment Regulatory Users Group
  - Establishment of a standing working/users group whose function is to share and communicate risk-informed information
  - Enhanced regulatory interface

- Regional PSA library
  - Establishment of up-to-date risk information for each facility

NRC Region IV SRA Activities

Notice of Enforcement Discretion (NOED)

- NOED process addresses temporary nonconformance with license conditions and Technical Specification requirements

  Usually granted to allow additional time to make necessary repairs or to perform required testing

- NRC Manual Chapter 9900 provides technical guidance on the NOED process (http://www.nrc.gov/NRC/GENACT/GC/AL/1995/al95005r1/attachment.txt)

- NRC Manual Chapter 9900, B.1.0, General Considerations

  ..."If appropriate and feasible, the staff should perform a qualitative probabilistic risk analysis (PRA) as an input to its decision process..."

  Each request for a NOED must include a discussion of the safety basis of the request and should include at least a qualitative risk assessment derived from the facility's PRA.

- Each request for Notice of Enforcement Discretion for reactor operational issues is reviewed by a senior reactor analyst for risk considerations

NRC Region IV SRA Activities

Enforcement Severity and Inspection Finding Significance Evaluations

- Violations of regulatory requirements and individual inspection findings are reviewed from a risk perspective
  (http://www.nrc.gov/OE/rpr/enfman/BR0195/R2old/br0195r2.html#_1_72)

- In the revised oversight process, inspection findings will be categorized via the NRC's "significance determination process" (SDP)

- Risk significance is used as an input into the Agency's overall actions

- Findings will be characterized as either "GREEN", "WHITE", "YELLOW", or "RED"
  Depending on the characterization, the level of Agency interaction will increase accordingly

USE OF RISK INFORMATION IN NRC AND INDUSTRY PROGRAMS

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[3]Office Letter 803 Reference to 10/30/98 Guidance Memo

[4]Regulatory Analysis Guidelines NUREG/BR-0058 for CCFP.1 to 1

(Riskinfo.wpd - 7/28/99)

NRC Region IV SRA Activities

Event Evaluations

- The risk significance of events at nuclear reactors is used as an input into the Agency's decision regarding the appropriate level of response

- Conditional Core Damage Probability (CCDP) is used as a measure of the risk significance of the event

- The Agency's response level can range from: 1) Follow up by the onsite inspectors, 2) Special inspection by specialists from the regional office, 3) Augmented Inspection Team (AIT) led by the regional office and supplemented with other Agency experts, and 4) Incident Investigation Team (IIT) led by a senior Agency manager and staffed by inspectors from other regional offices

NRC Region IV SRA Activities

Outage Risk Reviews

- Refueling outages are reviewed from a risk perspective
  - Configuration risk (i.e., risk of individual outage configurations)
  - General modification risk (i.e., modifications or major maintenance on risk significant plant equipment)

- Risk informed input is provided to resident inspection staff for inspection planning and prioritization

- Outage Insights
  - In calendar year 1999, Nineteen refueling outages were performed in Region IV
  - Of the 16 PWR outages reviewed, all 16 employed reduced inventory or midloop configurations
  - The average time to boil during this configuration was about 15 minutes
  - The average time after shutdown before reaching midloop was about 94 hours

NRC Region IV SRA Activities

Inspection Planning and Prioritization

- Risk input is used to aid in planning and prioritizing major routine inspection activities such as engineering team inspections of selected safety systems

- Risk input is also used in investigating events and in other reactive inspection activities

- System and component selection (for routine inspections), important operator and recovery actions (for reactive inspections)

NRC Region IV SRA Activities

Other Activities

- Lead reactive team inspections
  - Special Inspections, Augmented Inspection Teams
- Professional conferences and programs
- PSA Technology Applications in other industries
  - National Aeronautics and Space Administration (NASA)
  - United States Banking Industry

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Overview of Risk-Informed Regulation at United States Nuclear Power Plants

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U.S. NRC Severe Accident Studies

WASH-740	Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants (1957)
WASH-1400	Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants (1975) ❖ Although WASH-1400 represented a significant advance in the use of PRA methods in the United States, the Commission did not regard the study's numerical estimate of the overall risk from this study as reliable.
NUREG-1150	Severe Accidents Risks: An Assessment for Five U.S. Nuclear Power Plants (1990) ❖ Reflected state of the art understanding of severe accident phenomenology and analysis methods

Examples of Past Staff Uses of Risk Information

- Regulatory Analysis, NUREG/BR-0058, Revision 2 (1995)
  - ❖ Guidelines utilized a subsidiary core damage frequency safety goal of 1E-4 per reactor year and a conditional containment failure probability of 0.1
- NUREG-0933 "A Prioritization of Generic Safety Issues", (1983)
- Station Blackout Rule, 10 CFR 50.63 (1988)
- Anticipated Transient Without Scram (ATWS) Rule, 10 CFR 50.62 (1984)

Commission Policy Statements on Severe Accident Risk and Probabilistic Risk Assessment

"Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants", August 1985 (50FR32138)

"Safety Goals for the Operation of Nuclear Power Plants; Policy Statement", August 1986 (51FR30028)

"Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities; Final Policy Statement", August 1995 (60FR42622)

NRC PRA POLICY STATEMENT

- The use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy.

- PRA and associated analyses (e.g., sensitivity studies, uncertainty analyses, and importance measures) should be used in regulatory matters, where practical within the bounds of the state-of-the-art, to reduce unnecessary conservatism associated with current regulatory requirements, regulatory guides, license commitments, and staff practices. Where appropriate, PRA should be used to support the proposal of additional regulatory requirements in accordance with 10 CFR 50.109 (Backfit Rule). Appropriate procedures for including PRA in the process for changing regulatory requirements should be developed and followed. It is, of course, understood that the intent of this policy is that existing rules and regulations shall be complied with unless these rules and regulations are revised.

NRC PRA POLICY STATEMENT (CON'T)

- PRA evaluations in support of regulatory decisions should be as realistic as practicable and appropriate supporting data should be publicly available for review.

- The Commission's safety goals for nuclear power plants and subsidiary numerical objectives are to be used with appropriate consideration of uncertainties in making regulatory judgments on the need for proposing and backfitting new generic requirements on nuclear power plant licensees.

RISK-INFORMED REGULATION

Insights and results derived from probabilistic risk assessments are used in combination with deterministic system and engineering analyses to focus licensee and regulatory attention on issues commensurate with their importance to safety.

Objectives for Risk-Informed Regulation

- Enhance safety decisions (e.g., configuration control, accident management)
- Efficient use of NRC resources (e.g., Individual Plant Examination (IPE) insights, risk-informed inspections)
- Reduce unnecessary licensee burden (e.g., graded Quality Assurance (QA), risk-informed InserviceTesting (IST)

The NRC Staff has been concentrating efforts in the areas of:

- **Risk-Informed Regulatory Guidance Development**
  (http://www.nrc.gov/NRC/RG/01/index.html)

- **Review of Risk-Informed Licensing Submittals**

- **Incorporating Risk Insights in Inspection Activities**
  (http://www.nrc.gov/NRC/IM/index.html)

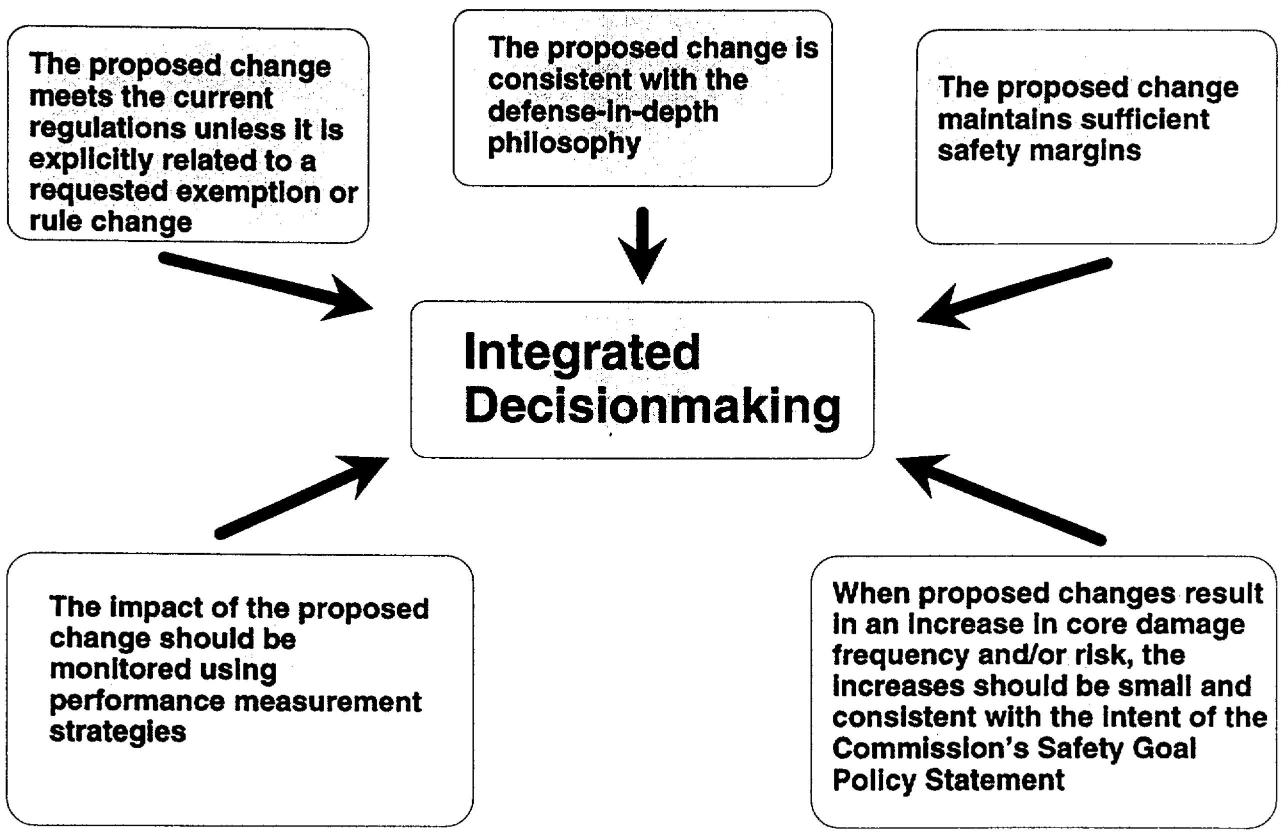
- **PRA Training**

- **Development of Risk Methodology for the Revised Reactor Oversight Program**
  (http://www.nrc.gov/NRR/OVERSIGHT/overview.html)

The NRC has finalized regulatory documents that will support risk-informed plant licensing changes:

Regulatory Guide (RG) 1.174 - General guidance to licensees	Standard Review Plan (SRP) Chapter 19, Revision P - General guidance to staff
RG-1.175 - Application specific guidance on inservice testing (IST)	SRP Section 3.9.7 - Application specific guidance on IST
RG-1.178 - Application specific guidance on inservice inspection (ISI)	SRP Section 3.9.8 - Application specific guidance on ISI
RG-1.176 - Application specific guidance on graded quality assurance (GQA)	GQA Inspection Guidance - under development
RG-1.177 - Application specific guidance on technical specifications (TS)	SRP Section 16.1 - Application specific guidance on TS

Principles of Risk-Informed Decisionmaking



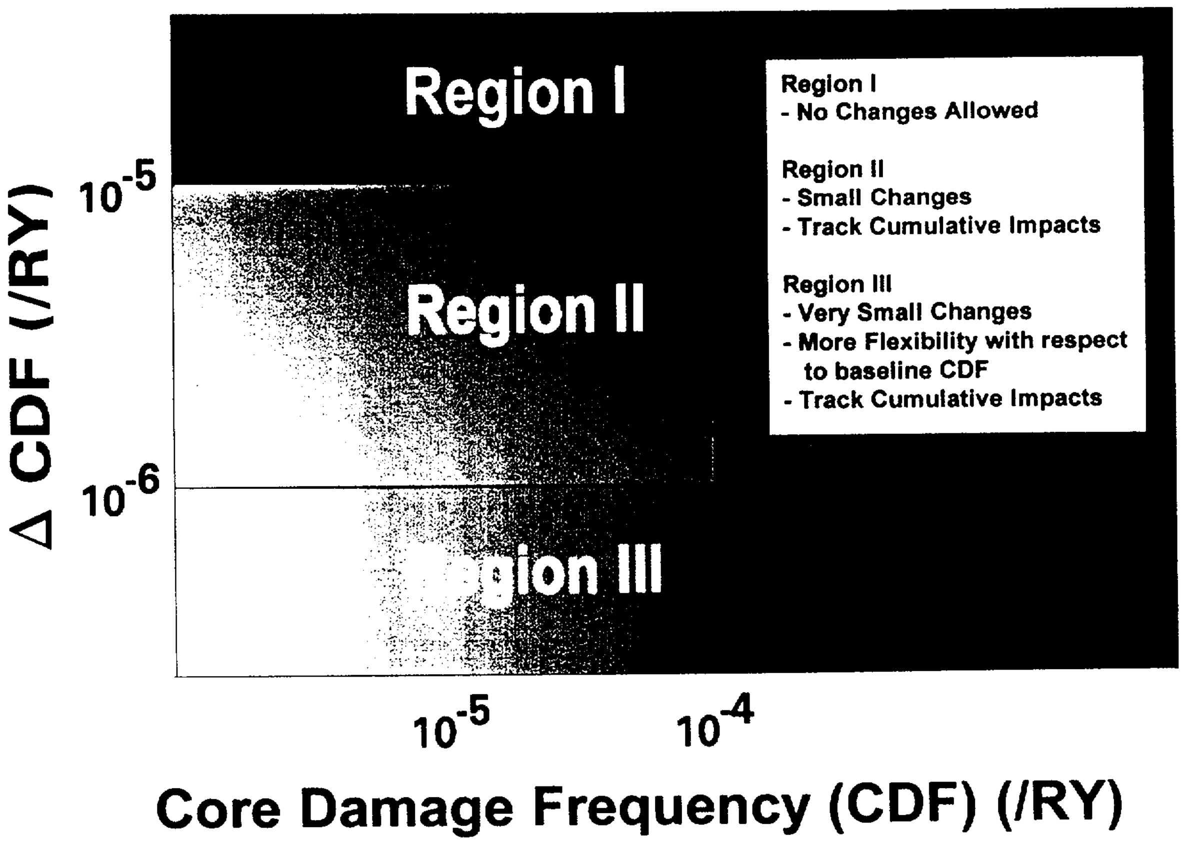
NRC Staff Expectations (Licensing Process)

- Proposed changes are evaluated in an integrated fashion that ensures that all principles are met.
- All safety impacts of the proposed change are evaluated in an integrated manner as part of an overall risk management approach in which the licensee is using risk analysis to improve operational and engineering decisions broadly by identifying and taking advantage of opportunities for reducing risk, and not just to eliminate requirements the licensee sees as undesirable.
- The use of Core Damage Frequency (CDF) and Large Early Release Frequency (LERF) as bases for probabilistic risk assessment guidelines is an acceptable approach to addressing risk impact and consistency with the intent of the Commission's Safety Goal Policy Statement.
- Increases in estimated CDF and LERF resulting from proposed licensing changes are limited to small increments, and the cumulative effect of such changes should be tracked and considered in the decision-making process.

NRC Staff Expectations (PRA Analysis)

- The scope and quality of the engineering analyses (including traditional and probabilistic analyses) conducted to justify the proposed licensing change should be appropriate for the nature and scope of the change, should be based on the as-built and as-operated and maintained plant, and should reflect operating experience at the plant.
- Appropriate consideration of uncertainty is given in analyses and interpretation of findings, including using a program of monitoring, feedback and corrective action to address significant uncertainties.
- The plant-specific PRA that is used to support licensee proposals has been subjected to quality controls such as an independent peer review or certification.
- Data, methods, and assessment criteria used to support regulatory decision-making are clearly documented and available for public review.

Acceptance Guidelines [1] for Risk-Informed Licensing Changes



[1] Similar Guidance exists for large early release frequency (LERF)

RISK-INFORMED PILOT APPLICATIONS

Graded Quality Assurance (GQA)

- **This program is structured to allow a “grading” of the quality assurance (QA) treatment of plant structures, systems, and components (SSCs) based upon their safety importance.**

- **The following essential elements to a graded QA Program have been identified:**

  - A process that identifies the appropriate safety significance of SSCs in a reasonable and consistent manner

  - The implementation of appropriate QA controls for SSCs, or groups of SSCs, based on safety function and safety significance

  - An effective monitoring, root cause analysis, and corrective action program

  - A means for reassessing SSC safety significance and QA controls when new information becomes available.

- **NRC Approval of the Graded QA Implementation at South Texas Project was granted on 11/6/97.**

RISK-INFORMED PILOT APPLICATIONS

Risk-Informed Technical Specifications (TS)

- **Program may result in an overall net risk decrease by minimizing unnecessary mode changes and plant shutdowns that subject the plant to an increased likelihood of transients while maintaining high equipment reliability for operating modes when the equipment is most needed.**

- **The NRC has proposed a three tier approach to ensure that undesirable combinations of out-of-service equipment are precluded when extended allowed outage times (AOTs) are utilized:**

  Tier 1: Evaluation of the PRA model and the impact of the proposed change

  Tier 2: Address the need to preclude potentially high risk configurations

  Tier 3: Evaluation of licensees configuration risk management program

- **The staff has issued risk-informed Technical Specification amendments for a number of different systems. (e.g. Low Pressure Safety Injection, Safety Injection Tank, Emergency Diesel Generator Allowed Outage Time extensions)**

RISK-INFORMED PILOT APPLICATIONS

Risk-Informed Inservice Testing (IST)

- **Program implementation might include improved testing methods capable of detecting equipment degradation and extension of IST program to components not currently covered by ASME code requirements. Industry burden may be reduced by increasing allowed testing intervals for low safety significant components.**

- **A risk-informed IST program approach is expected to include:**

  - ❏ Test strategy relaxations and possible improvements to test strategies (i.e., when test intervals are extended, consideration should be given to improved test methods that would detect component degradation)

  - ❏ Focusing of resources commensurate with safety significance

  - ❏ Estimation of overall program effect on plant risk

  - ❏ Performance monitoring and feedback ensure potential problems are promptly detected and corrected

- **The NRC has issued a number of approvals for Risk-Informed Inservice Testing Programs.**

Risk-Informed Inservice Inspection (ISI)

- **Program benefits include improved inspection location selection, utilization of inspection methods that will increase the likelihood of detecting flaws, and reduction in worker radiation exposure. Program might incorporate plant components not currently covered by ASME code requirements.**

- **The general goals of a risk-informed ISI program are to:**

    - ❑ Improve inspection procedures to increase the likelihood of detecting significant flaws.

    - ❑ Consider the possibility of net risk decrease compared to present program through better selection of locations and inspection methods, so that the more safety significant locations are given appropriate attention.

    - ❑ Reduce the number of welds inspected by improving selection process.

- **The NRC has approved risk-informed inservice inspection programs at several US nuclear power plants.**

Implementation Challenges

- ❑ Development of Numerical Acceptance Criteria
- ❑ Treatment of Uncertainties
- ❑ PRA Scope & Quality Required to Support Licensing Changes
- ❑ Completion of First of a Kind Pilot Reviews

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Overview of the USNRC Significance Determination Process

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Overview of the NRC's Significance Determination Process (SDP)

- Secy 99-007, "Recommendations for Reactor Oversight Process Improvements", March 22, 1999, highlighted the need for a method to characterize inspection program findings from a risk perspective (http://www.nrc.gov/NRC/COMMISSION/SECYS/secy1999-007/1999-007scy_attach.pdf)

- Separate processes exist for characterizing findings depending on the nature of the finding(s):
  - Reactor Operations
  - Emergency Preparedness
  - Radiation Safety
  - Safeguards

- Reactor operations includes separate processes for power operations and fire protection (Methods for assessing shutdown conditions and external events are under development, only the reactor SDP is derived from quantitative considerations)

Objectives of the Significance Determination Process

- Characterize the risk significance of an inspection finding consistent with the regulatory response thresholds used for performance indicators in the NRC assessment process

- Provide a risk-informed framework for discussing and communicating the potential significance of inspection findings

- Provide a basis for assessment or enforcement actions associated with an inspection finding

- Specify the minimum amount of documentation needed to reconstruct the basis for decisions associated with the risk significance characterization of inspection findings

General Methodology
(Reactor Power Operations)

- Specific inspection finding is identified, the impact is defined and the critical assumptions are formulated

- Initial "Phase 1" screening is conducted
  Screening is based on a number of qualitative and deterministic criteria (i.e. assessment of functionality vs. operability)
  If the finding "screens" out at Phase 1 it is characterized as "GREEN" otherwise proceed to "Phase 2" ("GREEN" generally indicates a change in core damage frequency of less than 1.0E-06/yr)

- Phase 2 Screening
  -Define applicable scenarios, determine likelihood, identify duration of condition and remaining mitigating capability
  -Assess risk significance using SDP tables
  -For issues which result in a "WHITE", "YELLOW", or "RED" characterization, engage the licensee and perform a "Phase 3" analysis

- Phase 3 Analysis
  -Perform a detailed risk analysis using state-of-the-art PSA model(s) and/or any other available information to determine the best-estimate of the risk of the finding

USE OF RISK INFORMATION IN NRC AND INDUSTRY PROGRAMS

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(Riskinfo.wpd - 7/28/99)

Other Methodology Considerations

- Fire protection significance determination process is similar to reactor power operations but uses a different methodology to determine scenario likelihood which was developed on the basis of assumptions regarding the level of degradation in key fire protection/mitigation elements

- Emergency Preparedness, Radiation Protection and Safeguards processes use a deterministic "flowchart" to determine significance

- Methods for assessing shutdown and external events findings are under development

Row	Approx. Freq.	Example Event Type	Estimated Likelihood Rating		
I	>1 per 1 - 10^{-1} yr	Reactor Trip Loss of Power Conv. Sys. (loss of condensor, closure of MSIVs, loss of feedwater)	**A**	**B**	**C**
II	1 per 10^{-1} - 10^{-2} yr	Loss of Offsite Power Small LOCA (BWR) (Stuck open SRV only) MSLB (outside cntmt)	**B**	**C**	**D**
III	1 per 10^{-2} - 10^{-3} yr	SGTR Stuck open PORV (PWR) Small LOCA (PWR) (RCP seal failures and stuck open SVs only) MFLB MSLB (inside PWR cntmt)	**C**	**D**	**E**
IV	1 per 10^{-3} - 10^{-4} yr	Small LOCA (pipe breaks) ATWS-PWR (elect only)	**D**	**E**	**F**
V	1 per 10^{-4} - 10^{-5} yr	Med LOCA Large LOCA (BWR) ATWS-BWR	**E**	**F**	**G**
VI	<1 per 10^{-5} yr	Large LOCA (PWR) ATWS-PWR (mech only) ISLOCA Vessel Rupture	**F**	**G**	**H**
			> 30 days	30-3days	<3 days
			Exposure Time for Degraded Condition		

Table 1 - Estimated Likelihood for Initiating Event Occurrence During Degraded Period

Initiating Event Likelihood	Remaining Mitigation Capability Rating (with Examples)						
	6	**5**	**4**	**3**	**2**	**1**	**0**
	3 diverse trains OR 2 multi-train systems OR 1 train + 1 multi-train system + recovery of failed train	1 train + 1 multi-train system OR 2 diverse trains + recovery of failed train	2 diverse trains OR 1 multi-train system + recovery of failed train	1 train + recovery of failed train OR 1 multi-train system OR Operator action + recovery of failed train	1 train OR Operator action OR Operator action under high stress + recovery of failed train	Recovery of failed train OR Operator action under high stress	none
A	Green	White	Yellow	Red	Red	Red	Red
B	Green	Green	White	Yellow	Red	Red	Red
C	Green	Green	Green	White	Yellow	Red	Red
D	Green	Green	Green	Green	White	Yellow	Red
E	Green	Green	Green	Green	Green	White	Yellow
F	Green	Green	Green	Green	Green	Green	White
G	Green	Green	Green	Green	Green	Green	Green
H	Green	Green	Green	Green	Green	Green	Green

Table 2 - Risk Significance Estimation Matrix (rev 6/10/99)

Type of Remaining Capability	Remaining Capability Rating
Operator Action Under High Stress Definition: Operator action assumed to have about a 1E-1 probability of failing when credited as "remaining mitigation capability".	1
Recovery of Failed Train Definition: Operator action to recover failed equipment that is capable of being recovered after an initiating event occurs that requires the equipment (e.g., equipment was unavailable due to a switch misalignment). Action may take place either in the control room or outside the control room and is assumed to have about a 1E-1 probability of failing when credited as "remaining mitigation capability".	1
Operator Action Definition: Operator action that can occur with sufficient time to have about a 1E-2 probability of failing when credited as "remaining mitigation capability".	2
1 Train (diverse as compared to other trains) Definition: A collection of associated equipment (e.g., pumps, valves, breakers, etc.) that together can provide 100% of a specified safety function and for which the probability of being unavailable due to failure, test, or maintenance is assumed to be about 1E-2 when credited as "remaining mitigation capability". Two or more trains are diverse if they are not considered to be susceptible to common cause failure modes.	2
1 Multi-Train System Definition: A system comprised of two or more trains (as defined above) that are considered susceptible to common cause failure modes. Such a system is assumed to have about a 1E-3 probability of being unavailable, regardless of how many trains comprise the system, when credited as "remaining mitigation capability".	3
2 (diverse) Trains [adding example] (2 diverse trains are assumed to have a combined 1E-4 probability of being unavailable)	4 (= 2 + 2)
1 Train + Recovery of Failed Train [adding example] (1 train plus recovery of failed train is assumed to have a combined 1E-3 probability of being unavailable or failed)	3 (=2 + 1)

Table 3 - Remaining Capability Rating Values

Significance Determination Process Example
(Reactor Power Operations)

Inspection Finding:

At a (hypothetical) Boiling Water Reactor (BWR) it is determined that due to inadequate design control measures involving the fuel oil systems, both emergency diesel generators (EDGs) would have failed after running for about 6 hours following a loss of offsite power.

Initiating Event Scenarios to be Considered

Affected System	Support Systems	Initiating Event Scenarios
SRVs	air/nitrogen, 125 Vdc	Transient[1], LOOP, SLOCA, MLOCA, ATWS
PCS	offsite power, 125 Vdc, TBCCW, air	Transient[1], SLOCA
RHR	4160 Vac, 125 Vac, RHRSW, Pump Room HVAC	Transient[1], LOOP, ATWS, SLOCA, MLOCA, LLOCA
SBCS	4160 Vac, 125 Vdc, SW	LLOCA, MLOCA, SLOCA, Transient[1], LOOP, ATWS
EDGs	125 Vdc, DGCW, EDG HVAC	LOOP
RHRSW	HVAC, 4160 Vac, 480 Vac, 125 Vdc	Transient[1], LOOP, ATWS, SLOCA, MLOCA, LLOCA
DGCW	480 Vac	Transient[1], LOOP, ATWS, SLOCA, MLOCA, LLOCA
SW	4160 Vac, 125 Vdc, air	Transient[1], LOOP, ATWS, SLOCA, MLOCA, LLOCA
TBCCW	SW, air, 4160 Vac	Transient[1], SBLOCA,
HPCI	125 Vdc, SW, Room HVAC	Transient[1], LOOP, ATWS, SLOCA, MLOCA
CS	4160 Vac, 125 Vdc, SW, Pump Room HVAC	Transient[1], LOOP, ATWS, SLOCA, MLOCA, LLOCA
SSMP	SW, HVAC, 4160 Vac	Transient[1], LOOP, ATWS
RCIC	125 Vdc, SW, Room HVAC	Transient,[1] LOOP, ATWS
Air	offsite power, SW	Transient[1], LOOP, ATWS, SLOCA, MLOCA, LLOCA
SLC	480 VAC, 125 Vdc	ATWS

[1]Note: Transient scenarios should be developed from those transient initiators that could have the greatest risk significance. For example, develop loss of DC bus transient scenarios for degraded 125v DC or AC power equipment, as well as other transient initiators that may depend on equipment being supplied from degraded power sources. The choice of which transient scenarios to develop should generally be apparent from the specific given condition.

WORKSHEET FOR REACTOR AND PLANT SYSTEM DEGRADED CONDITIONS

Reference/Title (LER #, Inspection Report #, etc): **BWR EXAMPLE 3**

Factual Description of Identified Condition (statement of facts known about the issue, without hypothetical failures included):
Due to inadequate design control measures regarding the emergency diesel fuel system, all emergency diesels would have failed after running for about 6 hours following a loss of offsite power.

System(s) and Train(s) with degraded condition: **All emergency diesel generators**

Licensing Basis Function (if applicable): **Mitigate consequences of LOOP**

Maintenance Rule category (check one): __✔__ risk-significant ______ non-risk-significant
Time degraded condition existed or assumed to exist: **Greater than 1 year.**

Functions and Cornerstones degraded as a result of this condition (check ✓)

INITIATING EVENT CORNERSTONE

_____ Transient initiator contributor (e.g., reactor/turbine trip, loss offsite power)

_____ Primary or Secondary system LOCA initiator contributor (e.g., RCS or main steam/feedwater pipe degradations and leaks)

MITIGATION CORNERSTONE

______ Core Decay Heat Removal

__✔__ Initial injection heat removal paths

_____ Primary (e.g., Safety Inj)

_____ Low Pressure

_____ High Pressure

_____ Secondary - PWR only (e.g., AFW)

__✔__ Long term heat removal paths (e.g., contmt sump recirculation, suppression pool cooling)

_____ Reactivity control

BARRIER CORNERSTONE

_____ RCS LOCA mitigation boundary degraded (e.g., PORV block valve, PTS issue)

_____ Containment integrity

_____ Breach or bypass

_✔___ Heat removal, hydrogen or pressure control

_____ Fuel cladding degraded

PHASE 1 SCREENING PROCESS
Check the appropriate boxes ✔

Cornerstone(s) assumed degraded:

☐ Initiating Event ☒ Mitigation Systems ☐ RCS Barrier ☐ Fuel Barrier ☒ Containment Barrier

If more than one Cornerstone is degraded, then go to Phase 2. If NO Cornerstone is degraded, then the condition screens OUT as "Green" and is not assessed further by this process.

If only one Cornerstone is degraded, continue in the appropriate column below.

Initiating Event	Mitigation Systems	RCS Barrier	Fuel Barrier	Containment Barrier
1. Does the issue contribute to the likelihood of a Primary or Secondary system LOCA initiator? ☐ **If YES → Go to Phase 2** If NO, continue 2. Does the issue contribute to both the likelihood of a reactor trip AND the likelihood that mitigation equipment will not be available? ☐ **If YES → Go to Phase 2** ☐ If NO, screen OUT	1. Is the issue a design or qualification deficiency that does NOT affect operability per GL 91-18 (rev 1)? ☐ **If YES → Screen OUT** If NO, continue 2. Does the issue represent an actual Loss of Safety Function of a System? ☐ **If YES → Go to Phase 2** If NO, continue 3. Does the issue represent an actual Loss of Safety Function of a Single Train, for > TS AOT? ☐ **If YES → Go To Phase 2** If NO, continue 4. Does the issue represent an actual Loss of Safety Function of a Single Train of non-TS equipment designated as risk-significant under 10CFR50.65, for > 24 hrs? ☐ **If YES → Go To Phase 2** ☐ If NO, screen OUT	☐ 1. **Go to Phase 2**	☐ 1. Screen OUT	1. TBD

Result of the Phase 1 screening process: _____ screen OUT as "Green" _✔____ go to Phase 2

Important Assumptions (as applicable): **No EDGS would be available following a LOOP.**

Row	Approx. Freq.	Example Event Type	Estimated Likelihood Rating		
I	>1 per 1 - 10^{-1} yr	Reactor Trip Loss of Power Conv. Sys. (loss of condensor, closure of MSIVs, loss of feedwater)	A	B	C
II	1 per 10^{-1} - 10^{-2} yr	Loss of Offsite Power Small LOCA (BWR) (Stuck open SRV only) MSLB (outside cntmt)	B	C	D
III	1 per 10^{-2} - 10^{-3} yr	SGTR Stuck open PORV (PWR) Small LOCA (PWR) (RCP seal failures and stuck open SVs only) MFLB MSLB (inside PWR cntmt)	C	D	E
IV	1 per 10^{-3} - 10^{-4} yr	Small LOCA (pipe breaks) ATWS-PWR (elect only)	D	E	F
V	1 per 10^{-4} - 10^{-5} yr	Med LOCA Large LOCA (BWR) ATWS-BWR	E	F	G
VI	<1 per 10^{-5} yr	Large LOCA (PWR) ATWS-PWR (mech only) ISLOCA Vessel Rupture	F	G	H
			> 30 days	30-3days	<3 days
			Exposure Time for Degraded Condition		

Table 1 - Estimated Likelihood for Initiating Event Occurrence During Degraded Period

Type of Remaining Capability	Remaining Capability Rating
Operator Action Under High Stress Definition: Operator action assumed to have about a 1E-1 probability of failing when credited as "remaining mitigation capability".	1
Recovery of Failed Train Definition: Operator action to recover failed equipment that is capable of being recovered after an initiating event occurs that requires the equipment (e.g., equipment was unavailable due to a switch misalignment). Action may take place either in the control room or outside the control room and is assumed to have about a 1E-1 probability of failing when credited as "remaining mitigation capability".	1
Operator Action Definition: Operator action that can occur with sufficient time to have about a 1E-2 probability of failing when credited as "remaining mitigation capability".	2
1 Train (diverse as compared to other trains) Definition: A collection of associated equipment (e.g., pumps, valves, breakers, etc.) that together can provide 100% of a specified safety function and for which the probability of being unavailable due to failure, test, or maintenance is assumed to be about 1E-2 when credited as "remaining mitigation capability". Two or more trains are diverse if they are not considered to be susceptible to common cause failure modes.	2
1 Multi-Train System Definition: A system comprised of two or more trains (as defined above) that are considered susceptible to common cause failure modes. Such a system is assumed to have about a 1E-3 probability of being unavailable, regardless of how many trains comprise the system, when credited as "remaining mitigation capability".	3
2 (diverse) Trains [adding example] (2 diverse trains are assumed to have a combined 1E-4 probability of being unavailable)	4 (= 2 + 2)
1 Train + Recovery of Failed Train [adding example] (1 train plus recovery of failed train is assumed to have a combined 1E-3 probability of being unavailable or failed)	3 (=2 + 1)

Table 3 - Remaining Capability Rating Values

PHASE 2 RISK ESTIMATION WORKSHEET **Transients**

Estimated Frequency (Table 1 Row) ________ Exposure time __________ Table 1 result (circle): A B C D E F G H

Safety Functions Needed:	*Full* Creditable Mitigation Capability for each Safety Function:
Power Conversion System (PCS)	1/3 trains condensate booster pumps etc. (Operator Action)
High Press Injection (HPI)	HPCI or RCIC (1 multi-train system) or SSMP (operator action)
Depressurization (DEP)	1/5 ADS valves (RVs) manually opened (high stress operator action)
Low Press Injection (LPI))	1/4 RHR pumps in LPCI Mode (1 multi-train system) or 1 / 2 LPCS trains (1 multi-train system)
Late Containment Heat Removal (LC)	1/4 RHR trains in SPC Mode (1 multi-train system) or SCSS (high stress operator action)

Circle affected functions	Recovery of failed train	*Remaining* Mitigation Capability Rating for each affected sequence:	Sequence Color
Trans - PCS - LC			
Trans - PCS - HPI - DEP			
Trans - PCS - HPI - LPI			

Identify any operator recovery actions[1] that are credited to directly restore the degraded equipment or initiating event:

Note 1: If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available

Estimated Frequency (Table 1 Row) __________ Exposure time ___________ Table 1 result (circle): A B C D E F G H

Safety Functions Needed:	*Full* Creditable Mitigation Capability for each Safety Function:
Power Conversion System (PCS)	1/3 trains condensate booster pumps etc. (Operator Action)
High Press Injection (HPI)	HPCI or RCIC (1 multi-train system) or SSMP (operator action)
Depressurization (DEP)	1/5 ADS valves manually opened (high stress operator action)
Low Press Injection (LPI)	1/4 RHR pumps in LPCI Mode (1 multi-train system) or 1 / 2 LPCS trains (1 multi-train system)
Late Containment Heat Removal (LC)	1/4 RHR trains in SPC Mode (1 multi-train system) or SCSS (high stress operator action)

Circle affected functions	Recover of failed train	*Remaining* Mitigation Capability Rating for each affected sequence:	Sequence Color
SLOCA - PCS - LC			
SLOCA - PCS - HPI - LPI			
SLOCA - HPI -DEP			

Identify any operator recovery actions[1] that are credited to directly restore the degraded equipment or initiating event:

Note 1: If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available

PHASE 2 RISK ESTIMATION WORKSHEET

Medium LOCA

Estimated Frequency (Table 1 Row) ________ Exposure time __________ Table 1 result (circle): A B C D E F G H

Safety Functions Needed:	*Full* Creditable Mitigation Capability for each Safety Function:
Early Inventory (EI)	HPCI (1 train)
Early Cont. Control (EC)	Passive operation of SP with 1/8 vacuum breakers (1 multi-train system)
Depressurization (DEP)	Operator opens 1/5 ADS valves (High stress operator action)
Late Inventory Control (LI)	1/4 RHR pumps in LPCI Mode (1 multi-train system) or 1 / 2 LPCS trains (1 multi-train system)
Late Cont. P/T Control (LC)	1/4 RHR trains in SPC Mode (1 multi-train system) or SCSS (High stress operator action)

Affected Sequences (circle affected functions):	Recover of failed train	*Remaining* Creditable Mitigation Capability for each affected sequence:	Sequence Color
MLOCA - LI			
MLOCA - LC			
MLOCA - EI - DEP			
MLOCA - EC			

Identify any operator recovery actions[1] that are credited to directly restore the degraded equipment or initiating event:

Note 1: If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available

Estimated Frequency (Table 1 Row) ________ Exposure time __________ Table 1 result (circle): A B C D E F G H

Safety Functions Needed:	*Full* Creditable Mitigation Capability for each Safety Function:
Early Inventory (EI)	1/4 RHR pumps in LPCI mode (1 multi-train system) or 1 / 2 LPCS trains (1 multi-train system)
Early Cont. Control (EC)	Passive operation of SP with 1/8 vacuum breakers (1 multi-train system)
Late Cont. P/T Control (LC)	1/4 RHR trains in SPC Mode (1 multi-train system) or SCSS (High stress operator action)

Circle affected functions:	Recovery of failed train	*Remaining* Mitigation Capability Rating for each affected sequence:	Sequence Color
LLOCA - EC			
LLOCA - EI			
LLOCA - LC			

Identify any operator recovery actions[1] that are credited to directly restore the degraded equipment or initiating event:

Note 1: If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available

PHASE 2 RISK ESTIMATION WORKSHEET

Loss of Offsite Power

Estimated Frequency (Table 1 Row) II Exposure time > 1 yr. Table 1 result (circle): A B C D E F G H

Safety Functions Needed:	*Full* Creditable Mitigation Capability for each Safety Function:
Emergency Power (EAC < 5 hrs)	1 / 2 EDGs less than 5 hrs (1 multi-train system)
Emergency Power (EAC > 5 hrs)	1 / 2 EDGs more than 5 hrs (1 multi-train system)
Recovery of LOOP (RLOOP)	Recovery of LOOP (recovery action)
High Press Injection (HPI)	HPCI or RCIC (1 multi-train system) or SSMP (operator action)
Depressurization (DEP)	1/5 ADS valves manually opened (high stess operator action)
Low Press Injection (LPI)	1/4 RHR pumps in LPCI Mode (1 multi-train system) or 1 / 2 LPCS trains (1 multi-train system)
Late Containment Heat Removal (LC)	1/4 RHR trains in SPC Mode (1 multi-train system) or SCSS (high stress operator action)

Circle affected Functions	Recovery of failed train	*Remaining* Mitigation Capability Rating for each affected sequence:	Sequence Color
LOOP - EAC< 5 hrs - HPI		(EAC = 3) + (HPCI & RCIC = 3) + (SSMP = 2) Total = 8	GREEN
LOOP - EAC> 5 hrs - RLOOP		(EAC = 0) + (RLOOP) Total =1	B1 RED
LOOP - HPI - DEP			
LOOP - HPI- LPI		(HPCI & RCIC = 3) + (SSMP = 2) + (RHR = 2) + (LPCS = 2) = 9	GREEN
LOOP - LC		(RHR = 2) + (SCSS = 1) = 3	YELLOW

Identify any operator recovery actions[1] that are credited to directly restore the degraded equipment or initiating event:

Note 1: If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available

Estimated Frequency (Table 1 Row) ________ Exposure time __________ Table 1 result (circle): A B C D E F G H

Safety Functions Needed:	*Full* Creditable Mitigation Capability for each Safety Function:
Over pressure Protection (OVERP)	9/13 Rvs/SRVs (1 multi-train system)
Reactivity Control (SLC)	SLC (high stress operator action)
High Press Injection (HPI)	HPCI or RCIC (1 multi-train system) or SSMP (operator action)
Depressurization (DEP)	1/5 ADS valves manually opened (high stress operator action)
Inhibit ADS and Lvl Control (INH)	operator inhibits ADS and controls RPV level (High stress operator action)
Containment overpressure protection (LC)	1/4 RHR trains in SPC Mode (1 multi-train system) or SCSS (high stress operator action)

Circle affected functions	Recovery of failed train	*Remaining* Mitigation Capability Rating for each affected sequence:	Sequence Color
ATWS - OVERP			
ATWS - SLC			
ATWS - HPI - DEP			
ATWS - INH			
ATWS - LC			

Identify any operator recovery actions[1] that are credited to directly restore the degraded equipment or initiating event:

Note 1: If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available

	Remaining Mitigation Capability Rating (with Examples)						
Initiating Event Likelihood	**6**	**5**	**4**	**3**	**2**	**1**	**0**
	3 diverse trains OR 2 multi-train systems OR 1 train + 1 multi-train system + recovery of failed train	1 train + 1 multi-train system OR 2 diverse trains + recovery of failed train	2 diverse trains OR 1 multi-train system + recovery of failed train	1 train + recovery of failed train OR 1 multi-train system OR Operator action + recovery of failed train	1 train OR Operator action OR Operator action under high stress + recovery of failed train	Recovery of failed train OR Operator action under high stress	none
A	Green	White	Yellow	Red	Red	Red	Red
B	Green	Green	White	Yellow	Red	Red	Red
C	Green	Green	Green	White	Yellow	Red	Red
D	Green	Green	Green	Green	White	Yellow	Red
E	Green	Green	Green	Green	Green	White	Yellow
F	Green	Green	Green	Green	Green	Green	White
G	Green	Green	Green	Green	Green	Green	Green
H	Green	Green	Green	Green	Green	Green	Green

Table 2 - Risk Significance Estimation Matrix (rev 6/10/99)