# *Risk-Informing the Technical Requirements in 10 CFR 50, Option 3 Public Workshop*

Division of Risk Analysis and Applications
Office of Nuclear Regulatory Research

October 2, 2000

# Workshop Objectives

- Discuss approach and guidelines to be used by the NRC staff in identifying recommended changes to Part 50 (Framework Document)
- Discuss status of work to risk-inform the technical requirements of 10CFR50.46
- Solicit and gather information on each topic from stakeholders

# Program Objectives

- Enhance safety by focusing NRC and licensee resources in areas commensurate with their importance to health and safety
- Provide NRC with the framework to use risk information to take action in reactor regulatory matters
- Allow use of risk information to provide flexibility in plant operation and design, which can result in unnecessary burden reduction without compromising safety

# OPTION 3 INVOLVES TWO PHASES:

- Phase 1: Identify and prioritize candidate design basis accidents (DBAs) and regulations (including their associated regulatory guides and standard review plans) for risk-informing, and identify proposed changes to requirements (feasibility study)

- Phase 2: For proposed changes that are approved by the Commission, develop detailed technical basis and proceed with rulemaking

# Status of Phase 1 Activities:

- SECY-00-0086 — provided preliminary framework document and recommendation to expedite evaluation of 50.44
- SECY-00-0198 — provided recommendations for a risk-informed 50.44, updated framework, and two policy issues
- Currently assessing preliminary recommendations on a risk-informed 50.46, and initial work on special treatment requirements

# Workshop Structure

- Presentations given without interruption, detailed questions and comment period will be held immediately after presentations on that topic
- Individuals are to speak at a microphone, state their name and affiliation
- Blank forms are available in each package and at each table for written comments
- All presentations, questions and comments (whether verbal or written) will be summarized in a workshop proceeding
- Workshop agenda times may be adjusted to match questions, comments and discussions
- Blank registration form in package, please complete and turn in

# Workshop Agenda

| | | | |
|---|---|---|---|
| 8:00 am | -- | 8:15 am | Introduction |
| 8:15 am | -- | 8:45 am | Discussion on 50.44 |
| | | | Presentation by B. Christie |
| | | | NRC presentation |
| | | | Open discussion |
| 8:45 am | -- | 10:15 am | Discussion on Framework |
| | | | NRC presentation |
| | | | Open Discussion |
| 10:15 am | -- | 10:30 am | BREAK |
| 10:30 am | -- | 12:00 pm | Discussion on 50.46 |
| | | | NRC presentation |
| 12:00 pm | -- | 1:15 pm | LUNCH |
| 1:15 pm | -- | 2:40 pm | Discussion on 50.46 (cont'd) |
| | | | WOG presentation |
| | | | BWR OGs presentation |
| | | | Open Discussion |
| 2:40 pm | -- | 3:00 pm | BREAK |
| 3:00 pm | -- | 3:30 pm | Discussion on 50.46 (cont'd) |
| | | | Open Discussion |
| 3:30 pm | -- | 4:00 pm | WRAPUP |

# Recommendations for a Risk- Informed 50.44

# RECOMMENDATIONS

- Specify in the regulation a specific combustible gas source term
  - Use best available calculational methods for a severe accident that includes in-vessel (and ex-vessel) hydrogen and carbon monoxide generation in such a way that the alternative regulation addresses the likely sources of combustible gases.
- Eliminate the requirement to measure hydrogen concentration in containment.
  - Hydrogen monitors have a limited significance in mitigating the threat to containment in the early stages of a core-melt accident.
- Retain the requirement to ensure a mixed atmosphere.
  - The intent of this requirement is to maintain those plant design features (e.g., open compartments) that promote atmospheric mixing and is considered an important defense-in-depth element (i.e., meeting the intent of GDC 50).

# RECOMMENDATIONS (cont'd)

- Eliminate the requirement to control combustible gas concentration resulting from a postulated LOCA.
  - This type of accident is not risk significant and the means to control combustible gas concentration (e.g., recombiners) does not provide any benefit for the risk-significant accidents or, if a vent-purge method is used, can result in unnecessary releases of radioactive material to the atmosphere. Long-term combustible gas control is addressed in Item 9 below.
- Retain the requirement the requirement to inert Mark I and Mark II containments.
  - Removal of this requirement would result in the integrity of these containments being highly vulnerable to gas combustion.
- Retain the requirement for high point vents in the reactor coolant system (RCS).
  - Combustible gases in the RCS can inhibit flow of coolant to the core, therefore, the capability to vent the RCS provides a safety benefit in its ability to terminate core damage.

# RECOMMENDATIONS (cont'd)

- Modify the requirement for the hydrogen control system for Mark III and ice condenser containments to control combustible gas during risk-significant core-melt accidents (e.g., station blackout).
  - Since the control system uses igniters that are alternating current (ac) dependent, under station blackout conditions, these containments may remain vulnerable to gas combustion..
- Include a performance-based second alternative within this regulation.
  - Allow a licensee to use risk information and plant-specific analysis on the generation and control of combustible gases to demonstrate that the plant would meet specified performance criteria (e.g., maintain containment integrity for at least 24 hours for all risk-significant events). This may be especially attractive to future plants.
- Recommend that long-term (more than 24 hours) control of combustible gas be included as part of the licensee's Severe Accident Management Guidelines (SAMG) since combustible gases still pose a challenge to containment integrity in the long term with the possibility of a large, late radionuclide release.

# Framework for Risk-Informing 10CFR50

# FRAMEWORK

- ***Objective*** Describes the process, approach and guidelines to be applied in reviewing, formulating and recommending risk-informed alternatives to 10 CFR 50 technical requirements

- ***User*** Guidance for the staff, not for licensees

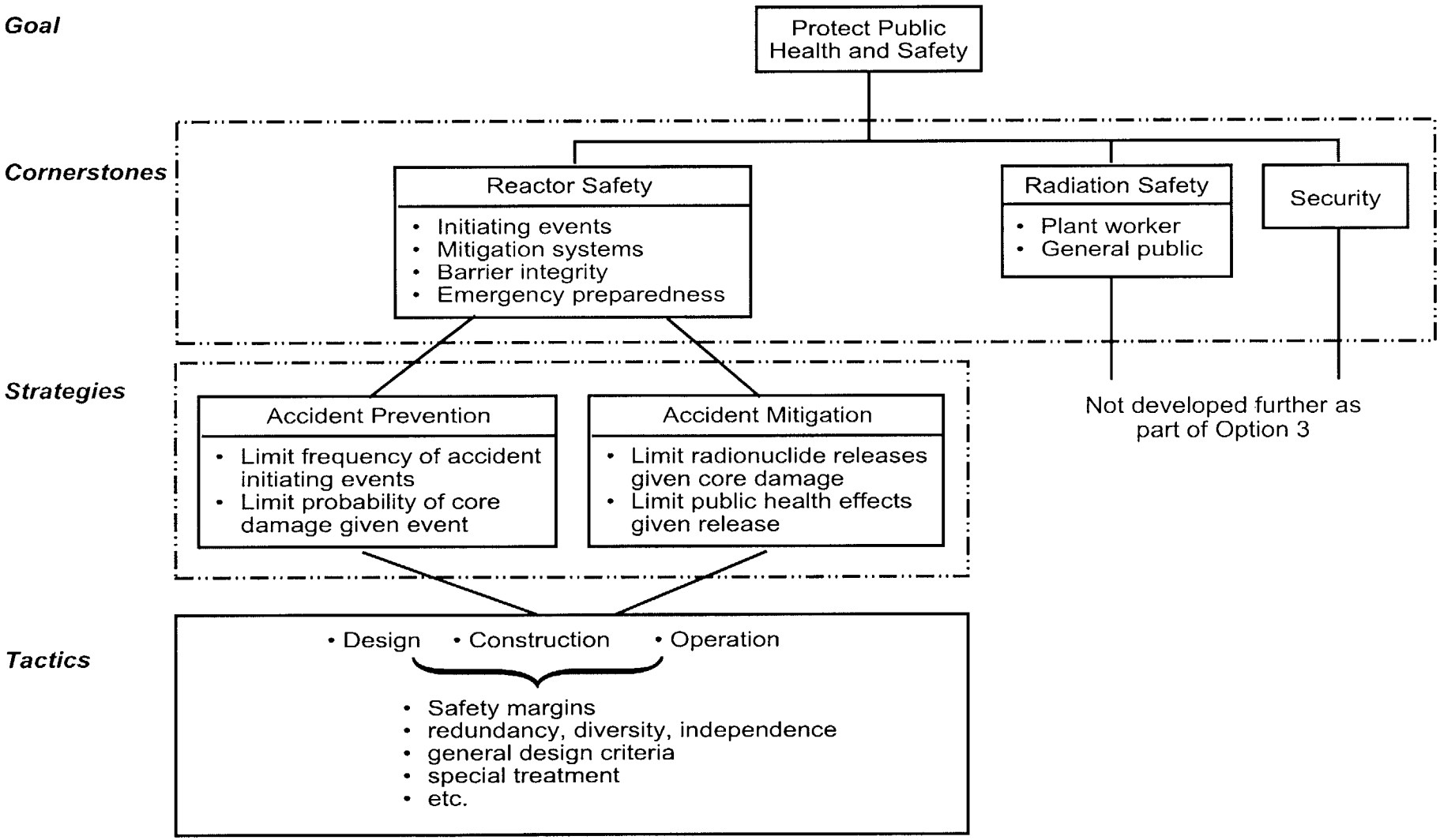
- ***Application*** Process to identify
  - — holes in current requirements
  - — current requirements not contributing to safety

  Used to make generic changes to requirements, not plant-specific changes

# FRAMEWORK

- Maintains the ***goal*** of "protect public health and safety"
- Uses the ***cornerstones*** for Safe Nuclear Power Plant Operation
- Includes ***strategies*** that focua on accident prevention and mitigation
- Defines ***tactics*** to implement prevention and mitigation
- Uses quantitative guidelines, based on the Safety Goal, to measure effectiveness of a regulation and requirements

# FRAMEWORK


Goal
Protect Public Health and Safety
Cornerstones
Reactor Safety
• Initiating events
• Mitigation systems
• Barrier integrity
• Emergency preparedness
Radiation Safety
• Plant worker
• General public
Security
Not developed further as part of Option 3
Strategies
Accident Prevention
• Limit frequency of accident initiating events
• Limit probability of core damage given event
Accident Mitigation
• Limit radionuclide releases given core damage
• Limit public health effects given release
Tactics
• Design • Construction • Operation
• Safety margins
• redundancy, diversity, independence
• general design criteria
• special treatment
• etc.

# QUANTITATIVE GUIDELINES

## Based on Commission's Safety Goals

— Quantitative Health Objectives —

• early fatality safety goal ($\leq 5 \times 10^{-7}$/year) • latent cancer fatality goal ($\leq 2 \times 10^{-6}$/year)

| Accident Prevention | Accident Mitigation |
|---|---|
| **Core Damage Frequency** $\leq 10^{-4}$/year | **Conditional Large Early Release Probability** $\leq 10^{-1}$ |

| | Limit frequency of accident initiating events | Limit probability of core damage given event | Limit radionuclide releases given core damage | Limit public health effects given release |
|---|---|---|---|---|
| | Initiator Frequency | Conditional core damage probability | Conditional large release probability | Conditional individual fatality probability |
| Frequent initiators | $\geq 1$/year | $\leq 10^{-4}$ | $\leq 10^{-1}$ | Note 3 |
| Infrequent initiators | $\leq 10^{-2}$/year | $\leq 10^{-2}$ | $\leq 10^{-1}$ | Note 3 |
| Rare initiators | $\leq 10^{-5}$/year | Note 4 | Note 4 | Note 3 |

Notes:
1. The product across each row gives a large early release frequency of <10-5/year.
2. It is preferable that no single type of initiator cause a large fraction of any frequency uidelines.
3. No quantiative guidelines is proposed for the fourth strategy, the LERF guidelines is used as a surrogate.
4. For rare initiators, emphasis is placed on Strategy 1, limit the initiaotr frequency.
5. Measures to mitigate late large releases are also appropriate. A conditional probability of a late large release (up to 24 hours after the onset of core damage) of $\leq 10^{-1}$ is proposed.

# FRAMEWORK

Implementation and Policy Issues

- The definition of defense-in-depth
- Use of Safety Goals
- Selective implementation
- Backfit considerations

# FRAMEWORK — IMPLEMENTATION ISSUE

- Risk-informed defense-in-depth approach
- Builds on defense-in-depth elements in RG 1.174
- Defined by accident prevention and mitigation strategies using
  - Reactor oversight cornerstones
  - ACRS recommendation of blending “structuralist” and “rationalist” views
- Strategies whose implementation contain elements
  - Tempered by quantitative risk insights
  - Not adjusted using risk insights

# DEFENSE-IN-DEPTH DEFINITION

- Defense-in-depth is the approach taken to protect the public by applying the following strategies in a risk-informed manner:
  - limit the frequency of accident initiating events
  - limit the probability of core damage given accident initiation
  - limit radionuclide releases during core damage accidents
  - limit public health effects due to core damage accident

- The strategies consider the following defense-in-depth elements:
  - reasonable balance is provided among the strategies
  - over- reliance is avoided on programmatic activities to compensate for weaknesses in plant design.
  - independence of barriers is not degraded.
  - safety function success probabilities commensurate with accident frequencies, consequences, and uncertainties are achieved via appropriate
    - redundancy, independence, and diversity,
    - defenses against common cause failure mechanisms,
    - defenses against human errors, and
    - safety margins
  - the defense-in-depth objectives of the current General Design Criteria in Appendix A to 10 CFR Part 50 are maintained.

# FRAMEWORK — USE OF SAFETY GOALS

- Safety Goals used to define level of safety
- Consistent with Commission's expectations and past practice
- Basis for quantitative guidelines to be used to screen and measure effectiveness of technical requirement

# FRAMEWORK — POLICY ISSUE

Selective Implementation

- Implementation of risk-informed Part 50 by licensee is voluntary
- Selective implementation to be determined on a case-by-case basis
- Selective implementation within the risk-informed 50.44 should not be allowed
  - Tend to reduce burden without the commensurate safety enhancement where needed
- Risk-informed alternative 50.44 represents a balance between reducing unnecessary burden and safety enhancements that address risk-significant concerns

# FRAMEWORK — POLICY ISSUE

Backfit Considerations

- Risk-informed alternatives may include elimination, modification and addition to the technical requirements
- Implementation of risk-informed alternative is voluntary, therefore, no need for backfit
- Licensees can do their own cost-benefit assessment before volunteering
- Use Generic Safety Issue process to perform backfit analysis on proposed safety enhancement (identified in the risk-informed alternative) to determine if cost-beneficial and, if appropriate, mandatory

# FRAMEWORK — UNCERTAINTIES

Development of Risk-Informed Alternatives

- Minimize the impact of uncertainties on the decision-making process
- Proposed risk-informed alternatives may be impacted by type of uncertainty
- Defense-in-depth elements address completeness uncertainty
- Safety margin can compensate for data and model uncertainty

# FRAMEWORK — UNCERTAINTIES

Safety Margin

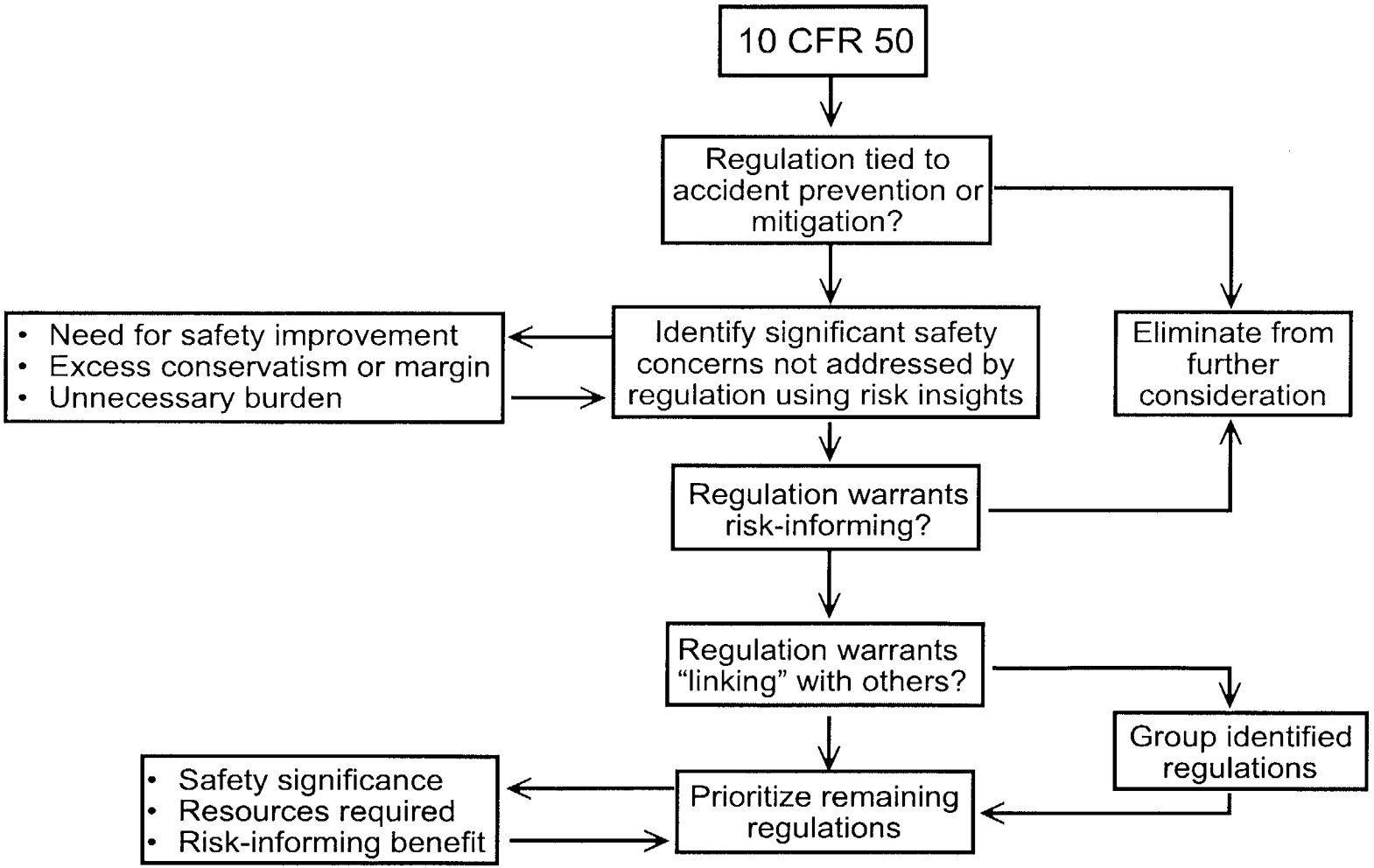
- Implies a measure of the conservatism employed in a design or process to assure a high degree of confidence that it will work to perform a needed function
- Excessive conservatism (i.e., safety margin) can lead to incorrect safety conclusions
- Safety margin imposed to account for uncertainties in data and models
- Framework approach —
  - Specify reasonable safety margin in acceptance criteria using risk insights
  - Use best-estimate calculations to demonstrate compliance based on a computed $95^{th}$ percentile

# FRAMEWORK

Three Major Implementation Tasks

- Selection and prioritization of candidate regulations
- Development of risk-informed alternative to technical requirements
- Evaluation of risk-informed alternative

# FRAMEWORK — SELECTION AND PRIORITIZATION

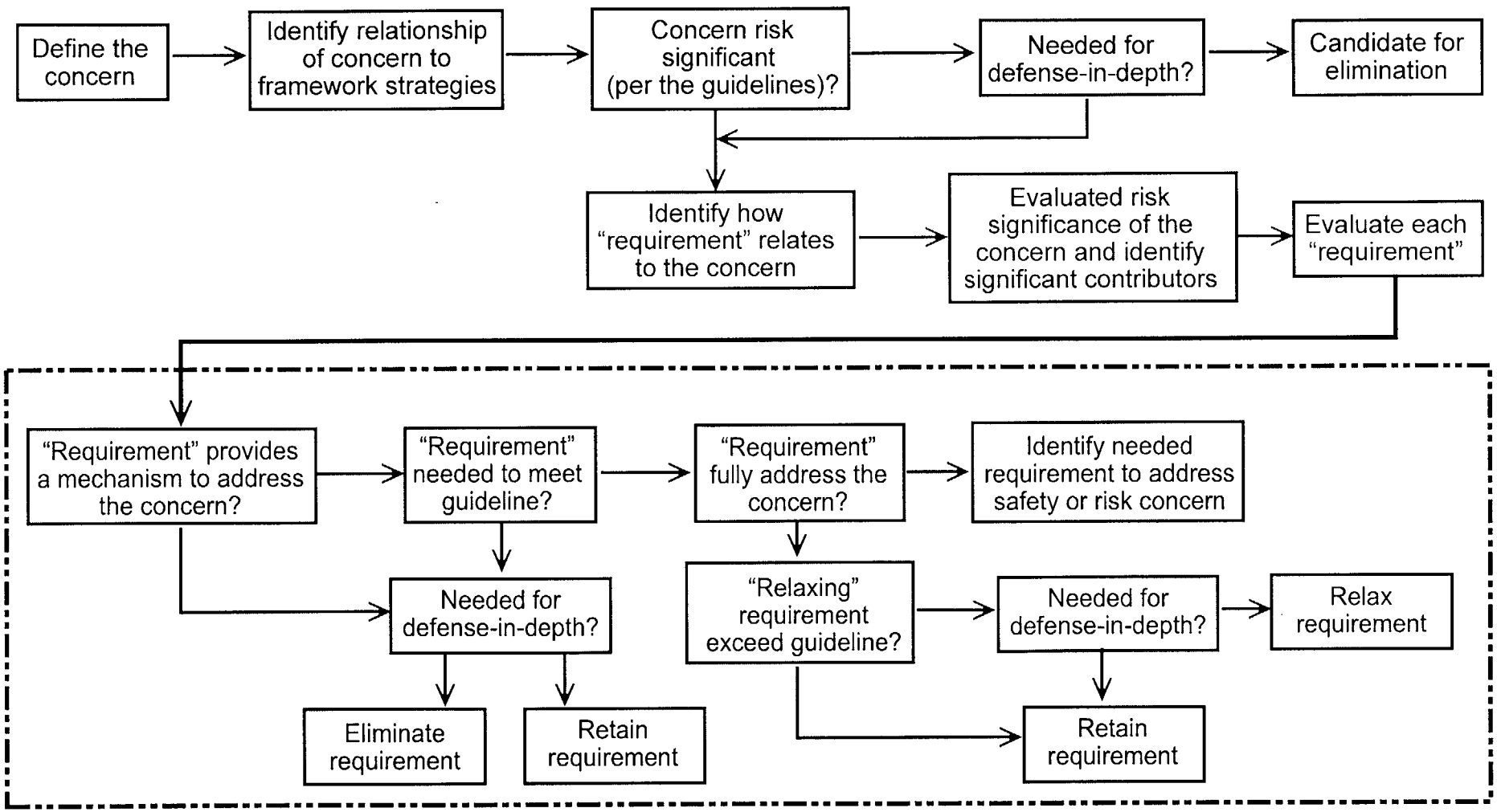


# FRAMEWORK — DEVELOPMENT OF RISK-INFORMED ALTERNATIVE

- Two paths to develop alternative:
  - First, evaluate the current set of technical requirements for either elimination, modification or enhancement
  - Second, apply the four strategies to identify performance-based develop options mechanism(s) to address concern

- Both paths based on same considerations:
  - Generic risk insights from plant-specific PRAs
  - Industry experience
  - Consistent with quantitative guidelines
  - Proven technology

# FRAMEWORK — DEVELOPMENT OF ALTERNATIVE, EVALUATE REQUIREMENTS



# FRAMEWORK — EVALUATION OF ALTERNATIVE

***Factors impacting NRC:***

- Need for rule change
- Impact on other regulations
- Need to revise/modify implementing documents
- Need to create implementing document
- Extent of regulatory analysis required
- Need and extent of NRC review of licensee submittal
- Impact on NRC inspection activities

***Factors impacting Licensees:***

- Need for new/modified equipment
- Need for analysis
- Impact on maintenance and inspection activities
- Impact on technical specifications
- Impact on procedures and training

# DISCUSSION TOPICS

- Defense-in-depth approach
- Implementation of safety margin
- Treatment of uncertainties
- Selection of numerical values for the quantitative guidelineswhile RG 1.174 measures individual plant risk
- Treatment of late containment failures

# Preliminary Status of 50.46

# OUTLINE

Part 1 - Candidate Regulatory Requirements

Part 2 - Risk Significance of LOCAs and ECCS

Part 3 - Potential Risk-Informed Options

# Part 1
# Candidate Regulatory Requirements

# DESIGN-BASIS AND LOSS-OF-COOLANT ACCIDENTS

- Design Basis Accident (DBA)
  - A design basis accident is one that is postulated in order to evaluate particular aspects of a plant against acceptance criteria specified in regulations or implementing documents.
- Loss of coolant accident (LOCA)
  - "Loss of coolant accidents mean those postulated accidents that result from the loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system from breaks in the reactor coolant pressure boundary." [10 CFR 50 Appendix A]
  - Regulatory definition is more narrowly focused than PRA usage

# EMERGENCY CORE COOLING [GDC 35]

- "A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following <u>any loss of reactor coolant</u> at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

- Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (<u>assuming offsite power is not available</u>) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, <u>assuming a single failure</u>."

# THE ECCS PERFORMANCE CONCERN

- Accidents with LOCA initiators and other (e.g., transient-initiated) accidents could proceed to core meltdown as a result of ECCS failures

- Examples of potential LOCA initiators include
  - Throughwall crack in reactor coolant system pipe
    - if undetected could grow sufficiently to result in pipe rupture
  - Seismic event in excess of safe shutdown earthquake (SSE)
  - Draindown event during plant shutdown

# LOCA SIZE CATETORIES

- Small LOCA
  - RCS does not depressurize quickly enough for the low pressure systems to automatically inject
  - Low capability systems (i.e. 100 to 1500 gpm) are sufficient to make up the inventory depletion
- Medium LOCA
  - RCS does not depressurize quickly enough for the low pressure systems to automatically inject
  - High capability systems (i.e. 1500 to 5000 gpm) are sufficient to make up the inventory depletion
- Large LOCA
  - RCS depressurizes to the point where low pressure system must inject automatically to prevent core damage

# NUREG-1150 PIPE-BREAK SIZES

- Small LOCA
  - BWR Steam Piping (<4 inch inside diameter)
  - BWR Liquid Piping (<1 inch inside diameter)
  - PWR (0.5 to 2 inch inside diameter)
- Medium LOCA
  - BWR Steam Piping (4 to 5 inch inside diameter)
  - BWR Liquid Piping (1 to 5 inch inside diameter)
  - PWR (2 to 6 inch inside diameter)
- Large LOCA
  - BWR (>5 inch inside diameter)
  - PWR (>6 inch inside diameter)

# PIPE-BREAK TERMINOLOGY

- Double-Ended Guillotine Break (DEGB)
  - A pipe rupture in which the two pipe ends separate fully resulting in blowdown of fluid from both ends
- Direct Breaks
  - Pipe ruptures due to the growth of cracks (primarily at welded joints)
- Indirect Breaks
  - pipe ruptures caused by failures (primarily seismically-induced) of critical supports or equipment

# IDENTIFYING EXISTING LOCA-RELATED REGULATORY REQUIREMENTS

| Framework Strategy | Keyword | Documents To Screen | To Identify LOCA |
|---|---|---|---|
| 1 | RCSPB<br>Leak detection<br>Pipe break<br>LBB | 10 CFR Sections | Regulatory Requirements |
| 2 | ECC<br>GDC-35<br>50.46<br>LOCA | SRP/FSAR Sections<br>Regulatory Guides<br>Branch Technical Positions<br>Task Action Plan Items | Regulatory Guidance |
| 3 | GDC-38<br>GDC-41<br>GDC-50 | Unresolved Safety Issues<br>Generic Safety Issues | Potential Excess Burden |
| 4 | SRP15.6.5 | | |

# LOCA-RELATED REGULATIONS (Partial List)

Strategy 1 - Prevent Initiators

- 50.55a - Codes and Standards
- 50.60 - Acceptance criteria for fracture prevention
- 50.61 - Fracture toughness requirements ... PTS
- GDC-14 - RCPB design, fabrication, erection, testing
- GDC-14 - RCS design
- GDC-30 - RCPB quality
- GDC-31 - RCPB fracture prevention
- GDC-32 - RCPB inspection
- GDC-35 - RCPB leak detection

# LOCA-RELATED REGULATIONS (Partial List)

Strategy 2 - Prevent core damage given initiator

- 50.46 - ECCS Acceptance Criteria
- GDC 4 - Environmental and dynamic effects design bases
- GDC 17 - Electric power systems
- GDC 27 - Reactivity control
- GDC 33 - Reactor coolant makeup
- GDC 35 - Emergency core cooling
- GDC 36 - Inspection of ECCS
- GDC 37 - Testing of ECCS
- Appendix K - ECCS Evaluation Models

# LOCA-RELATED REGULATIONS (Partial List)

Strategy 3 - Contain radionuclides given core damage

- GDC 50 - Containment must "accomodate without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss of coolant accident."
- GDC 38 - Containment heat removal
- GDC 41 - Containment atmospheric cleanup

# LOCA-RELATED REGULATIONS (Partial)

Strategy 4 - Protect public given core damage

- 10 CFR 100 (as implemented in SRP 15.6.5) -Limit offsite LOCA doses to 10 CFR 100 guidelines
  - 300 rem thyroid*
  - 25 rem whole body*
  - 10 CFR 100 refers to "major accident hypothesized for purposes of site analysis or from considerations of possible accident events that would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products."
- Appendix E - Transmit ECC parameters via ERDS

* or alternatively 25 rem TEDE using new ST under 50.67

# EVOLUTION OF LOCA AS DBA

- Before 1966
  - Large LOCAs assummed to lead to core melt
  - Containments designed for DEGBs
- 1965 to 1974
  - AEC Core Cooling Task Force
    - Knowledge base was insufficient to design for meltdowns
    - More reliable, high-capacity ECCS needed
    - Small LOCAs important
  - General Design Criteria developed (11/65 to 2/71)
  - ECCS Rulemaking, 50.46 & App. K (1/72 to 1/74)

# 10 CFR 50.46 - EMERGENCY CORE COOLING ACCEPTANCE CRITERIA

- Peak cladding temperature
  - "The calculated maximum fuel element cladding temperature shall not exceed 2200 F."
- Maximum cladding oxidation
  - "The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation. ..."
- Maximum hydrogen generation
  - "The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react."

# 10 CFR 50.46 - EMERGENCY CORE COOLING ACCEPTANCE CRITERIA

- Coolable geometry
  - "Calculated changes in core geometry shall be such that the core remains amenable to cooling."
- Long-term cooling
  - "After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core."

# ECCS EVALUATION MODELS

- "ECCS cooling performance must be calculated in accordance with an acceptable evaluation model."
  - Appendix K (1972)
    - Required and acceptable features intended to provide a substantial level of conservatism in ECCS performance analyses
    - A few changes over the years (e.g. <1.02 power multiplier)
    - Remaining conservatism (decay heat & oxidation models)
  - SECY-83-472 (1983)
    - A SECY-83-472 model is a best-estimate code with all of the required features of Appendix K
    - Best-estimate peak cladding temperature at 95% probability level must be less than SECY-83-472 model results
  - Realistic (best-estimate) with quantified uncertainty (1988)

# DYNAMIC EFFECTS OF RCS PIPE BREAKS

- Adoption of DEGB as DBA for ECCS raised concerns regarding dynamic effects
  - Jet impingement
  - Pipe whip
  - Assymetric loads on reactor vessel and internals
- Resulted in
  - Jet impingement barriers
  - Snubbers to limit DEGB reaction forces
  - USI A-2 & NUREG-0609 -Assymetric blowdown loads (1975)

# LEAK BEFORE BREAK

- Westinghouse Owner's Group analyses
  - Used advanced fracture mechanics techniques
  - Indicated DEGB of PWR primary loop piping would not occur mechanistically
- NRC Generic Letter 84-04
  - Agreed that DEGB in primary loop piping was unlikely provided it could be demonstrated by fracture mechanics that through-wall flaws would be detected by the plant's leakage monitoring systems long before the flaws could grow to unstable sizes.
- Drawbacks of devices to protect against dynamic effects
  - Makes inservice inspection more difficult
  - Leads to high occupational exposures (person-rems)
  - Removal & reinstallation may damage pipes or other safety-significant components or impede piping system thermal movement

# REVISION TO GDC 4

- Interim Measure
  - NRC permitted use of LBB technology to request exemptions from installing protective devices on PWR primary coolant piping
  - NUREG-1061 set forth limitations and acceptance criteria
- Limited-scope rule (1986) amended GDC 4 to permit use of LBB analyses to eliminate dynamic effects of postulated PWR primary coolant pipe ruptures from the design basis
- Broad-scope rule (1987) ammended GDC 4 to permit use of LBB analyses in high-energy piping (>275 psi or 200 F)
- RG 1.45 provides guidance on leak detection systems

# APPLICATIONS OF LEAK BEFORE BREAK

- NRC has approved 76 PWRs for the application of LBB in the primary coolant system to eliminate pipe whip restraints and jet impingement barriers
- Some licensees have successfully applied LBB to other high-energy lines including
  - Pressurizer surge lines
  - Safety injection accumulator lines
  - Residual heat removal lines
  - Reactor coolant loop bypass piping systems
- Smallest line approved: 6-inch diameter
- LBB has not been approved for BWRs due to intergranular stress corrosion cracking
- Reg Guide for LBB scheduled for issue in 2003

# CONSIDERATION OF APPLYING LBB TO ECCS & EQUIPMENT QUALIFICATION

- NRC solicited public comment on the application of LBB to ECCS and EQ (April 1988)
- Those opposed cited
  - Surry pipe rupture of December 1986
  - GAO report dated March 1988
  - Purported unreliability of ultrasonic testing to detect piping flaws
  - Public statements made by prior Director of NRR (August 1983)
- NRC determined none of these citations discredited the contemplated application

# NRC ACKNOWLEDGED SAFETY BENEFITS OF APPLYING LBB TO ECCS

- Relax requirement for fast-starting emergency diesel generators
  - Fast-start testing degrades bearing, gears, the governor, and power transmission
  - Using LBB arguments would lengthen the required starting time and assist in preserving the reliability of emergency diesel generators
  - Fewer plant scrams and challenges associated with lower ECCS set points might result in reliability improvements for other equipment
- Permit higher fuel peaking limits
  - Fuel configurations could be designed to yield less radial neutron leakage
  - This would reduce the threat of pressurized thermal shock

# REASONS NRC DECIDED NOT TO APPLY LBB TO ECCS AND EQ IN 1988

- NRC argued the safety benefits could be obtained more expeditiously and efficiently under the 1988 revision to 10 CFR 50.46
  - By applying the best-estimate methodology with quantified uncertainty
  - In retrospect, only 20 of 104 units have applied this methodology because it costs substantially more than Appendix K calculations
  - Also, licensees applying best-estimate methods may choose several ways to realize their benefits
- No clear safety benefits of applying LBB to EQ
- Research would be required to develop replacement DBAs
- Rulemaking was estimated to require at least 2 years and considerable effort

# STATEMENT ON APPLICATION OF LBB TO ECCS AND EQ

- Kept open option for future rulemaking
- Encouraged industry to develop quantitative information that could justify a rulemaking
- Primary emphasis should be given to establishing an appropriate substitute or replacement for the double-ended pipe rupture used in ECCS and EQ evaluation
- Recognized industry may develop justification that would allow a limited number of case-by-case exemptions.

# STATEMENT (FR Doc. 89-10505)

Having considered all public comments received, the Commission has decided not to undertake any rulemaking to extend the applicability of LBB to ECCS or EQ at this time. In large part, any safety benefits associated with ECCS can presently be more readily obtained under the recent ECCS rule. The use of exemptions for applying LBB to EQ was permitted in the revision to General Design Criterion 4 (52 FR 41288). This option continues to remain open.

Nonetheless, the Commission has decided to keep open an avenue for future consideration of rulemaking which would permit the application of LBB to ECCS and EQ. The Commission encourages industry to develop quantitative information that could justify the diversion of resources to the rulemaking efforts. Primary attention should be given to establishing an appropriate substitute or replacement for the double-ended pipe rupture used in ECCS and EQ evaluations. The Commission will consider modifying its current ECCS and EQ regulations when adequate technical justification supports the feasibility and benefits of the proposed modifications. In the interim, the Commission recognizes that situations may arise where justification can be developed by the industry for alternative ECCS and EQ requirements. Such justification, if accepted by the Commission pursuant to the existing exemption process, would allow a limited number of case-by-case modifications to ECCS and EQ requirements. This could support future amendments to applicable requirements addressing ECCS and EQ.

# SCOPE: REGULATIONS SELECTED FOR INITIAL RISK-INFORMED EVALUATION

- The selected regulations are those associated with ECCS performance
  - 10 CFR 50.46 - ECCS Acceptance Criteria
  - GDC 35 - Emergency Core Cooling
  - Appendix K - ECCS Perfomance Models
- Rationale for selection:
  - Complex undertaking which needs to be done in step-by-step fashion
  - Regulations associated with Strategy 1 (initiator prevention) appear effective (see Part 2)
  - Most of the perceived unnecessary burden is associated with the selected ECCS regulations
- Design-basis LOCAs for containment are primarily governed by implementing documents not regulations

# ESTIMATES OF POTENTIAL COST SAVINGS BY WOG

- Technical specification requirements related to 10-second diesel generator start times (up to $1,100,000 per plant year)
- Increases in peaking factors for many plants ($100,000 to $300,000 per plant year)
- Potential for 1 to 3% power uprates for plants whose power conversion systems permit such upgrades ($1,700,000 to $2,800,000 per plant year)
- Reductions in analysis and maintenance costs related to post-LOCA control rod insertion and hot leg switchover would be reduced ($50,000 to $300,000 per plant year).

# ESTIMATES OF POTENTIAL COST SAVINGS BY WOG (Cont'd)

- Relaxation of technical specification requirements related to accumulators ($17,000 per plant year)
- Avoidance of the one-time cost associated with reactor vessel internals--barrel baffle bolt replacement for some plants ($3,600,000 to $8,300,000 per plant)
- Reductions in licensee response costs associated with the potential elimination or simplification of generic issues and letters related to 50.46 and design basis LOCAs ($75,000 per plant year).
- Reduction in costs of 50.46 reporting requirements ($20,000 per plant year)

# POTENTIAL COST SAVINGS OF ELIMINATING LARGE-BREAK LOCA as DBA

- Westinghouse, > $700,000 per plant year
- Combustion Engineering, to be determined
- Babcock & Wilcox, to be determined
- General Electric - not limited by 10 CFR 50.46

# HIGH-LEVEL REGULATORY REQUIREMENTS

- Postulate spectrum of breaks [GDC 35, 50.46, App.K]
- Postulate simultaneous loss of offsite power [GDC 35]
- Apply single failure criterion [GDC 35]
- ECCS Acceptance criteria [50.46(b)]
  - Peak cladding temperature < 2200 F
  - Local oxidation limit < 17%
  - Global hydrogen production < 1%
  - Coolable geometry
  - Long-term cooling
- ECCS evaluation model [50.46(a) & Appendix K]
- Reporting requirements [50.46(a)(3)]

# REGULATIONS REFERENCED FROM 50.46

- 50.4 - Written communications
- 50.55 - Conditions of construction permits
- 50.72 - Immediate notification requirements
- 50.73 - Licensee event report system
- 50.82 - Application for termination of license
- GDC 35 - Emergency core cooling
- App.K - ECCS evaluation models

# REGULATIONS REFERENCING 50.46

- 50.8 - Information collection requirements
- 50.34 - Contents of applications, technical information
- 50.44 - Combustible gas control
- App.K - ECCS Evaluation Models

# KEY IMPLEMENTING DOCUMENTS

- SRP 3.6.2 - Determination of rupture locations and dynamic effects associated with the postulated rupture of piping
- SRP 6.3 - Emergency core cooling system
- SRP 15.6.5 - LOCAs resulting from spectrum of breaks
- RG 1.1 - Net positive suction head
- RG 1.14 - Reactor coolant pump flywheel integrity
- RG 1.45 - RCPB leakage detection systems
- RG 1.82 - Water sources for long-term recirculation cooling
- RG 1.157 - Best-estimate calculations of ECCS performance

# INDUSTRY IMPLEMENTATION OF ECCS PERFORMANCE MODEL REQUIREMENT

- Appendix K - 51 units
- SECY-83-472 - 33 units
  - 29 of 33 BWRs for at least one fuel type
  - Most two-loop Westinghouse PWRs
- Best-estimate with uncertainty quantification - 20 PWRs

# Part 2
# Risk Significance of LOCAs and ECCS

# RCS PIPE BREAK LOCA FREQUENCIES

- Most PRAs (NUREG-1150 and IPEs) have used LOCA frequency estimates that have ties to WASH-1400
- WASH-1400 values were based upon data from both nuclear and non-nuclear, US and foreign sources
- No RCS pipe breaks have occurred in commercial US nuclear power plant history - results in following:
  - BWR LOCA frequency = 7E-4/plant year
  - PWR LOCA frequency = 4E-4/plant year
- Recent LOCA frequency estimates (NUREG/CR-5750) based on frequency of rupture given presence of a through-wall crack
- Other frequency estimates (EPRI) based upon number of pipe segments and welds

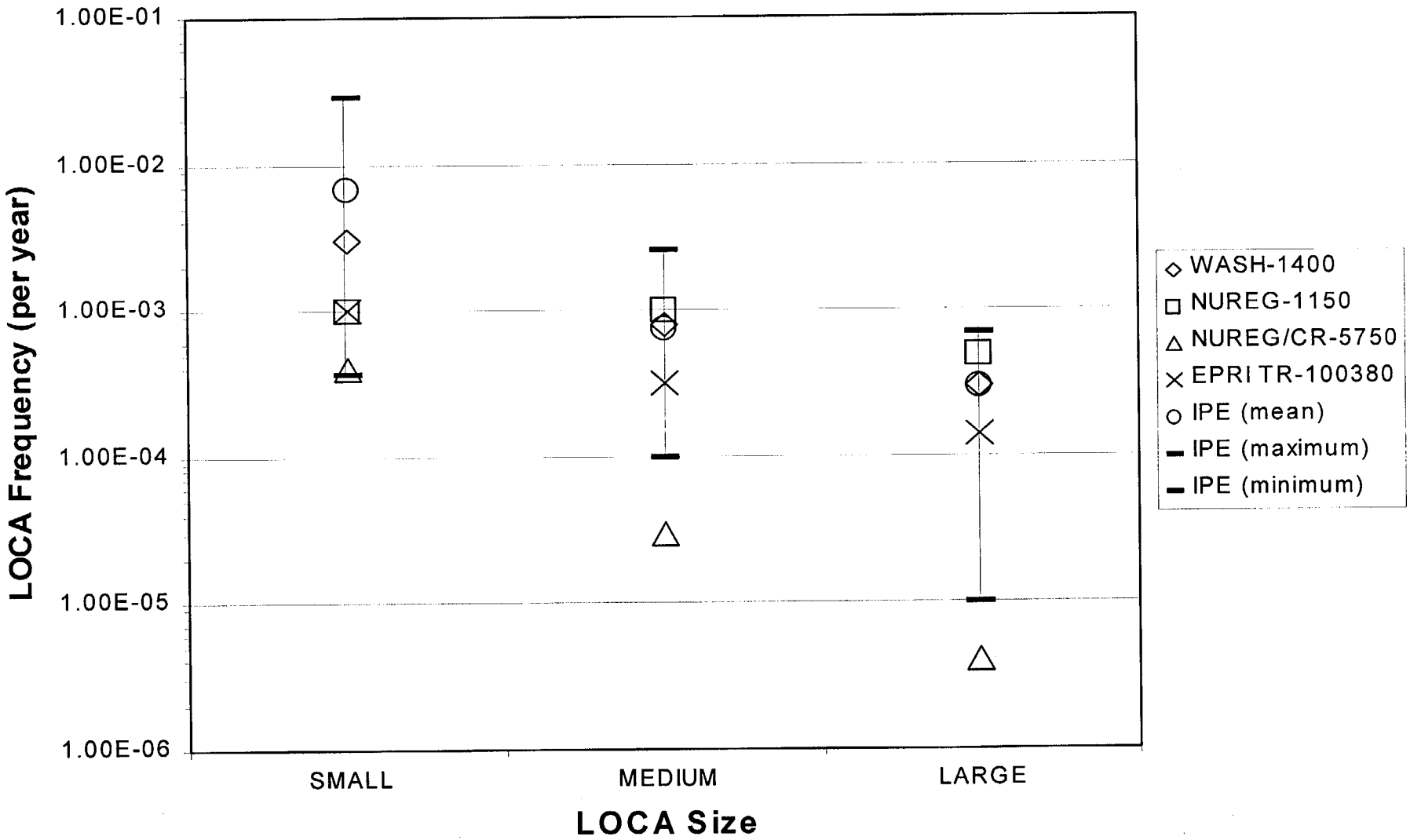
# NUREG/CR-5750 METHODS FOR ESTIMATING PIPE-BREAK LOCA FREQUENCIES

- Estimate through-wall crack frequency based on data
- Adjust downward for IGSCC mitigation (BWRs)
- Multiply by conservative estimate of probability of rupture given a through-wall crack ($P_{R:TW}$) based on
  - Technical review of information on fracture mechanics
  - Data on high-energy pipe failures and cracks
  - Assessments of pipe-break frequencies by others
  - $P_{R:TW}$ = max(2.5/diam(mm)), 0.01)
    - 0.1 for 1" pipe
    - 0.01 for >10" pipe

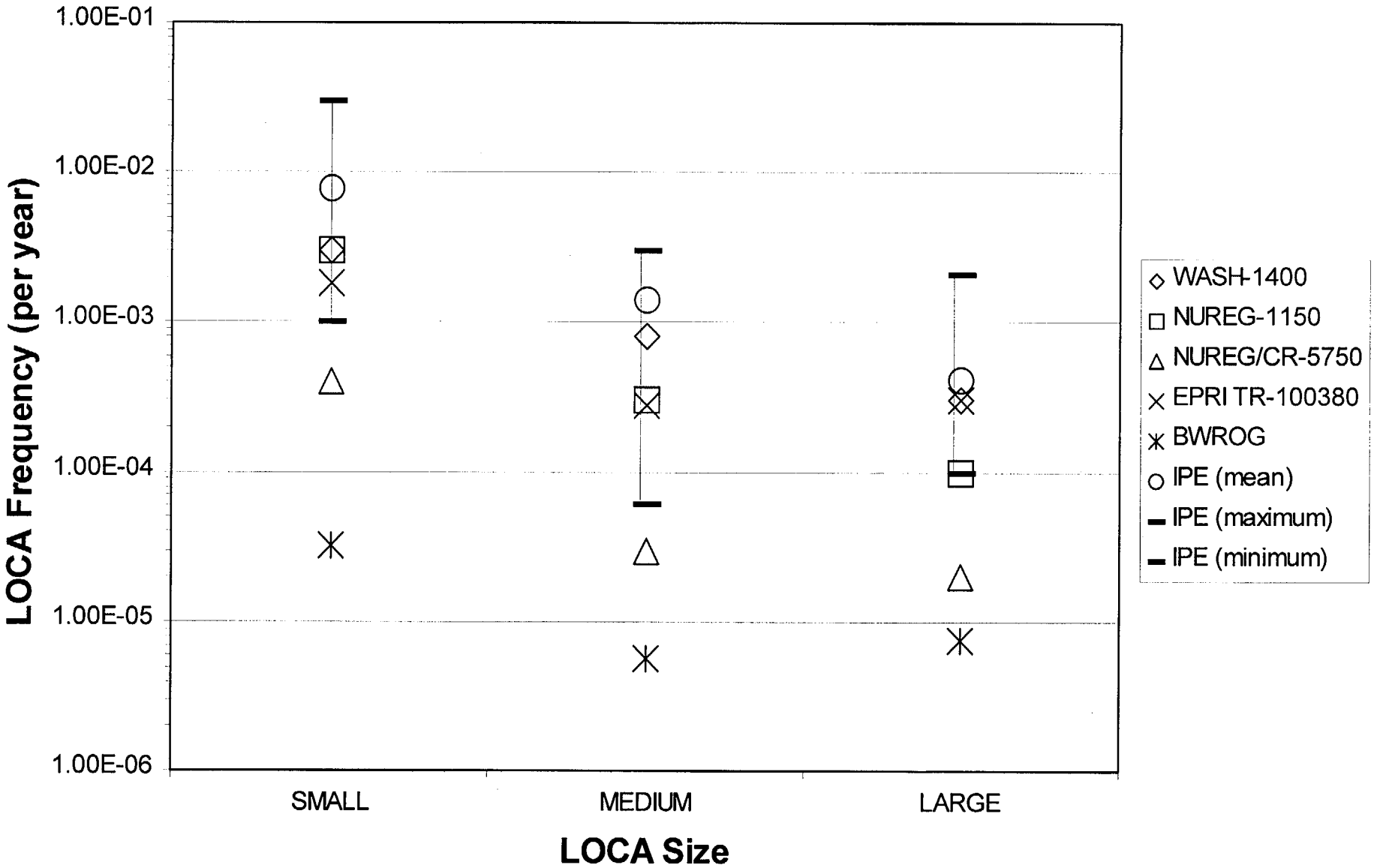
# THROUGH-WALL CRACK DATA (NUREG/CR-5750)

- Through-wall cracks
  - PWRs
    - Data from 3362 years of U.S. and foreign PWR operation
    - Dominant mechanism is thermal fatigue
    - One large (8" pipe), five medium (2" to 6" pipes)
  - BWRs
    - Data from 710 years of U.S. BWR operation
    - Dominant mechanism is intergranular stress corrosion cracking (IGSCC)
    - Most in recirculation bypass lines and riser pipe welds
    - 34 in large pipes (> 10"), 15 in medium pipes (4" to 6")
    - One since IGSCC mitigation efforts began in mid-1980s
- Only 3 U.S. through-wall cracks discovered by leak detection systems while operating at power

# PWR Pipe Break LOCA Frequencies



# BWR Pipe Break LOCA Frequencies



# DOUBLE-ENDED GUILLOTINE BREAKS (DEGB)

- Frequency of DEGBs have been estimated using fracture mechanics as part of LBB assessment (NUREG-1061 and NUREG/CR-4792)
- For CE plants,
  - Point estimates range from 6E-14/yr to 5E-13/yr
  - 90 percentile values range from 4E-12/yr to 7E-11
- For Westinghouse plants,
  - Median frequencies range from 2E-13/yr to 3E-11/yr
  - 90 percentile values range from 8E-10/yr to 1E-9/yr)
- For a BWR (Brunswick), DEGB frequencies (with IGSCC mitigation) range from 1E-12/yr to 4E-12/yr

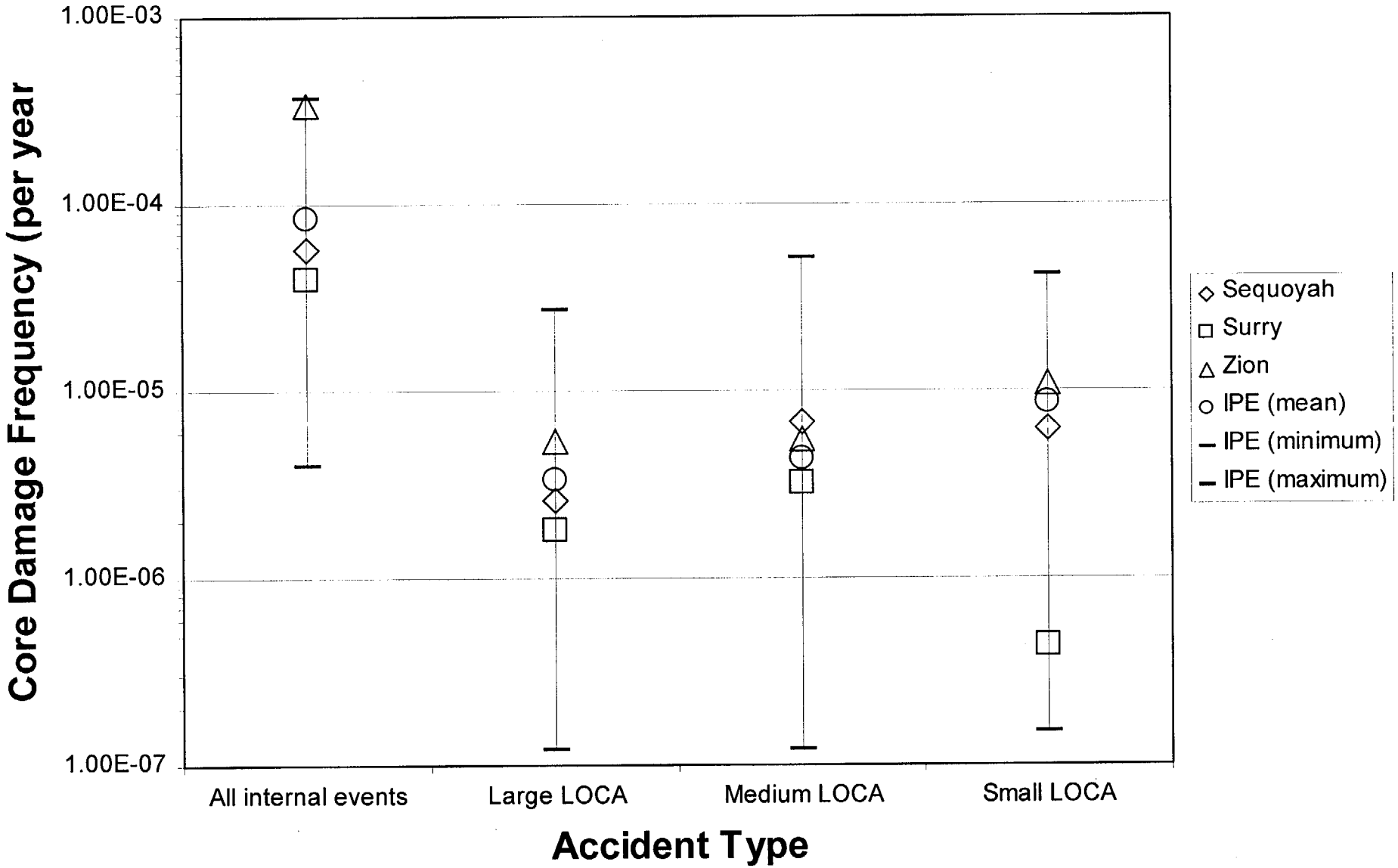
# SEISMIC-INDUCED LOCA FREQUENCIES

- Estimates using fracture mechanics provided in LBB analysis in NUREG/CR-3660 and NUREG/CR-3663
  - Frequency of direct seismic-induced pipe break LOCAs are 1 to 3 orders of magnitude lower than random LOCAs
  - Frequency of indirect seismic-induced DEGBs was significantly higher than frequency of direct seismic-induced DEGBs at many plants
    - CE: median frequencies range from 5E-17/yr to 6E-6/yr
    - W: median frequencies range from 5E-8/yr to 5E-6/yr
- Peach Bottom and Surry (NUREG-1150) studies indicated LBLOCA was dominated by RCS pump and SG (Surry) support failures, smaller breaks caused by piping failure

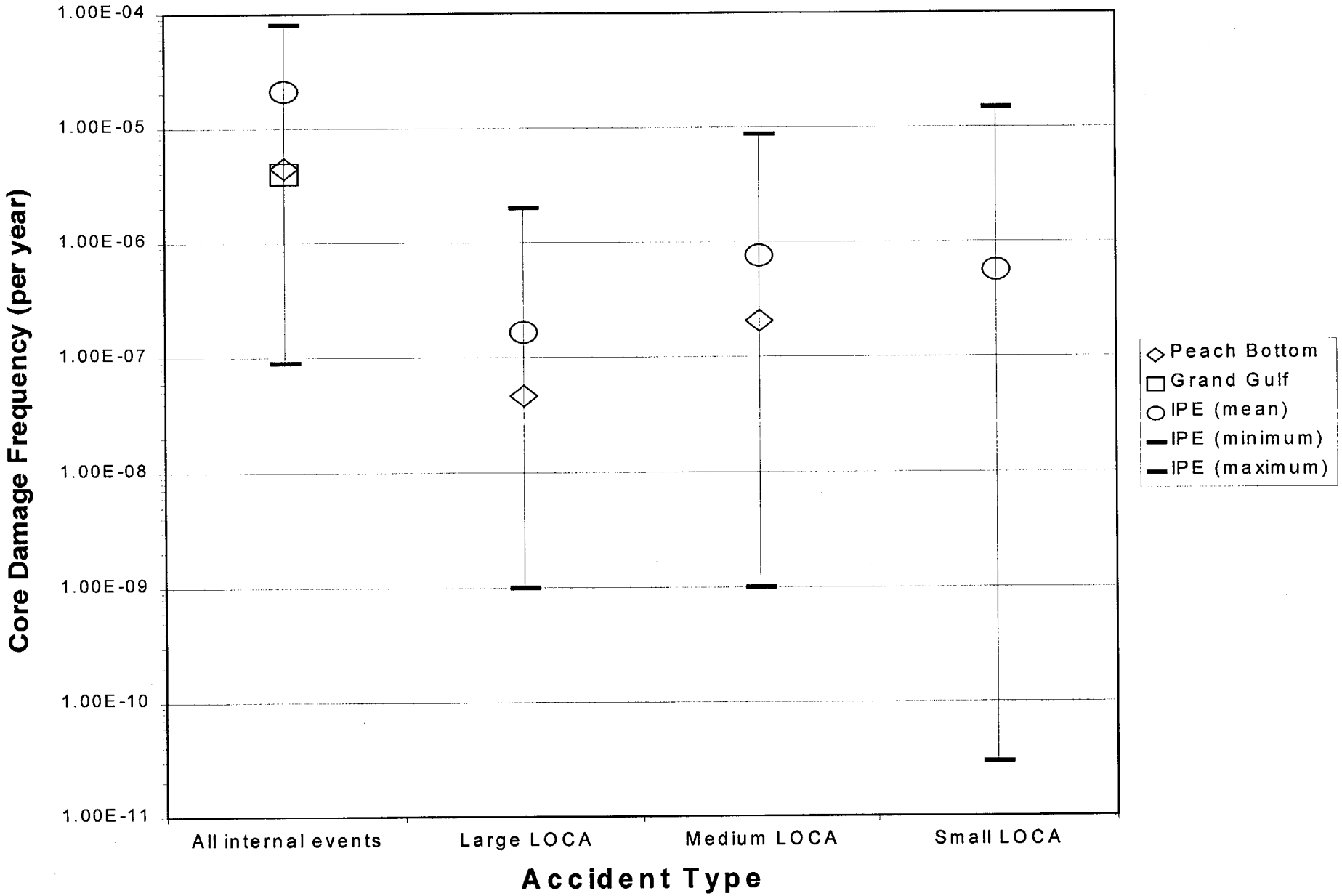
# LOCA CONTRIBUTION TO CDF

- LOCAs relatively unimportant for BWRs due to high redundancy and diversity in coolant injection systems (Negligible to 20% of total CDF calculated in IPEs)
- LOCAs important for most PWRs (5% to 60% of total CDF calculated in IPEs), contribution affected by:
  - Method for switchover to recirculation
  - Size of RWST and ability to refill it
  - Ability to depressurize RCS to mitigate SBLOCA
  - Containment spray actuation
- CDF estimates may be high due to conservative LOCA frequencies
- However, modeling of LOCAs in most PRAs have not addressed some potentially important phenomena (e.g., asymetrical loads and sump plugging)

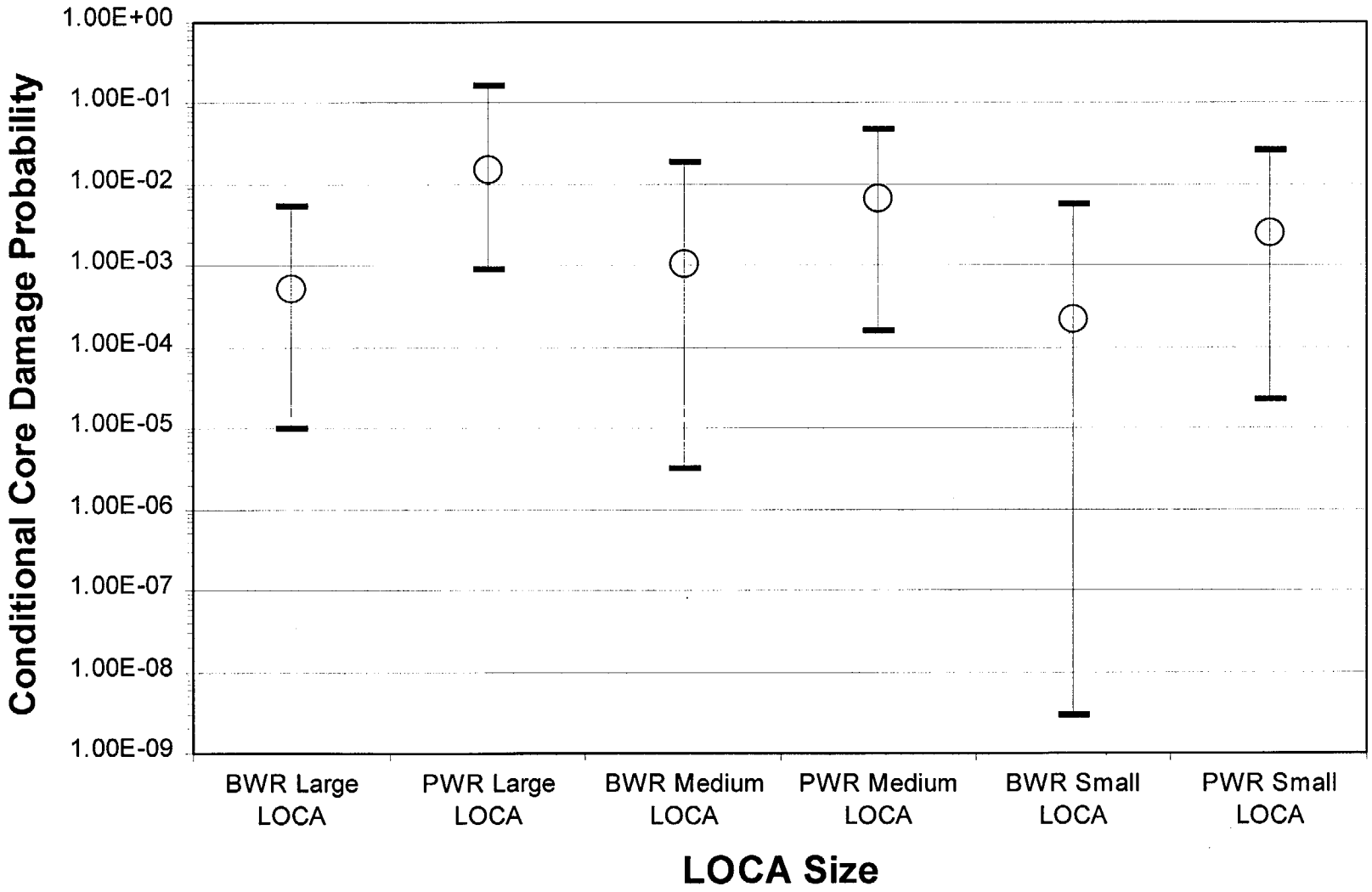
# PWR Pipe Break LOCA CDFs



# BWR Pipe Break LOCA CDFs



# LOCA Conditional Probability of Core Damage (IPEs)



# CDF FROM LOCAs DURING SHUTDOWN

| Accident Initiating Event | Initiating Event Frequency | Core Damage Frequency | | |
|---|---|---|---|---|
| | | Cold Shutdown (POS 5) | Refueling (POS 6) | Refueling (POS 7) |
| **Grand Gulf** | | | | |
| Large LOCA | 3.6E-05 | 4.8E-07 | Screened | Screened |
| Large LOCA during hydro test | 1.3E-04 | 2.1E-07 | NA | NA |
| Medium LOCA | 3.6E-05 | 2.5E-07 | Screened | Screened |
| Medium LOCA during hydro test | 1.3E-04 | 2.1E-07 | NA | NA |
| Diversion to suppression pool via RHR | 6.1E-02 | 1.3E-07 | 1.3E-08 | 7.6E-09 |
| LOCA in RHR | 1.6E-02 | 2.1E-08 | 4.2E-07 | 3.7E-07 |
| Total | | 1.3E-06 | 4.3E-07 | 3.8E-08 |
| **Surry** | | | | |
| Large LOCA | | 2.5E-6 | | |

# CDF FROM SEISMIC-INDUCED LOCAs

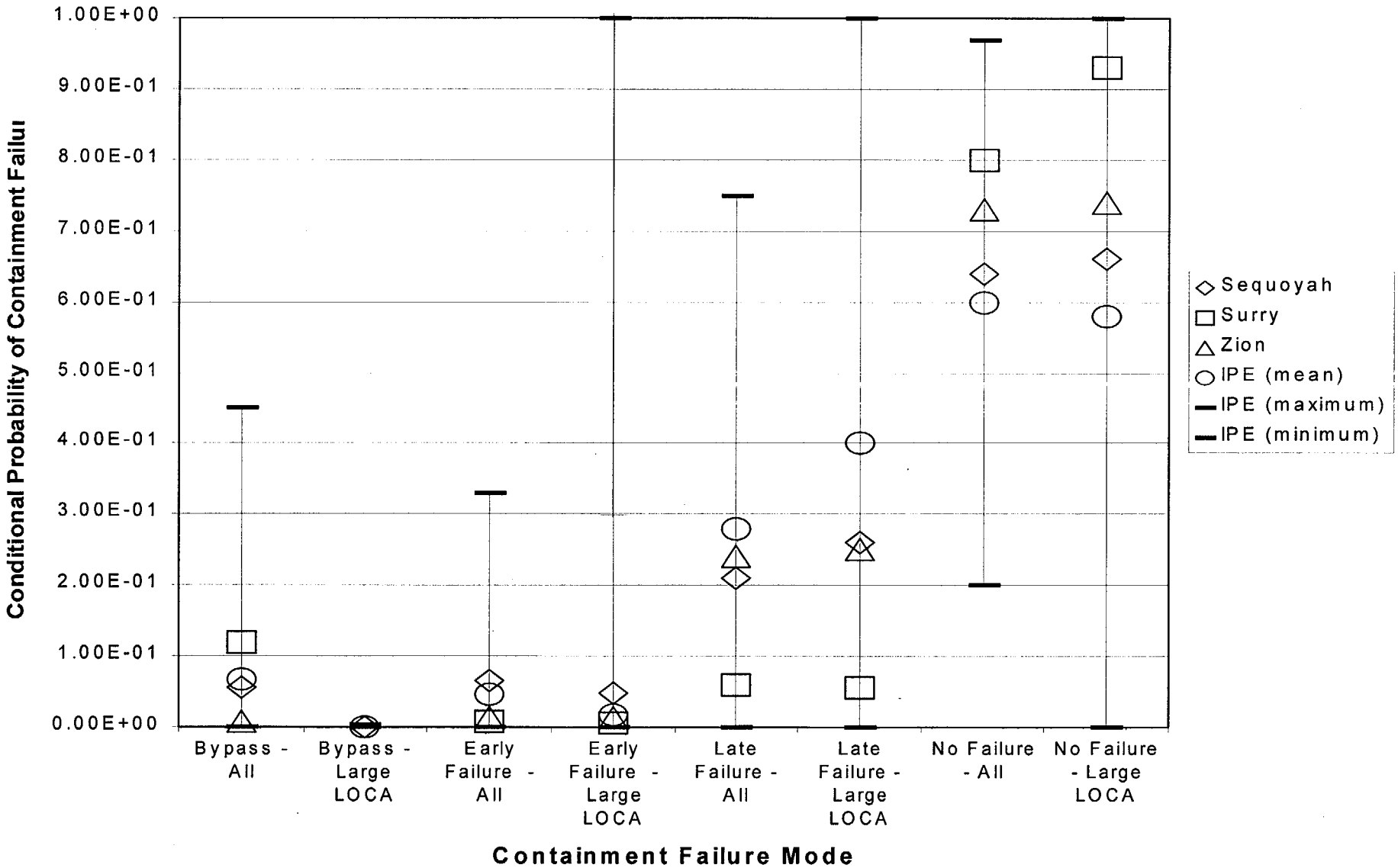
| Accident Type | Mean Core Damage Frequency | |
|---|---|---|
| | LLNL Hazard Curve | EPRI Hazard Curve |
| **Large LOCA** | | |
| Peach Bottom (NUREG-1150) | 1.9E-05 | 6.8E-07 |
| Surry (NUREG-1150) | 7.7E-06 | 1.3E-06 |
| **Medium LOCA** | | |
| Peach Bottom (NUREG-1150) | 7.4E-06 | 2.1E-07 |
| Surry (NUREG-1150) | 1.5E-06 | 1.7E-07 |
| **Small LOCA** | | |
| Peach Bottom (NUREG-1150) | 1.5E-06 | 5.5E-08 |
| Surry (NUREG-1150) | 6.8E-06 | 1.3E-06 |
| **Vessel Rupture** | | |
| Peach Bottom (NUREG-1150) | 8.9E-06 | 3.3E-07 |
| Surry (NUREG-1150) | 3.3E-06 | 5.5E-07 |

- Results are dominated by LOCA with LOOP
- CDFs are comparable to CDFs from random LOCAs

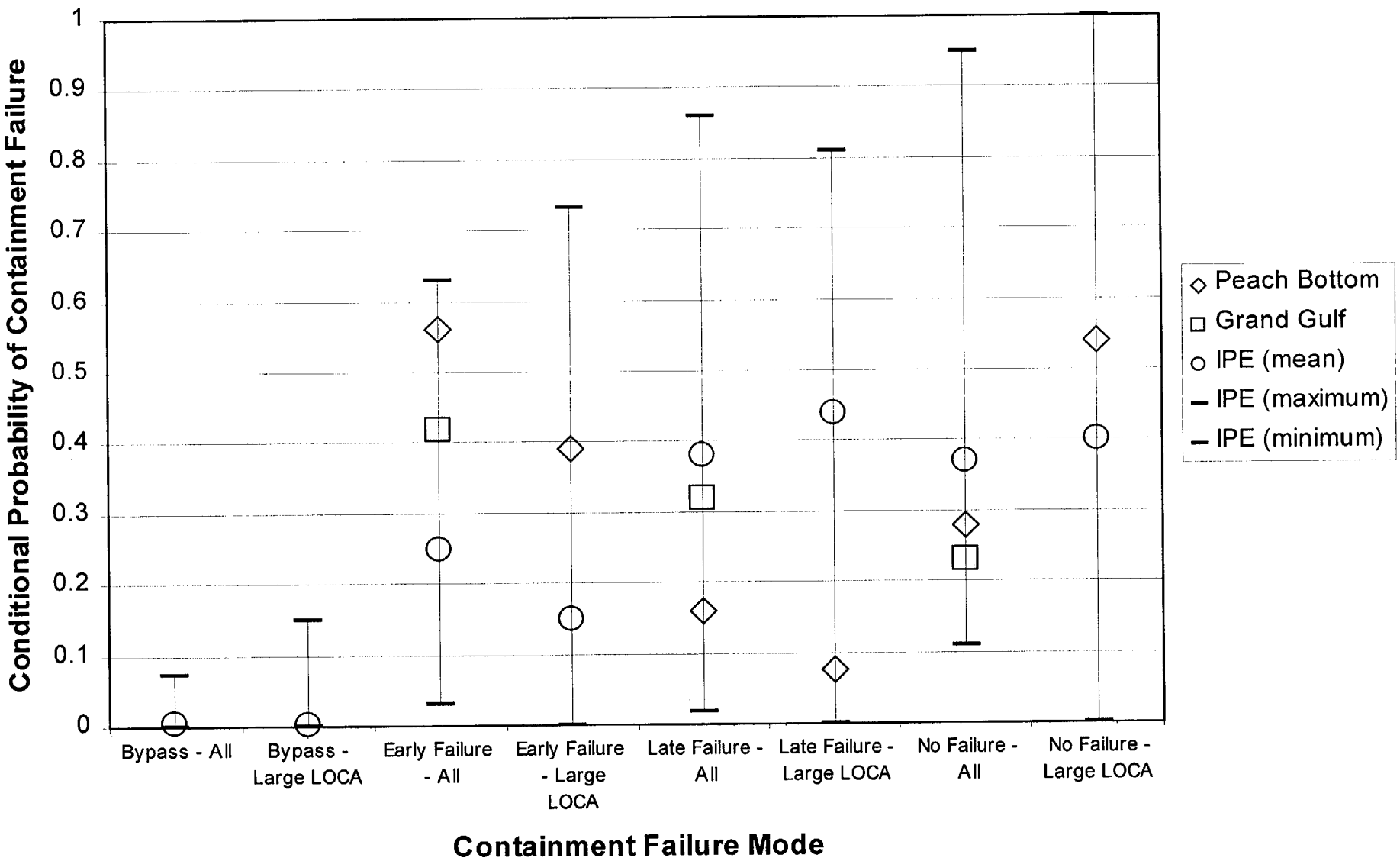
# PROBABILITY OF CONTAINMENT FAILURE FOLLOWING A LOCA

- Containments are designed for large LOCA blowdown loads with considerable margin
- LOCA scenarios can result in beneficial impacts concerning hydrogen production:
  - LOCA with ECCS failure would reduce early in-vessel hydrogen production due to steam deprivation
  - Steam concentrations resulting from blowdown could render large dry containments inert
- Vessel depressurization would preclude high-pressure melt ejection
- However, data indicates the probability of different containment failure modes during LOCAs is not substantially different than during other events

# PWR Conditional Probability of Containment Failure



# BWR Conditional Probability of Containment Failure



# LERF FROM LARGE LOCAs (IPEs)

| IPE Results | Mean Large Early Release Frequency | | Mean Conditional Large Early Release Probability | |
|---|---|---|---|---|
| | All Internal Events | Large LOCAs | All Internal Events | Large LOCAs |
| BWRs | 5E-6/yr | 3E-8/yr | 6E-2 | 1E-2 |
| PWRs | 2E-6/yr | 4E-9/yr | 9E-2 | 3E-2 |

# LEAK-BEFORE-BREAK (LBB)

- Fracture mechanic evaluations documented in NUREG-1061 were performed to help the NRC make decisions on LBB
  - Results show that ratio of the frequencies of DEGBs to leaks in RCS pipes range from 2E-6 to 1E-4
- Beliczey and Schultz correlation used in NUREG/CR-5750 provides a probability of pipe rupture given a through-wall crack ($P_{R:TW}$) as a function of pipe size
  - $P_{R:TW}$ = 1E-2 for 10" pipe
  - $P_{R:TW}$ = 3E-3 for 30" pipe

# PROBABILITY OF A PIPE BREAK LOCA WITH SIMULTANEOUS LOOP

- Evaluation of pipe break LOCA followed by a LOOP was performed to resolve GSI-171 (NUREG/CR-6538). Three reasons for an increase in the likelihood of a LOOP were identified:
  - LOCAs will cause reactor and generator trips and EDG starts. Reactor trip can disturb grid and lead to LOOP.
  - Problems in fast transfer of buses to offsite power resulting from reactor trip can result in loss of power to safety buses requiring the EDGs to start.
  - Addition of ECCS loads following a LOCA can cause an undervoltage trip of buses requiring EDGs to be loaded.
- First two causes can occur subsequent to any reactor trip and third can occur anytime ECCS is actuated.
- LER search was performed for all reactor trips and ECCS actuations leading to a LOOP.

# PROBABILITY OF A PIPE BREAK LOCA WITH SIMULTANEOUS LOOP (cont.)

| Plant Type | Probability of a LOOP given a reactor trip | Probability of a LOOP given an ECCS actuation | Probability of a pipe break LOCA followed by a LOOP[1] | | |
|---|---|---|---|---|---|
| | | | Point Estimate | 5th Percentile | 95th Percentile |
| BWR | 3.7E-03 | 5.6E-02 | 6.0E-02 | 4.5E–03 | 2.5E-01 |
| PWR | 3.9E-03 | 1.0E-02 | 1.4E-02 | 2.7E-03 | 5.5E-02 |
| Total | 3.8E-03 | 1.7E-02 | 2.1E-02 | 5.7E-03 | 6.0E-02 |

1 The point estimate is the sum of the probabilities of a LOOP given a reactor trip and a LOOP given an ECCS actuation.

- These probabilities are two orders of magnitude higher than those used in NUREG-1150 and the IPEs.
- Typically, these studies modeled a LOOP as being independent of the LOCA. As such, the probability of a LOOP during a 24 hour mission time was typically evaluated by multiplying the LOOP initiating event frequency by 24 hours.
- In the NUREG-1150 studies, this resulted in a probability of a LOOP following any reactor trip of 2E-04.

# RISK SIGNIFICANCE OF ECCS

- ECCS is important for preventing core damage for a wide variety of accident types - not just LOCAs
  - Transients, ATWS, SBO, external events, LP/SD events
- Accident sequences involving failure of ECCS results in significant fraction of total CDF for all plants
- ECCS is also important for arresting core damage, removing heat from the containment, and scrubbing fission products and therefore impacts containment performance and the magnitude of radionuclide releases

# CONCLUSIONS FROM RISK INSIGHTS

- Frequency estimates for LBLOCAs preclude their elimination as DBAs
- Frequency estimates for DEGBs in RCS piping suggest they can be eliminated as DBAs
- Frequency of indirect seismic-induced LOCAs may be as large as random LOCAs, direct seismic-induced LOCA frequencies appear to be smaller
- Mean CDFs from LOCAs are <1E-4/yr
- CDFs from LBLOCAs and drain down events during LP/SD and from seismic-induced LBLOCAs can be as significant as from random pipe break LOCAs
- Mean CCDPs for LBLOCAs are typically <1E-2

# CONCLUSIONS FROM RISK INSIGHTS

- Probability of different containment failure modes during LBLOCAs is not substantially different than frequency weighted values for all internal events
- Mean LERFs for LBLOCAs are typically <0.1 and less than frequency weighted values for all internal events
- Data and calculations support LBB concept for large pipes
- Estimates of probability of a LOOP given a LOCA are higher than has been modeled in many PRAs
- Potential for a LOOP coincident with a seismic-induced LOCA is very high
- Even though LOCAs are not always the major contributors to risk, most ECC systems are important for mitigating other risk-significant accidents

# Part 3
# Potential Risk-Informed Options

# TWO OPTIONS FOR POSTULATED SPECTRUM OF BREAKS

- No change
- Apply LBB to ECCS
  - Need to demonstrate frequency of excluded breaks is not a significant fraction of 1E-5/year per framework document
  - Deals with most of perceived unnecessary burden
  - Alternative DBAs for ECCS would be evaluated (e.g. draindown events)
  - Actions should not result in significant increases in seismically induced LOCA frequencies

# TWO OPTIONS FOR POSTULATED SIMULTANEOUS LOOP

- No change
- Relax requirement
  - For example:
    - Unavailability of an electric power system need not be assumed when analyses reviewed and approved by the Commission demonstrate that the resulting set of coincident failure events would have an extremely low probability
  - Could be applied to
    - reduce diesel generator start time or
    - reduce costs in other ways
  - Evaluation feasible in near term

# TWO OPTIONS FOR SINGLE FAILURE CRITERION

- No change

- Modify single-failure criterion
  - For example:
    - Single failures need not be postulated when analyses reviewed and approved by the Commission demonstrate that the resulting set of coincident failure events would have an extremely low probability
    - Failure criterion could be based on frequency
  - Could result in availability of two emergency trains thereby reducing the calculated peak cladding temperature
  - It may be appropriate to defer work on this option until the single-failure criterion can be examined in a broader context

# FOUR OPTIONS FOR ECCS PERFORMANCE MODELS

- No change
- Relax Appendix K conservatisms
  - See earlier presentation on decay heat model
  - Revised models would have to be approved
- Make realistic models less burdensome
  - Both model/method approval and computational costs are high
  - Efficient uncertainty analysis methods exist to reduce computational costs (see Haskin et al, Nucl. Eng. & Design, Vol 166, pp 225-248)
  - Could apply similar methods to automate audit analyses
  - Demonstration and acceptance of improved methods could be time consuming and resource intensive
- Propagate uncertainty in break size
  - Uncertainties in other initial conditions already propagated

# THREE OPTIONS FOR 50.46 REPORTING REQUIRMENTS

- No change
- Relax 50.46 reporting requirements
  - Report only errors or changes that cause peak cladding temperature, local oxidation, or total hydrogen production to fall within specified intervals of the acceptance criteria values
- Eliminate 50.46 reporting requirements
  - Peak cladding temperature is the only core parameter whose calculated value is required to be reported to the NRC
  - Results of PCT calculations would still be available upon request for NRC inspectors

# ALTERNATE OPTIONS BASED ON FRAMEWORK DEFENSE-IN-DEPTH STRATEGIES

- Quantitative objectives are stated for mean values from full-scope PRAs (internal & external events, all modes of operation)
- For the plant
  - Core damage frequency, CDF < $10^{-4}$/year
  - Conditional probability of large early release, CP-LER <0.1
  - Conditional probability of large late release, CP-LLR < 0.1
- For any specific initiator type, e.g., large-break LOCA
  - $CDF_{LBLOCA}$ not a substantial fraction of $10^{-4}$/year
  - $LERF_{LBLOCA}$ not a substantial fraction of $10^{-5}$/year
  - $LLRF_{LBLOCA}$ not a substantial fraction of $10^{-5}$/year
- To eliminate RCS pipe breaks larger than a certain size as design-basis initiators their collective mean frequency should be demonstrably less than $10^{-6}$/year

# ALTERNATE OPTION DEMONSTRATE LOW RISK ASSOCIATED WITH ECCS FAILURE OR INADEQUACY

- Demonstrate core damage frequency for accidents involving ECCS failure or inadequacy is not a substantial fraction of framework guideline (1E-4 per year)
- IF NOT
- Demonstrate LERF and large late release frequencies are not substantial fractions of framework guideline (1E-5 per year)

# ALTERNATE OPTION - DEVELOP PROCESS FOR SELECTING DESIGN-BASIS LOCA INITIATORS AND COINCIDENT FAILURES

- Develop a process for selecting design-basis LOCA initiators and postulated coincident failures based on quantitative estimates of event frequencies and probabilities
- Devise the process so that meeting ECCS acceptance criteria for the selected LOCAs would assure
  - Accidents with LOCA initiators are not substantial contributors to framework core damage and large release frequency guidelines
  - ECCS capacity and reliability are sufficient for other classes of accidents
- Uncertainties in quantitative estimates of event frequencies and probabilities would have to be addressed as part of the process

# OBSERVATIONS

- Scope places emphasis on LOCAs as DBAs for ECCS not containment or equipment qualification
  - Improves safety by not causing unnecessary harsh testing demands on equipment
  - Addresses most of perceived unnecessary burden
  - Allows near-term focus on three regulations (50.46, Appendix K and GDC 35) and key implementing documents
- Risk-informing may require
  - Analysis of potential for failures induced by dynamic effects
  - Fracture-mechanics based estimates of break frequencies
  - Analysis of potential for seismically-induced LOCAs and LOCAs at low power and shutdown

# DISCUSSION ITEMS

# FUTURE ACTIVITIES

- Public workshop, tentatively scheduled for November 8/9, 2000
- ACRS briefings (Nov and Dec, 2000)
- Recommendations to Commission, Dec 2000
- ECCS Acceptance Criteria
- Implications for containment