Letter Report
December 2000

Summary of Information Presented at an NRC-Sponsored Public Workshop on Risk-Informing the Technical Requirements of 10 CFR Part 50 (Option 3), October 2, 2000, Rockville, Maryland

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Abstract

This report summarizes a public workshop that was held on October 2, 2000, in Rockville, Maryland. The workshop was conducted as part of the United States Nuclear Regulatory Commission's (NRC) efforts to explore changes to the body of the 10 CFR Part 50 regulations, to incorporate risk-informed attributes. During the workshop the NRC staff discussed issues and requested feedback from the public on risk-informed revisions to the technical requirements of 10 CFR Part 50.46 and the latest version of the framework for risk-informing changes to technical requirements of 10 CFR Part 50. The staff also provided a brief discussion on their recommendations for a risk-informed 10 CFR 50.44.

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List of Acronyms and Initialisms

ACRS	Advisory Committee on Reactor Safeguards
B&WOG	Babcock & Wilcox Owners' Group
CCFP	Conditional Containment Failure Probability
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
ECCS	Emergency Core Cooling System
GDC	General Design Criterion
LBLOCA	Large Break Loss-of-Coolant Accident
LCO	Limiting Condition for Operation
LERF	Large Early Release Frequency
LOCAs	Loss-of-Coolant Accidents
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PRAs	Probabilistic Risk Assessments
QHOs	Quantitative Health Objectives
SBO	Station Blackout
TAF	Top of Active Fuel
WOG	Westinghouse Owners Group

1 INTRODUCTION

1.1 Background

The Office of Nuclear Regulatory Research of the United States Nuclear Regulatory Commission (NRC) has initiated a program to explore changes to Part 50 of the Code of Federal Regulations (CFR) (i.e., 10 CFR Part 50) to incorporate risk-informed attributes. These changes include: (1) identifying provisions to be added to Part 50 as risk-informed alternatives, (2) revising specific requirements in Part 50 to reflect risk-informed considerations, and (3) deleting unnecessary or ineffective regulations. To support NRC's exploration of risk-informed changes to the technical requirements of Part 50, a public workshop was conducted on October 2, 2000, in Rockville, Maryland. The objective of the workshop was to provide for the exchange of information with all stakeholders regarding:

- the staff's efforts to risk-inform the technical requirements of 10 CFR 50.46–Emergency Core Cooling System (ECCS) Acceptance Criteria, and
- the latest version of the framework for risk-informed changes to the technical requirements of 10 CFR Part 50.

This report summarizes the workshop.

1.2 Workshop Structure

The one-day workshop consisted of presentations by the NRC and representatives of the public and discussions on the presentations. The workshop was well attended and very successful in generating significant feedback from interested parties. Most of the feedback was given verbally during the general discussion session; however, one stakeholder submitted written comments as well. This report summarizes the comments received in both forms.

1.3 Organization of the Report

The intent of this report is to capture the main points of the presentations and comments offered as well as those of the written comments. A verbatim transcript of the workshop was not recorded. This document was prepared based on notes taken during the workshop. Thus, although it is the intent to provide information as presented and discussed, the possibility exists that some points may have been inadvertently omitted or missed.

Chapters 2 and 3 summarize the various presentations. Chapter 4 summarizes information gathered during the question and answer session following each presentation and the open discussion sessions on the major topics, including information from written comments. Appendix A provides the workshop agenda. Appendix B contains the attendance list of those who completed attendance forms; Appendix C, copies of the viewgraphs used by the NRC; and Appendix D, copies of the viewgraphs used by representatives of the public.

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2. NRC PRESENTATION ON RISK-INFORMING THE TECHNICAL REQUIREMENTS IN 10 CFR 50

The workshop opened with introductory material by Mary Drouin (Section Leader in the Probabilistic Risk Analysis Branch, NRC) on the workshop and program objectives; the two phases of Option 3, including a status report on Phase 1 activities; and the workshop agenda. The viewgraphs for this introductory material are provided in Appendix C.

Following this introductory material, Mary Drouin presented nine recommendations for risk-informing 50.44. The viewgraphs containing the nine recommendations are provided in Appendix C.

The next topic covered in the NRC presentation dealt with the framework for risk-informing 10 CFR Part 50 (hereafter referred to as the framework). The presentation is summarized below, and copies of the viewgraphs are provided in Appendix C.

1. The objective of the framework was given. The user of the framework was identified along with its intended application.

2. The structure of the framework was presented using both words and a pictorial representation.

3. The basis for the framework's quantitative guidelines was presented.

4. Implementation and policy issues were identified and discussed. The implementation issues included defense-in-depth and use of Safety Goals. The policy issues included selective implementation and backfit considerations.

5. How the framework addresses uncertainties in the development of risk-informed alternatives was discussed. Particular attention was paid to the use of safety margin.

6. The three major implementation tasks were identified and discussed. These included: selection and prioritization of candidate regulations, development of risk-informed alternative(s) to technical requirements, and evaluation of risk-informed alternative(s).

7. Five topics were identified for additional discussion during the open discussion period following the presentation. The fire topics included: defense-in-depth approach, implementation of safety margin, treatment of uncertainties, selection of numerical values for the quantitative guidelines, and treatment of late containment failures.

The next topic covered in the NRC presentation dealt with the status of risk-informed changes to 10 CFR 50.46. The presentation was divided into three parts: 1) candidate regulatory requirements, 2) risk significance of loss-of-coolant accidents (LOCAs) and ECCS, and 3) potential risk-informed options. The presentation is summarized below, and copies of the viewgraphs are provided in Appendix C.

Candidate Regulatory Requirements

1. The meaning of design basis and loss-of-coolant accidents was discussed.

2. The meaning and design requirements of emergency core cooling were discussed.

3. Performance concerns associated with ECCS were presented.

4. LOCA size categories and NUREG-1150 pipe-break sizes were given.

5. Terminology associated with pipe breaks was discussed.

6. A graphical representation of the methodology used to identify existing LOCA-related regulatory requirements for the four framework strategies was presented.

7. LOCA-related regulations for the four framework strategies were identified. These included regulations to:
   - help prevent initiators,
   - prevent core damage given an initiator,
   - contain radionuclides given core damage, and
   - protect the public given core damage.

8. The evolution of LOCA as a design basis accident was discussed.

9. The emergency core cooling acceptance criteria contained in 10 CFR 50.46 were presented.

10. The requirement that ECCS cooling performance must be calculated with an accepted evaluation model was discussed.

11. The dynamic effects of reactor coolant system pipe breaks were discussed. In addition, issues associated with leak-before-break were presented.

12. Revisions to general design criterion 4 associated with leak-before-break were discussed, along with examples of applications of leak-before-break.

13. Background material associated with applying leak-before-break to ECCS and equipment qualification was discussed. Included in the discussion was an NRC acknowledgment of safety benefits associated with applying leak-before-break to ECCS, the reasons why NRC decided not to apply leak-before-break to ECCS and equipment qualification in 1988, and the 1989 Federal Register Notice statement on the application of leak-before-break to ECCS and equipment qualification.

14. The scope of the initial phase of risk-informing the regulations and the rationale for selecting the regulations were discussed.

15. Potential cost savings associated with modifying the regulations were discussed.

16. The high-level regulatory requirements associated with ECCS performance were presented along with regulations referenced from 50.46 and those regulations referencing 50.46.

17. A list of key implementing documents was provided.

18. Finally, the industry's implementation of the various ECCS performance model requirements was discussed.

Risk Significance of LOCAs and ECCS

1. Various bases for reactor coolant system pipe-break LOCA frequencies were presented.

2. The methods for estimating pipe-break LOCA frequencies contained in NUREG/CR-5750 were identified and briefly discussed. In addition, the through-wall crack data from NUREG/CR-5750 were discussed.

3. Plots depicting the spread in frequencies for various sized LOCAs were presented.

4. The basis for and estimates of double-ended guillotine breaks were presented for various plant types. In addition, seismic-induced LOCA frequency estimates and their bases were discussed.

5. LOCA contributions to core damage frequency (CDF) from full power, shutdown, and seismic analyses were presented.

6. Issues associated with the probability of containment failure following a LOCA were discussed, and estimates of conditional containment failure probability (or large early release frequency) were provided.

7. The information sources used by the NRC to help make its decision on leak-before-break were identified and discussed.

8. Issues associated with the probability of a pipe break LOCA with the simultaneous loss of offsite power were discussed. In addition, numerical results were provided for the probability of a pipe break LOCA followed by a loss of offsite power.

9. High level observations on the risk significance of ECCS were provided.

10. Finally, high level risk insight conclusions associated with pipe break frequencies, core damage and containment failure estimates, and the coincident loss of offsite power given a LOCA were presented.

Potential Risk-Informed Options

1. Various options were identified and discussed for the following high level requirements:
   - spectrum of breaks,
   - simultaneous loss of offsite power,
   - single failure criterion,
   - ECCS performance models, and
   - 50.46 reporting requirements.

2. The following alternate options were identified and discussed:
   - specify quantitative objectives based on the framework's defense-in-depth strategies,
   - a demonstration of low risk associated with ECCS failure or inadequacy, and
   - a process for selecting design-basis LOCA initiators and coincident failures.

3. Observations on the scope and required effort associated with making risk-informed changes to ECC performance requirements were presented.

4. Finally, additional discussion items and future activities were identified and discussed.

The next issue discussed by NRC involved an evaluation of possible revisions to required features of 10 CFR 50 Appendix K. The presentation is summarized below, and copies of the viewgraphs are provided in Appendix C.

1. Background information associated with the NRC program to determine how much Appendix K conservatism is tied up in the decay heat requirement was presented. The discussion identified two options for calculation ECCS performance that involve Appendix K conservatisms, plus one option that allows reduction in non-Appendix K conservatisms.

2. The process used by NRC to evaluate the effect of allowing more realistic models for decay heat and metal water reactions in Appendix K calculations was described. This process included identifying both the reduction in conservatism and the retained conservatism.

3. Summary results from the project were presented.

4. Suggested Appendix K parameter values were provided for use with the 1979 ANS Decay Heat Standard.

5. Finally, preliminary suggestions and observations from the program were identified and discussed.

3. PRESENTATIONS FROM THE PUBLIC

Representatives of the public gave presentations, which are summarized below. Viewgraphs for all presentations except for the Herschel Specter presentation are provided in Appendix D. No viewgraphs were used during the Herschel Specter presentation.

3.1 Risk-Informing the Technical Requirements of 10CFR50 – Performance Technology

Bob Christie of Performance Technology, presented a table summarizing the activities associated with changing the hydrogen control regulation. He then presented a slide indicating that in the 13 months since approving San Onofre Task Zero (i.e., the combustible gas control task) of the Whole Plant Study there had been no change at any nuclear plant to enhance safety and reduce burden with respect to combustible gas control and no progress made on the petition for rulemaking. The slide indicated that exemption request were now being submitted by utilities to enhance safety and reduce burden. Finally, the question of why is this necessary was posed.

3.2 Westinghouse Owners Group LBLOCA Redefinition Program – Westinghouse Owners Group

After an introductory overview slide, Bob Osterrieder of Westinghouse Electric Company continued his presentation by providing background information about the Westinghouse Owners Group's (WOG's) large break loss-of-coolant accident (LBLOCA) redefinition program. He then discussed the program's approach and the safety benefits that would result from redefining the LBLOCA. He presented a flowchart of the program plan along with supporting slides describing various aspects of the plan. Next, he discussed how the LBLOCA plan would continue to maintain safety margin. He then discussed industry support for the LBLOCA program. Finally, he ended with a list of conclusions identifying that the LBLOCA redefinition was the best option to pursue.

3.3 Presentation on Behalf of the B&W Owner's Group – B&W Owner's Group

Bert Dunn of Framatome Technologies opened his presentation by indicating that the B&W Owner's Group (B&WOG) supports the effort to risk inform the LBLOCA regulations, and that B&WOG is pursuing support of the activity as an industry program. Next, he discussed an evaluation of different options/programs for risk informing the regulations, indicating that changing design basis events has the greatest potential benefit. In his last slide he discussed implementation guidance.

3.4 BWR Perspectives – Herschel Specter

Herschel Specter of RBR Consultants, Inc. requested time to make a presentation not originally included in the agenda. The following points summarize the information provided in his presentation. (Note: No viewgraphs were used during the presentation.)

1. Extrapolating available data (with no leak before break considerations), it was determined that in order to get a frequency of less than 1E-6/year, the break size would have to be on the order of 0.35-0.5 ft^2.

2. Using MAAP calculations, the time to reach the top of active fuel (TAF) with no ECCS injection is 244 seconds for a 0.5 ft^2 liquid break, 288 seconds for a 0.35 ft^2 liquid break, and 1025 seconds for a 0.5 ft^2 steam break.

3. This implies that there is much greater time available to start emergency diesel generators than currently designed for (i.e., 10 second required start time). Also, closure of main steam isolation valves in 3-10 seconds can be relaxed (i.e., stretched out). Furthermore, the time by which circuit breakers must close can be stretched out.

4. Some technical specification requirements associated with the large break LOCA are in conflict with performance-based maintenance (i.e., the maintenance rule). Also, under Generic Letter 89-10, the large delta-P's for motor operated valves can be damaging to the valves.

5. Risk-informing the large break LOCA can lead to more risk-informed testing (e.g., more realistic, lower delta-P's), and can increase safety by resulting in less unavailability of equipment due to unrealistic testing conditions.

4. DISCUSSIONS

This section summarizes the discussions that took place after each presentation, presented first, the open discussion that occurred after all presentations were completed, presented next, and the written comments received at the Workshop, presented last.

The summaries in the ensuing three sections include both verbal and written comments and are ***not*** verbatim transcripts of the discussions that took place.

4.1 Summaries of the Discussion After Each Presentation

4.1.1 50.44 Discussion

Public: Why is it necessary for licensees to have to submit exemption requests (regarding combustible gas control) in order to enhance safety rather than having a rule change?

No response recorded.

Public: Items 1 through 8 in SECY-00-0198 are recommendations. What input are you looking for from industry?

NRC: None.

Public: Ice condensers and Mark III's would have to modify hydrogen control for risk significant accident sequences. Do you have a definition for risk significant and will it be in a guidance document?

NRC: That is still to be developed. It will be defined and guidance will be provided in a Regulatory Guide. It will be consistent with Regulatory Guide 1.174 and the framework document.

Public: For Mark I containments, what were the assumptions on venting used in the analysis for risk-informing 50.44? If it was vented, what concentration of H_2 was assumed? In SBO (station blackout), there would be no power. It seems that inerting would be lost during most accident scenarios. Also, what is a large late release; what isotopes and what type of consequences?

NRC Contractor: If you vent, you have early containment failure. Venting is performed under emergency procedures.

Public: Could someone define what a large late release is?

NRC Contractor: "Large" is the same for large early release. One could define it in terms of containment leak rate. With regard to 24 hrs, this is the time period when severe accident management guidelines would likely be in place. The interim period between early and 24 hrs covers the "late" definition.

Public: There is a weak connection between late release and acute fatalities and little impact on latent fatalities. This seems to be inconsistent with use of framework guidelines which are connected to Safety Goals.

NRC: The objective is not with respect to latent fatalities. Large late release will be addressed under framework discussion.

4.1.2 Framework Discussion

NRC Contractor: I want to address Hershel Specter's earlier question concerning large late release. A specific concern exists regarding severe accident scenarios that proceed to core concrete interactions in dry reactor cavities. If containment atmospheric cleanup systems (e.g. containment sprays) are not operable in such scenarios, the containment atmosphere is filled with aerosols, some of which contain relatively nonvolatile radionuclides. Some of the worst source terms in NUREG-1150 (from an offsite dose point of view) arose from such scenarios. Containment failure after vessel breach but while the containment was still filled with aerosols from core concrete interactions would not constitute a large early release. Generally, natural processes such as agglomeration and settling would significantly reduce aerosol concentrations within the 24-hour period set for the large late release in the framework document.

NRC/Contractor: There are additional concerns related to the environment that expanded beyond the Quantitative Health Objectives (QHOs) (e.g., environmental damage).

Public: This is a large expansion in scope. That is why there is the Price-Anderson Act. Based on QHOs, only need to look out to one mile from the plant. Why are no quantitative guidelines provided for the emergency preparedness strategy? What quantitative risk reduction value are you associating with emergency preparedness?

NRC: Need to consider that people beyond one mile from the plant can be impacted by the consequences of an accident.

Public: There appears to be a shift from the safety goals to some fuzzy definition of defense-in-depth. There is no technical basis for the conditional large late release probability of 0.1, except for this fuzzy notion of defense-in-depth. Emphasis should be placed on accident prevention (i.e., elimination of initiators). This has economic benefits.

Public: What if you don't meet the framework's late release guideline, but you have substantial margin with regard to the safety goals (QHOs)?

NRC: Since the program is risk-informed, not risk-based, we need to consider defense-in-depth, not just QHOs and risk insights.

Public: It appears that defense-in-depth can arbitrarily be used to countermand the QHOs, and can be used as justification to never eliminate anything.

NRC: The specific elements of defense-in-depth being considered under the framework are specified clearly, and they are used since we do not want to rely totally on risk insights.

Public: The framework replaces the safety goals with a defense-in-depth strategy. The framework text discussing application of quantitative guidelines references Regulatory Guide 1.174, the revised reactor oversight program, etc., but does not reference the safety goals. Also, the phrase "reasonable assurance of adequate protection of public health and safety" is often repeated elsewhere, but in the framework the words have been changed to "protect public health and safety," so that the NRC does not have to define adequate protection in terms of risk.

NRC: It is maintained that the framework guidelines are based on the QHOs.

Public: There are plants out there that do not meet a CDF of 1E-4 or LERF (large early release frequency) of 1E-5, but the Commission says that they meet the safety goals. The ACRS (Advisory Committee on Reactor Safeguards) has sent a letter confirming that you do not need to meet the CDF and LERF goals in order to meet the safety goals.

Public: In the 10 CFR 50.44 report (p. 6-10), it is stated that the alternate method, which is based on the framework guidelines, is most likely for future plants, and that it is anticipated that current reactors would most likely choose the first method. Therefore, the framework is essentially written for new plants.

NRC: No, this is not true.

Public: The framework document specifies that for risk significant sequences, if one or more strategies cannot be met, the other strategies may be more tightly regulated. This implies that the strategies are not an "OR," rather they are an "AND."

NRC: They are not an "OR" or an "AND."

Public: Years ago, when consideration was first given to conditional containment failure probability (CCFP), it was realized that this resulted in stupid design decisions. For example, for seismic design, it was required to prove that with 90% confidence the core would be on the floor before the containment would fail. Ultimately, it was promised not to do anything stupid.

Now, plants and containments are set, and you can't really do any more to enhance containment performance. However, you can do a lot more to reduce core damage frequency, and the industry has been doing this for the past ten years (e.g., improving procedures, etc.). With this framework, we will eventually get to the point that plants will be making stupid decisions to maintain CCFP.

Regulatory Guide 1.174 decoupled CDF from LERF; but the Option 3 framework re-couples them. Since we can't just go and add another inch to the containment wall, we need to focus on CDF.

The "red-herring" argument at the time CCFP was first being considered was that CCFP requirements were necessary to keep industry from coming forward with a containment-less design, even though there are regulations that require a leak-tight containment. This same red-herring argument is being presented in the framework.

NRC: The framework is not meant to imply that you need to meet the guideline values for all of the strategies. The guidelines are to help the staff determine whether to keep, modify, or eliminate requirements. The guideline values are not hard-fast numbers.

Public: It is not clear how backfit considerations are being accounted for.

NRC: Using 10 CFR 50.44 as an example, since the entire alternative is voluntary, no backfit analysis is required. However, whenever a safety enhancement is proposed (e.g., the igniter issue), it will be sent over to the generic issues program for backfit analysis, to see if it should be applied to all reactors in a mandatory fashion.

For risk-informed alternatives, no backfit analysis will be performed since the alternative is voluntary. However, a value impact analysis will still be performed as part of the regulatory analysis for the rulemaking.

Public: Does this mean that there are no enforcement aspects associated with the voluntary alternatives?

NRC: No, there will be an enforcement aspect related to voluntary alternatives, though we haven't looked at this yet.

4.1.3 Westinghouse Owners' Group 50.46 Discussion

Public: Don't want a prescriptive alternative, i.e., the alterative shouldn't specify that a licensee must use leak-before-break for a specified break size. Rather, the alternative should be written in a manner like, "...licensees must demonstrate adequate performance based on LOCAs up to the size that can be justified by the licensee."

4.1.4 Babcock and Wilcox Owners' Group 50.46 Discussion

Public: Up till now, B&WOG has been following WOG efforts regarding redefining the LBLOCA; however, B&WOG will begin their main activities in this area in 2001.

Public: The B&WOG favors elimination of the LBLOCA as the preferred option for risk-informing 10 CFR 50.46.

Public: Some selective implementation should be allowed in risk-informing 10 CFR 50.46, so that plants do not need to redesign reactor internals based on the lower accident loading.

4.2 Summary of Open Discussion

Public: Can large break LOCAs be eliminated from consideration (i.e., frequency less than 1E-6/yr)?

NRC: Current data does not justify eliminating LOCAs down to a 5 or 6 inch line. Would need to still consider a LOCA larger than 6 inches.

Public: Westinghouse is performing fracture mechanics analyses to justify elimination of breaks down to 6 inches.

Public: If an uncertainty distribution is applied to a group of assemblies, can get decay heat down to zero.

NRC: A one-sided decay heat multiplication factor of 1.2 is used in Appendix K; a factor of 1.1 is used in the suggested Appendix K.

Public: A peak clad temperature difference of 50°F is insignificant for small breaks (it only implies a difference in water level of approximately 3 inches) and should not be used in reportability requirements. It makes sense to use this differential for large breaks, but for small breaks a differential of several hundred degrees is more appropriate.

Are we just dealing with 50.46, or more far reaching?

NRC: If we are considering changes to Appendix K that would remove some of the conservatisms, should attention also be focused on removing known non-conservatisms? Would we be entering backfit space if the wording of 10 CFR 50.46 were changed to include "...errors, changes and non-conservatisms?"

Public: I agree that there needs to be a balanced examination, i.e., looking at both conservatisms and non-conservatisms in Appendix K features.

Public: Before today, I thought that the staff was looking at a broad scope for risk-informing design basis LOCAs, not just the 10 CFR 50.46 impacts.

NRC: Under Option 3, the scope will be much broader than what will be covered in the December Commission paper. Due to time constraints, it was necessary to limit the scope of the design basis LOCA and 10 CFR 50.46 evaluation in order to have something meaningful ready for the December Commission paper.

NRC: Do stakeholders feel that the staff has the proper focus for the initial efforts? Are there specific conservatisms within the current scope that the staff has overlooked?

Public: There is a concern over the staff rushing to develop options over the next couple of months. Written information on the staff's plan for risk-informing 10 CFR 50.46, Appendix K and GDC-35 (General Design Criterion 35) should be provided so that feedback can be given. The public needs sufficient time to consider and think about the proposed options, and provide comments back to the staff.

NRC: It is not envisioned that for the December Commission paper, we would be at the same point as for 10 CFR 50.44. Even though the current evaluation of 10 CFR 50.44 is just a feasibility study, a lot more analysis has been done than will be done by December for 10 CFR 50.46. By December, we intend to provide the Commission with recommendations on options to consider looking at, not recommendations on actual changes, since time is short and more stakeholder feedback is necessary.

NRC: In the WOG presentation, you mentioned that you would not want a prescriptive definition of a large break LOCA. The staff is concerned that the WOG effort could result in a large number of plant-specific limiting break sizes. Does WOG agree?

Public: The WOG is attempting to group plants and come in with a generic analysis for Westinghouse plants. The WOG does not want to see the NRC specify a specific break size that all of the plants would have to meet; rather, let each owner's group justify the break size they feel is appropriate for their plants.

NRC: By redefining the maximum break size, is the WOG envisioning that plants would remove equipment (e.g., accumulators), or eliminate maintenance of equipment?

Public: The WOG is not envisioning that plants would remove equipment since physical removal of equipment cost money. What is expected is that there would be relaxed requirements for equipment. Even if, for example, one accumulator was removed from technical specifications, if it wasn't maintained at all, then the plant would still face LCO (limiting condition for operation) shutdown concerns if one of the remaining accumulators went out of service. Thus, there would be economic incentive to still maintain to some extent the accumulator not covered by technical specifications.

Public: With regards to the December Commission paper, it is premature to go to the Commission with preliminary recommendations until there is a detailed information exchange between the staff and the various owner's groups. Need working level meetings to resolve some issues.

NRC: For the December Commission paper, we currently envision identifying options that we feel should still be pursued versus options that we are no longer considering; i.e., we are not intending to make concrete recommendations similar to what we have done for 10 CFR 50.44.

Public: Hearing today's update on risk-informing 10 CFR 50.46 has been useful, but we need to take the information back and look it over, and then have a detailed dialogue, i.e., a working meeting as opposed to another public workshop.

Public: Important to exchange information with the WOG before making preliminary recommendations to the Commission by November - December time frame. A real working meeting is required.

NRC: I would agree with you. To make the meeting more productive, we should exchange a list of questions/topics before the meeting. I would propose that in December, we just identify options that may be feasible and those which are not. We should not make recommendations. We should ask for X months to examine the identified options.

Public: Agree. Information can be sent to NRC before the meeting.

NRC: The staff would like feedback on the implementation and conclusion slides from the Appendix K presentation.

Public: What procedure should we follow to give feedback to the staff on the Appendix K work in order to get it moving forward? Should it be discussed at the NRC/NEI (Nuclear Energy Institute) senior management meeting?

Public: The biggest thing on licensee's horizon is life extension. There can be tremendous benefit from current risk-informed activities. This benefit can be maximized by having the Option 3 team coordinate with the NRR's life extension teams to identify the most beneficial areas to focus on.

NRC: We agree. This topic has recently been discussed.

Public: Will the risk-informed 10 CFR 50.46 apply to the plants from all vendors? Will it be vender-specific?

NRC: The risk-informed 10 CFR 50.46 would probably not be more vendor-specific than it is now.

NRC: Assuming agreement can be reached on a limiting LOCA size, what consideration should be given to other than pipe-break LOCAs (e.g., draindown events or stuck open safety relief valves)? This needs to be considered during our next meeting.

Public: Stuck open safety relief valves should be considered.

NRC: We are tentatively scheduling another public workshop for the November 8/9, 2000, time-frame. We can still have that meeting, but use a different format, i.e., a more detail-specific working meeting. Is this amenable to the Owners Group?

Public: We will check with owner's group representatives.

NRC: If those dates are not convenient, the following week would also be okay.

Public: How soon should we exchange question lists?

NRC: List of questions should be generated 2 weeks before meeting.

Public: B&WOG can probably have their list ready in the next 2-3 weeks.

Public: Do the values presented in the table on core damage frequencies at shutdown account for the fraction of time at shutdown?

NRC: The numbers in the table are pro-rated; all numbers are converted to a per calendar-year basis. There are times (modes) during shutdown when the instantaneous risk can be a lot higher than during full-power operation.

Public: At the Indian Point hearings, incremental risk from seismic events was not that great, since a seismic event of the magnitude necessary for significant offsite consequences would probably kill off most of the surrounding population anyway.

NRC: For the detailed public working meeting, is one day sufficient, or should we plan for two days?

Public: We will probably spend most of the day discussing PRAs (probabilistic risk assessments) and numbers. With other topics included, we may need more than one day.

NRC: It is more difficult to block out two days.

Public: Probably should block out two days, just in case.

4.3 Additional Public Comments

This section presents the one written public comment that was received.

> The Qualitative Guidelines are (loosely) based on the Safety Goals because they parse the probabilities of the QHOs. In February, the earlier workshop noted that the product across the rows was 10^{-6}, slightly above the early fatality goal. Nevertheless, that was OK because there would be conservatisms in each of the columns. Since then, we have dropped 10^{-1} for

limiting public health effects (column 4) and changed the overall product to 10^{-5} (a subtle but transparent change). The problem is that the conservatisms remain. Concerns expressed from the audience can be summarized as that NRC will concentrate its focus on the values in each column (e.g., 10^{-4}). This is likely to result in decisions that are NOT consistent with the safety goals. If the framework is applied in that manner, then indeed it is not consistent with the safety goals.

The concept that changes may reduce the initiating frequency enough to change rows is an intriguing way to avoid this trap. It is subject to at least two problems:

1. NRC–and its contractors–need to focus on factors that would reduce initiating frequency, since this is an NRC tool. Historically, the industry has not seen that kind of focus on initiating frequencies from NRC.

2. The NRC must be willing to accept that some accident sequences historically treated can become "Rare" events, and to defend that change from criticisms that will undoubtedly arise.

Skepticism that these two problems will prevent application of the framework from being overly conservative underlies the concerns heard at the workshop.

APPENDIX A. WORKSHOP AGENDA

Workshop Agenda

Monday, October 2, 2000

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 (Continued) WOG Presentation B&W OG Presentation Open Discussion
2:40 pm - 3:00 pm	Break
3:00 pm - 3:30 pm	Discussion on 50.46 (Continued) Open Discussion
3:30 pm - 4:00 pm	WRAPUP

APPENDIX B. WORKSHOP REGISTRATION LIST

LIST OF ATTENDEES

Table B.1 contains a list of the workshop attendees, based on the registration forms that were returned and from personal recollection of who attended. Since a large number of attendees did not submit their registration forms, this list is not complete. It is believed that the workshop was attended by approximately 30 people in addition to those listed below.

Table B.1 Workshop Registration

Name	Affiliation
John Barry	Westinghouse
Allen Camp	Sandia National Laboratories
Nancy Chapman	SERCH/Bechtel
Mike Cheok	Nuclear Regulatory Commission (NRR/DSSA)
Bob Christie	Performance Technology
Dr. Hugo C. Da Silva	TXU Electric
Mary Drouin	Nuclear Regulatory Commission (RES/DRAA/PRAB)
Bert Dunn	B&W OG, FTI
Răducu Gheorghe	Canadian Nuclear Safety Commission
Don J. Green	Tennessee Valley Authority
Eric Haskin	ERI Consulting
Ken Heck	Nuclear Regulatory Commission (NRR/DIPM/IQMB)
Adrian Heymer	Nuclear Energy Institute
Robert Hill	Carolina Power & Light
Roger Huston	Licensing Support Services
Tom King	Nuclear Regulatory Commission (DRAA)
Alan Kuritzky	Nuclear Regulatory Commission (RES/DRAA/PRAB)
Jeffrey L. LaChance	Sandia National Laboratories
John Lehner	Brookhaven National Laboratory
Stewart Magruder	Nuclear Regulatory Commission (NRR/DIPM/IQMB)
Eileen McKenna	Nuclear Regulatory Commission (NRR/DRIP/RGEB)
Gary D. Miller	Dominion Generation (Virginia Power)
Vinod Mubayi	Brookhaven National Laboratory
Paige T. Negus	GE Nuclear Energy
Mitch Nissley	Westinghouse Electric Company

Table B.1 Workshop Registration

Name	Affiliation
Bob Osterrieder	Westinghouse/WOG
Sid Powell	Florida Power Corporation / Crystal River-3
Trevor Pratt	Brookhaven National Laboratory
Terrance A. Riech	Commonwealth Edison
Zoltan R. Rosztoczy	Zeetech, Inc.
Glen E. Schinzel	South Texas Project N.O.C.
Mike Snodderly	Nuclear Regulatory Commission (NRR/DSSA/SPSB)
Herschel Specter	RBR Consultants, Inc.
Gary Vine	Electric Power Research Institute
Charles Willbanks	NUS Information Services

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APPENDIX C. NRC PRESENTATION MATERIAL

Letter Report
December 2000

Summary of Information Presented at an NRC-Sponsored Public Workshop on Risk-Informing the Technical Requirements of 10 CFR Part 50 (Option 3), October 2, 2000, Rockville, Maryland

Donnie W. Whitehead (Editor)

Prepared by
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a Lockheed Martin Company, for the United States Department of
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Letter Report
December 5, 2000

Summary of Information Presented at an NRC-Sponsored Public Workshop on Risk-Informing the Technical Requirements of 10 CFR Part 50 (Option 3), November 2, 2000 Rockville, Maryland

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Abstract

This report summarizes a public workshop that was held on October 2, 2000, in Rockville, Maryland. The workshop was conducted as part of the United States Nuclear Regulatory Commission's (NRC) efforts to explore changes to the body of the 10 CFR Part 50 regulations, to incorporate risk-informed attributes. During the workshop the NRC staff discussed issues and requested feedback from the public on risk-informed revisions to the technical requirements of 10 CFR Part 50.46 and the latest version of the framework for risk-informing changes to technical requirements of 10 CFR Part 50. The staff also provided a brief discussion on their recommendations for a risk-informed 10 CFR 50.44.

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The editor thanks the following for notes taken of important points made during the 10 CFR Part 50 Workshop:

Jeff LaChance, Sandia National Laboratories
Eric Haskin, ERI Consulting
Alan Kuritzky, Nuclear Regulatory Commission

CONTENTS

List of Acronyms and Initialisms

ACRS	Advisory Committee on Reactor Safeguards
B&WOG	Babcock & Wilcox Owners' Group
CCFP	Conditional Containment Failure Probability
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
ECCS	Emergency Core Cooling System
GDC	General Design Criterion
LBLOCA	Large Break Loss-of-Coolant Accident
LCO	Limiting Condition for Operation
LERF	Large Early Release Frequency
LOCAs	Loss-of-Coolant Accidents
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PRAs	Probabilistic Risk Assessments
QHOs	Quantitative Health Objectives
SBO	Station Blackout
TAF	Top of Active Fuel
WOG	Westinghouse Owners Group

1 INTRODUCTION

1.1 Background

The Office of Nuclear Regulatory Research of the United States Nuclear Regulatory Commission (NRC) has initiated a program to explore changes to Part 50 of the Code of Federal Regulations (CFR) (i.e., 10 CFR Part 50) to incorporate risk-informed attributes. These changes include: (1) identifying provisions to be added to Part 50 as risk-informed alternatives, (2) revising specific requirements in Part 50 to reflect risk-informed considerations, and (3) deleting unnecessary or ineffective regulations. To support NRC's exploration of risk-informed changes to the technical requirements of Part 50, a public workshop was conducted on October 2, 2000, in Rockville, Maryland. The objective of the workshop was to provide for the exchange of information with all stakeholders regarding:

- the staff's efforts to risk-inform the technical requirements of 10 CFR 50.46–Emergency Core Cooling System (ECCS) Acceptance Criteria, and
- the latest version of the framework for risk-informed changes to the technical requirements of 10 CFR Part 50.

This report summarizes the workshop.

1.2 Workshop Structure

The one-day workshop consisted of presentations by the NRC and representatives of the public and discussions on the presentations. The workshop was well attended and very successful in generating significant feedback from interested parties. Most of the feedback was given verbally during the general discussion session; however, one stakeholder submitted written comments as well. This report summarizes the comments received in both forms.

1.3 Organization of the Report

The intent of this report is to capture the main points of the presentations and comments offered as well as those of the written comments. A verbatim transcript of the workshop was not recorded. This document was prepared based on notes taken during the workshop. Thus, although it is the intent to provide information as presented and discussed, the possibility exists that some points may have been inadvertently omitted or missed.

Chapters 2 and 3 summarize the various presentations. Chapter 4 summarizes information gathered during the question and answer session following each presentation and the open discussion sessions on the major topics, including information from written comments. Appendix A provides the workshop agenda. Appendix B contains the attendance list of those who completed attendance forms; Appendix C, copies of the viewgraphs used by the NRC; and Appendix D, copies of the viewgraphs used by representatives of the public.

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2. NRC PRESENTATION ON RISK-INFORMING THE TECHNICAL REQUIREMENTS IN 10 CFR 50

The workshop opened with introductory material by Mary Drouin (Section Leader in the Probabilistic Risk Analysis Branch, NRC) on the workshop and program objectives; the two phases of Option 3, including a status report on Phase 1 activities; and the workshop agenda. The viewgraphs for this introductory material are provided in Appendix C.

Following this introductory material, Mary Drouin presented nine recommendations for risk-informing 50.44. The viewgraphs containing the nine recommendations are provided in Appendix C.

The next topic covered in the NRC presentation dealt with the framework for risk-informing 10 CFR Part 50 (hereafter referred to as the framework). The presentation is summarized below, and copies of the viewgraphs are provided in Appendix C.

1. The objective of the framework was given. The user of the framework was identified along with its intended application.

2. The structure of the framework was presented using both words and a pictorial representation.

3. The basis for the framework's quantitative guidelines was presented.

4. Implementation and policy issues were identified and discussed. The implementation issues included defense-in-depth and use of Safety Goals. The policy issues included selective implementation and backfit considerations.

5. How the framework addresses uncertainties in the development of risk-informed alternatives was discussed. Particular attention was paid to the use of safety margin.

6. The three major implementation tasks were identified and discussed. These included: selection and prioritization of candidate regulations, development of risk-informed alternative(s) to technical requirements, and evaluation of risk-informed alternative(s).

7. Five topics were identified for additional discussion during the open discussion period following the presentation. The fire topics included: defense-in-depth approach, implementation of safety margin, treatment of uncertainties, selection of numerical values for the quantitative guidelines, and treatment of late containment failures.

The next topic covered in the NRC presentation dealt with the status of risk-informed changes to 10 CFR 50.46. The presentation was divided into three parts: 1) candidate regulatory requirements, 2) risk significance of loss-of-coolant accidents (LOCAs) and ECCS, and 3) potential risk-informed options. The presentation is summarized below, and copies of the viewgraphs are provided in Appendix C.

Candidate Regulatory Requirements

1. The meaning of design basis and loss-of-coolant accidents was discussed.

2. The meaning and design requirements of emergency core cooling were discussed.

3. Performance concerns associated with ECCS were presented.

4. LOCA size categories and NUREG-1150 pipe-break sizes were given.

5. Terminology associated with pipe breaks was discussed.

6. A graphical representation of the methodology used to identify existing LOCA-related regulatory requirements for the four framework strategies was presented.

7. LOCA-related regulations for the four framework strategies were identified. These included regulations to:
   - help prevent initiators,
   - prevent core damage given an initiator,
   - contain radionuclides given core damage, and
   - protect the public given core damage.

8. The evolution of LOCA as a design basis accident was discussed.

9. The emergency core cooling acceptance criteria contained in 10 CFR 50.46 were presented.

10. The requirement that ECCS cooling performance must be calculated with an accepted evaluation model was discussed.

11. The dynamic effects of reactor coolant system pipe breaks were discussed. In addition, issues associated with leak-before-break were presented.

12. Revisions to general design criterion 4 associated with leak-before-break were discussed, along with examples of applications of leak-before-break.

13. Background material associated with applying leak-before-break to ECCS and equipment qualification was discussed. Included in the discussion was an NRC acknowledgment of safety benefits associated with applying leak-before-break to ECCS, the reasons why NRC decided not to apply leak-before-break to ECCS and equipment qualification in 1988, and the 1989 Federal Register Notice statement on the application of leak-before-break to ECCS and equipment qualification.

14. The scope of the initial phase of risk-informing the regulations and the rationale for selecting the regulations were discussed.

15. Potential cost savings associated with modifying the regulations were discussed.

16. The high-level regulatory requirements associated with ECCS performance were presented along with regulations referenced from 50.46 and those regulations referencing 50.46.

17. A list of key implementing documents was provided.

18. Finally, the industry's implementation of the various ECCS performance model requirements was discussed.

Risk Significance of LOCAs and ECCS

1. Various bases for reactor coolant system pipe-break LOCA frequencies were presented.

2. The methods for estimating pipe-break LOCA frequencies contained in NUREG/CR-5750 were identified and briefly discussed. In addition, the through-wall crack data from NUREG/CR-5750 were discussed.

3. Plots depicting the spread in frequencies for various sized LOCAs were presented.

4. The basis for and estimates of double-ended guillotine breaks were presented for various plant types. In addition, seismic-induced LOCA frequency estimates and their bases were discussed.

5. LOCA contributions to core damage frequency (CDF) from full power, shutdown, and seismic analyses were presented.

6. Issues associated with the probability of containment failure following a LOCA were discussed, and estimates of conditional containment failure probability (or large early release frequency) were provided.

7. The information sources used by the NRC to help make its decision on leak-before-break were identified and discussed.

8. Issues associated with the probability of a pipe break LOCA with the simultaneous loss of offsite power were discussed. In addition, numerical results were provided for the probability of a pipe break LOCA followed by a loss of offsite power.

9. High level observations on the risk significance of ECCS were provided.

10. Finally, high level risk insight conclusions associated with pipe break frequencies, core damage and containment failure estimates, and the coincident loss of offsite power given a LOCA were presented.

Potential Risk-Informed Options

1. Various options were identified and discussed for the following high level requirements:
   - spectrum of breaks,
   - simultaneous loss of offsite power,
   - single failure criterion,
   - ECCS performance models, and
   - 50.46 reporting requirements.

2. The following alternate options were identified and discussed:
   - specify quantitative objectives based on the framework’s defense-in-depth strategies,
   - a demonstration of low risk associated with ECCS failure or inadequacy, and
   - a process for selecting design-basis LOCA initiators and coincident failures.

3. Observations on the scope and required effort associated with making risk-informed changes to ECC performance requirements were presented.

4. Finally, additional discussion items and future activities were identified and discussed.

The next issue discussed by NRC involved an evaluation of possible revisions to required features of 10 CFR 50 Appendix K. The presentation is summarized below, and copies of the viewgraphs are provided in Appendix C.

1. Background information associated with the NRC program to determine how much Appendix K conservatism is tied up in the decay heat requirement was presented. The discussion identified two options for calculation ECCS performance that involve Appendix K conservatisms, plus one option that allows reduction in non-Appendix K conservatisms.

2. The process used by NRC to evaluate the effect of allowing more realistic models for decay heat and metal water reactions in Appendix K calculations was described. This process included identifying both the reduction in conservatism and the retained conservatism.

3. Summary results from the project were presented.

4. Suggested Appendix K parameter values were provided for use with the 1979 ANS Decay Heat Standard.

5. Finally, preliminary suggestions and observations from the program were identified and discussed.

3. PRESENTATIONS FROM THE PUBLIC

Representatives of the public gave presentations, which are summarized below. Viewgraphs for all presentations except for the Herschel Specter presentation are provided in Appendix D. No viewgraphs were used during the Herschel Specter presentation.

3.1 Risk-Informing the Technical Requirements of 10CFR50 – Performance Technology

Bob Christie of Performance Technology, presented a table summarizing the activities associated with changing the hydrogen control regulation. He then presented a slide indicating that in the 13 months since approving San Onofre Task Zero (i.e., the combustible gas control task) of the Whole Plant Study there had been no change at any nuclear plant to enhance safety and reduce burden with respect to combustible gas control and no progress made on the petition for rulemaking. The slide indicated that exemption request were now being submitted by utilities to enhance safety and reduce burden. Finally, the question of why is this necessary was posed.

3.2 Westinghouse Owners Group LBLOCA Redefinition Program – Westinghouse Owners Group

After an introductory overview slide, Bob Osterrieder of Westinghouse Electric Company continued his presentation by providing background information about the Westinghouse Owners Group's (WOG's) large break loss-of-coolant accident (LBLOCA) redefinition program. He then discussed the program's approach and the safety benefits that would result from redefining the LBLOCA. He presented a flowchart of the program plan along with supporting slides describing various aspects of the plan. Next, he discussed how the LBLOCA plan would continue to maintain safety margin. He then discussed industry support for the LBLOCA program. Finally, he ended with a list of conclusions identifying that the LBLOCA redefinition was the best option to pursue.

3.3 Presentation on Behalf of the B&W Owner's Group – B&W Owner's Group

Bert Dunn of Framatome Technologies opened his presentation by indicating that the B&W Owner's Group (B&WOG) supports the effort to risk inform the LBLOCA regulations, and that B&WOG is pursuing support of the activity as an industry program. Next, he discussed an evaluation of different options/programs for risk informing the regulations, indicating that changing design basis events has the greatest potential benefit. In his last slide he discussed implementation guidance.

3.4 BWR Perspectives – Herschel Specter

Herschel Specter of RBR Consultants, Inc. requested time to make a presentation not originally included in the agenda. The following points summarize the information provided in his presentation. (Note: No viewgraphs were used during the presentation.)

1. Extrapolating available data (with no leak before break considerations), it was determined that in order to get a frequency of less than 1E-6/year, the break size would have to be on the order of 0.35-0.5 ft^2.

2. Using MAAP calculations, the time to reach the top of active fuel (TAF) with no ECCS injection is 244 seconds for a 0.5 ft^2 liquid break, 288 seconds for a 0.35 ft^2 liquid break, and 1025 seconds for a 0.5 ft^2 steam break.

3. This implies that there is much greater time available to start emergency diesel generators than currently designed for (i.e., 10 second required start time). Also, closure of main steam isolation valves in 3-10 seconds can be relaxed (i.e., stretched out). Furthermore, the time by which circuit breakers must close can be stretched out.

4. Some technical specification requirements associated with the large break LOCA are in conflict with performance-based maintenance (i.e., the maintenance rule). Also, under Generic Letter 89-10, the large delta-P's for motor operated valves can be damaging to the valves.

5. Risk-informing the large break LOCA can lead to more risk-informed testing (e.g., more realistic, lower delta-P's), and can increase safety by resulting in less unavailability of equipment due to unrealistic testing conditions.

4. DISCUSSIONS

This section summarizes the discussions that took place after each presentation, presented first, the open discussion that occurred after all presentations were completed, presented next, and the written comments received at the Workshop, presented last.

The summaries in the ensuing three sections include both verbal and written comments and are ***not*** verbatim transcripts of the discussions that took place.

4.1 Summaries of the Discussion After Each Presentation

4.1.1 50.44 Discussion

Public: Why is it necessary for licensees to have to submit exemption requests (regarding combustible gas control) in order to enhance safety rather than having a rule change?

No response recorded.

Public: Items 1 through 8 in SECY-00-0198 are recommendations. What input are you looking for from industry?

NRC: None.

Public: Ice condensers and Mark III's would have to modify hydrogen control for risk significant accident sequences. Do you have a definition for risk significant and will it be in a guidance document?

NRC: That is still to be developed. It will be defined and guidance will be provided in a Regulatory Guide. It will be consistent with Regulatory Guide 1.174 and the framework document.

Public: For Mark I containments, what were the assumptions on venting used in the analysis for risk-informing 50.44? If it was vented, what concentration of H_2 was assumed? In SBO (station blackout), there would be no power. It seems that inerting would be lost during most accident scenarios. Also, what is a large late release; what isotopes and what type of consequences?

NRC Contractor: If you vent, you have early containment failure. Venting is performed under emergency procedures.

Public: Could someone define what a large late release is?

NRC Contractor: "Large" is the same for large early release. One could define it in terms of containment leak rate. With regard to 24 hrs, this is the time period when severe accident management guidelines would likely be in place. The interim period between early and 24 hrs covers the "late" definition.

Public: There is a weak connection between late release and acute fatalities and little impact on latent fatalities. This seems to be inconsistent with use of framework guidelines which are connected to Safety Goals.

NRC: The objective is not with respect to latent fatalities. Large late release will be addressed under framework discussion.

4.1.2 Framework Discussion

NRC Contractor: I want to address Hershel Specter's earlier question concerning large late release. A specific concern exists regarding severe accident scenarios that proceed to core concrete interactions in dry reactor cavities. If containment atmospheric cleanup systems (e.g. containment sprays) are not operable in such scenarios, the containment atmosphere is filled with aerosols, some of which contain relatively nonvolatile radionuclides. Some of the worst source terms in NUREG-1150 (from an offsite dose point of view) arose from such scenarios. Containment failure after vessel breach but while the containment was still filled with aerosols from core concrete interactions would not constitute a large early release. Generally, natural processes such as agglomeration and settling would significantly reduce aerosol concentrations within the 24-hour period set for the large late release in the framework document.

NRC/Contractor: There are additional concerns related to the environment that expanded beyond the Quantitative Health Objectives (QHOs) (e.g., environmental damage).

Public: This is a large expansion in scope. That is why there is the Price-Anderson Act. Based on QHOs, only need to look out to one mile from the plant. Why are no quantitative guidelines provided for the emergency preparedness strategy? What quantitative risk reduction value are you associating with emergency preparedness?

NRC: Need to consider that people beyond one mile from the plant can be impacted by the consequences of an accident.

Public: There appears to be a shift from the safety goals to some fuzzy definition of defense-in-depth. There is no technical basis for the conditional large late release probability of 0.1, except for this fuzzy notion of defense-in-depth. Emphasis should be placed on accident prevention (i.e., elimination of initiators). This has economic benefits.

Public: What if you don't meet the framework's late release guideline, but you have substantial margin with regard to the safety goals (QHOs)?

NRC: Since the program is risk-informed, not risk-based, we need to consider defense-in-depth, not just QHOs and risk insights.

Public: It appears that defense-in-depth can arbitrarily be used to countermand the QHOs, and can be used as justification to never eliminate anything.

NRC: The specific elements of defense-in-depth being considered under the framework are specified clearly, and they are used since we do not want to rely totally on risk insights.

Public: The framework replaces the safety goals with a defense-in-depth strategy. The framework text discussing application of quantitative guidelines references Regulatory Guide 1.174, the revised reactor oversight program, etc., but does not reference the safety goals. Also, the phrase "reasonable assurance of adequate protection of public health and safety" is often repeated elsewhere, but in the framework the words have been changed to "protect public health and safety," so that the NRC does not have to define adequate protection in terms of risk.

NRC: It is maintained that the framework guidelines are based on the QHOs.

Public: There are plants out there that do not meet a CDF of 1E-4 or LERF (large early release frequency) of 1E-5, but the Commission says that they meet the safety goals. The ACRS (Advisory Committee on Reactor Safeguards) has sent a letter confirming that you do not need to meet the CDF and LERF goals in order to meet the safety goals.

Public: In the 10 CFR 50.44 report (p. 6-10), it is stated that the alternate method, which is based on the framework guidelines, is most likely for future plants, and that it is anticipated that current reactors would most likely choose the first method. Therefore, the framework is essentially written for new plants.

NRC: No, this is not true.

Public: The framework document specifies that for risk significant sequences, if one or more strategies cannot be met, the other strategies may be more tightly regulated. This implies that the strategies are not an "OR," rather they are an "AND."

NRC: They are not an "OR" or an "AND."

Public: Years ago, when consideration was first given to conditional containment failure probability (CCFP), it was realized that this resulted in stupid design decisions. For example, for seismic design, it was required to prove that with 90% confidence the core would be on the floor before the containment would fail. Ultimately, it was promised not to do anything stupid.

Now, plants and containments are set, and you can’t really do any more to enhance containment performance. However, you can do a lot more to reduce core damage frequency, and the industry has been doing this for the past ten years (e.g., improving procedures, etc.). With this framework, we will eventually get to the point that plants will be making stupid decisions to maintain CCFP.

Regulatory Guide 1.174 decoupled CDF from LERF; but the Option 3 framework re-couples them. Since we can’t just go and add another inch to the containment wall, we need to focus on CDF.

The "red-herring" argument at the time CCFP was first being considered was that CCFP requirements were necessary to keep industry from coming forward with a containment-less design, even though there are regulations that require a leak-tight containment. This same red-herring argument is being presented in the framework.

NRC: The framework is not meant to imply that you need to meet the guideline values for all of the strategies. The guidelines are to help the staff determine whether to keep, modify, or eliminate requirements. The guideline values are not hard-fast numbers.

Public: It is not clear how backfit considerations are being accounted for.

NRC: Using 10 CFR 50.44 as an example, since the entire alternative is voluntary, no backfit analysis is required. However, whenever a safety enhancement is proposed (e.g., the igniter issue), it will be sent over to the generic issues program for backfit analysis, to see if it should be applied to all reactors in a mandatory fashion.

For risk-informed alternatives, no backfit analysis will be performed since the alternative is voluntary. However, a value impact analysis will still be performed as part of the regulatory analysis for the rulemaking.

Public: Does this mean that there are no enforcement aspects associated with the voluntary alternatives?

NRC: No, there will be an enforcement aspect related to voluntary alternatives, though we haven't looked at this yet.

4.1.3 Westinghouse Owners' Group 50.46 Discussion

Public: Don't want a prescriptive alternative, i.e., the alterative shouldn't specify that a licensee must use leak-before-break for a specified break size. Rather, the alternative should be written in a manner like, "...licensees must demonstrate adequate performance based on LOCAs up to the size that can be justified by the licensee."

4.1.4 Babcock and Wilcox Owners' Group 50.46 Discussion

Public: Up till now, B&WOG has been following WOG efforts regarding redefining the LBLOCA; however, B&WOG will begin their main activities in this area in 2001.

Public: The B&WOG favors elimination of the LBLOCA as the preferred option for risk-informing 10 CFR 50.46.

Public: Some selective implementation should be allowed in risk-informing 10 CFR 50.46, so that plants do not need to redesign reactor internals based on the lower accident loading.

4.2 Summary of Open Discussion

Public: Can large break LOCAs be eliminated from consideration (i.e., frequency less than 1E-6/yr)?

NRC: Current data does not justify eliminating LOCAs down to a 5 or 6 inch line. Would need to still consider a LOCA larger than 6 inches.

Public: Westinghouse is performing fracture mechanics analyses to justify elimination of breaks down to 6 inches.

Public: If an uncertainty distribution is applied to a group of assemblies, can get decay heat down to zero.

NRC: A one-sided decay heat multiplication factor of 1.2 is used in Appendix K; a factor of 1.1 is used in the suggested Appendix K.

Public: A peak clad temperature difference of 50^{o}F is insignificant for small breaks (it only implies a difference in water level of approximately 3 inches) and should not be used in reportability requirements. It makes sense to use this differential for large breaks, but for small breaks a differential of several hundred degrees is more appropriate.

Are we just dealing with 50.46, or more far reaching?

NRC: If we are considering changes to Appendix K that would remove some of the conservatisms, should attention also be focused on removing known non-conservatisms? Would we be entering backfit space if the wording of 10 CFR 50.46 were changed to include "...errors, changes and non-conservatisms?"

Public: I agree that there needs to be a balanced examination, i.e., looking at both conservatisms and non-conservatisms in Appendix K features.

Public: Before today, I thought that the staff was looking at a broad scope for risk-informing design basis LOCAs, not just the 10 CFR 50.46 impacts.

NRC: Under Option 3, the scope will be much broader than what will be covered in the December Commission paper. Due to time constraints, it was necessary to limit the scope of the design basis LOCA and 10 CFR 50.46 evaluation in order to have something meaningful ready for the December Commission paper.

NRC: Do stakeholders feel that the staff has the proper focus for the initial efforts? Are there specific conservatisms within the current scope that the staff has overlooked?

Public: There is a concern over the staff rushing to develop options over the next couple of months. Written information on the staff's plan for risk-informing 10 CFR 50.46, Appendix K and GDC-35 (General Design Criterion 35) should be provided so that feedback can be given. The public needs sufficient time to consider and think about the proposed options, and provide comments back to the staff.

NRC: It is not envisioned that for the December Commission paper, we would be at the same point as for 10 CFR 50.44. Even though the current evaluation of 10 CFR 50.44 is just a feasibility study, a lot more analysis has been done than will be done by December for 10 CFR 50.46. By December, we intend to provide the Commission with recommendations on options to consider looking at, not recommendations on actual changes, since time is short and more stakeholder feedback is necessary.

NRC: In the WOG presentation, you mentioned that you would not want a prescriptive definition of a large break LOCA. The staff is concerned that the WOG effort could result in a large number of plant-specific limiting break sizes. Does WOG agree?

Public: The WOG is attempting to group plants and come in with a generic analysis for Westinghouse plants. The WOG does not want to see the NRC specify a specific break size that all of the plants would have to meet; rather, let each owner's group justify the break size they feel is appropriate for their plants.

NRC: By redefining the maximum break size, is the WOG envisioning that plants would remove equipment (e.g., accumulators), or eliminate maintenance of equipment?

Public: The WOG is not envisioning that plants would remove equipment since physical removal of equipment cost money. What is expected is that there would be relaxed requirements for equipment. Even if, for example, one accumulator was removed from technical specifications, if it wasn't maintained at all, then the plant would still face LCO (limiting condition for operation) shutdown concerns if one of the remaining accumulators went out of service. Thus, there would be economic incentive to still maintain to some extent the accumulator not covered by technical specifications.

Public: With regards to the December Commission paper, it is premature to go to the Commission with preliminary recommendations until there is a detailed information exchange between the staff and the various owner's groups. Need working level meetings to resolve some issues.

NRC: For the December Commission paper, we currently envision identifying options that we feel should still be pursued versus options that we are no longer considering; i.e., we are not intending to make concrete recommendations similar to what we have done for 10 CFR 50.44.

Public: Hearing today's update on risk-informing 10 CFR 50.46 has been useful, but we need to take the information back and look it over, and then have a detailed dialogue, i.e., a working meeting as opposed to another public workshop.

Public: Important to exchange information with the WOG before making preliminary recommendations to the Commission by November - December time frame. A real working meeting is required.

NRC: I would agree with you. To make the meeting more productive, we should exchange a list of questions/topics before the meeting. I would propose that in December, we just identify options that may be feasible and those which are not. We should not make recommendations. We should ask for X months to examine the identified options.

Public: Agree. Information can be sent to NRC before the meeting.

NRC: The staff would like feedback on the implementation and conclusion slides from the Appendix K presentation.

Public: What procedure should we follow to give feedback to the staff on the Appendix K work in order to get it moving forward? Should it be discussed at the NRC/NEI (Nuclear Energy Institute) senior management meeting?

Public: The biggest thing on licensee's horizon is life extension. There can be tremendous benefit from current risk-informed activities. This benefit can be maximized by having the Option 3 team coordinate with the NRR's life extension teams to identify the most beneficial areas to focus on.

NRC: We agree. This topic has recently been discussed.

Public: Will the risk-informed 10 CFR 50.46 apply to the plants from all vendors? Will it be vender-specific?

NRC: The risk-informed 10 CFR 50.46 would probably not be more vendor-specific than it is now.

NRC: Assuming agreement can be reached on a limiting LOCA size, what consideration should be given to other than pipe-break LOCAs (e.g., draindown events or stuck open safety relief valves)? This needs to be considered during our next meeting.

Public: Stuck open safety relief valves should be considered.

NRC: We are tentatively scheduling another public workshop for the November 8/9, 2000, time-frame. We can still have that meeting, but use a different format, i.e., a more detail-specific working meeting. Is this amenable to the Owners Group?

Public: We will check with owner's group representatives.

NRC: If those dates are not convenient, the following week would also be okay.

Public: How soon should we exchange question lists?

NRC: List of questions should be generated 2 weeks before meeting.

Public: B&WOG can probably have their list ready in the next 2-3 weeks.

Public: Do the values presented in the table on core damage frequencies at shutdown account for the fraction of time at shutdown?

NRC: The numbers in the table are pro-rated; all numbers are converted to a per calendar-year basis. There are times (modes) during shutdown when the instantaneous risk can be a lot higher than during full-power operation.

Public: At the Indian Point hearings, incremental risk from seismic events was not that great, since a seismic event of the magnitude necessary for significant offsite consequences would probably kill off most of the surrounding population anyway.

NRC: For the detailed public working meeting, is one day sufficient, or should we plan for two days?

Public: We will probably spend most of the day discussing PRAs (probabilistic risk assessments) and numbers. With other topics included, we may need more than one day.

NRC: It is more difficult to block out two days.

Public: Probably should block out two days, just in case.

4.3 Additional Public Comments

This section presents the one written public comment that was received.

> The Qualitative Guidelines are (loosely) based on the Safety Goals because they parse the probabilities of the QHOs. In February, the earlier workshop noted that the product across the rows was 10^{-6}, slightly above the early fatality goal. Nevertheless, that was OK because

there would be conservatisms in each of the columns. Since then, we have dropped 10^{-1} for limiting public health effects (column 4) and changed the overall product to 10^{-5} (a subtle but transparent change). The problem is that the conservatisms remain. Concerns expressed from the audience can be summarized as that NRC will concentrate its focus on the values in each column (e.g., 10^{-4}). This is likely to result in decisions that are <u>NOT</u> consistent with the safety goals. If the framework is applied in that manner, then indeed <u>it</u> is not consistent with the safety goals.

The concept that changes may reduce the initiating frequency enough to change rows is an intriguing way to avoid this trap. It is subject to at least two problems:

1. NRC–and its contractors–need to focus on factors that would reduce initiating frequency, since this is an NRC tool. Historically, the industry has not seen that kind of focus on initiating frequencies from NRC.

2. The NRC must be willing to accept that some accident sequences historically treated can become "Rare" events, and to defend that change from criticisms that will undoubtedly arise.

Skepticism that these two problems will prevent application of the framework from being overly conservative underlies the concerns heard at the workshop.

APPENDIX A. WORKSHOP AGENDA

Workshop Agenda

Monday, October 2, 2000

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 (Continued) WOG Presentation B&W OG Presentation Open Discussion
2:40 pm - 3:00 pm	Break
3:00 pm - 3:30 pm	Discussion on 50.46 (Continued) Open Discussion
3:30 pm - 4:00 pm	WRAPUP

APPENDIX B. WORKSHOP REGISTRATION LIST

LIST OF ATTENDEES

Table B.1 contains a list of the Workshop attendees. This list was generated from the attendance forms that were returned and from personal recollection of who attended. As such, the list may not be complete.

Table B.1 Workshop Registration

Name	Affiliation
John Barry	Westinghouse
Allen Camp	Sandia National Laboratories
Nancy Chapman	SERCH/Bechtel
Mike Cheok	Nuclear Regulatory Commission (NRR/DSSA)
Bob Christie	Performance Technology
Dr. Hugo C. Da Silva	TXU Electric
Mary Drouin	Nuclear Regulatory Commission (RES/DRAA/PRAB)
Bert Dunn	B&W OG, FTI
Răducu Gheorghe	Canadian Nuclear Safety Commission
Don J. Green	Tennessee Valley Authority
Eric Haskin	ERI Consulting
Ken Heck	Nuclear Regulatory Commission (NRR/DIPM/IQMB)
Adrian Heymer	Nuclear Energy Institute
Robert Hill	Carolina Power & Light
Roger Huston	Licensing Support Services
Tom King	Nuclear Regulatory Commission (DRAA)
Alan Kuritzky	Nuclear Regulatory Commission (RES/DRAA/PRAB)
Jeffrey L. LaChance	Sandia National Laboratories
John Lehner	Brookhaven National Laboratory
Stewart Magruder	Nuclear Regulatory Commission (NRR/DIPM/IQMB)
Eileen McKenna	Nuclear Regulatory Commission (NRR/DRIP/RGEB)
Gary D. Miller	Dominion Generation (Virginia Power)
Vinod Mubayi	Brookhaven National Laboratory
Paige T. Negus	GE Nuclear Energy
Mitch Nissley	Westinghouse Electric Company
Bob Osterrieder	Westinghouse/WOG
Sid Powell	Florida Power Corporation / Crystal River-3

Table B.1 Workshop Registration

Name	Affiliation
Trevor Pratt	Brookhaven National Laboratory
Terrance A. Riech	Commonwealth Edison
Zoltan R. Rosztoczy	Zeetech, Inc.
Glen E. Schinzel	South Texas Project N.O.C.
Mike Snodderly	Nuclear Regulatory Commission (NRR/DSSA/SPSB)
Herschel Specter	RBR Consultants, Inc.
Gary Vine	Electric Power Research Institute
Charles Willbanks	NUS Information Services

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APPENDIX C. NRC PRESENTATION MATERIAL

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

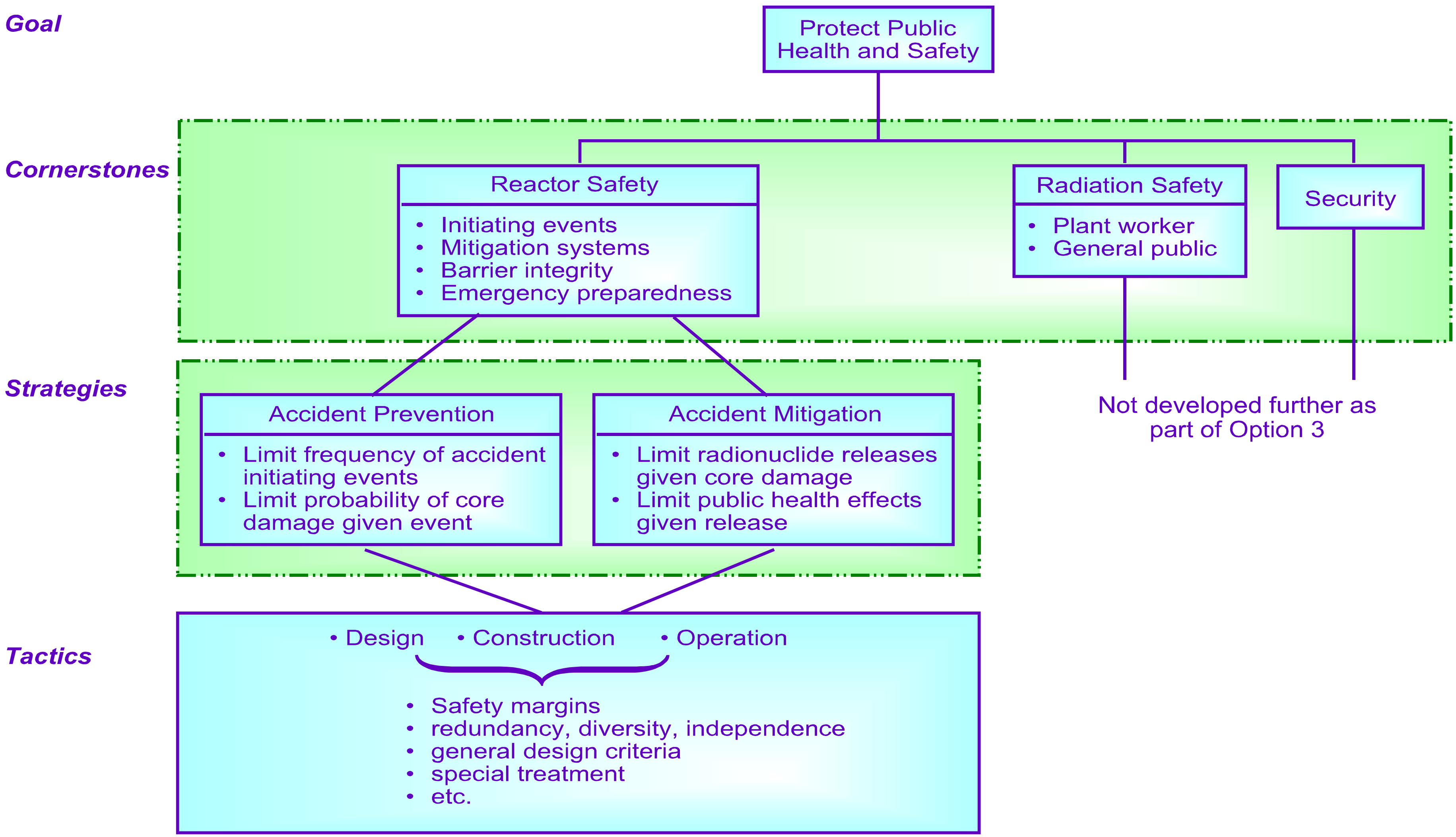
- ***Application*** Process to identify
  — holes in current requirements
  — current requirements not contributing to safety

  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
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
Not developed further as part of Option 3
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 ($\leq 5x10^{-7}$/year) • latent cancer fatality goal ($\leq 2x10^{-6}$/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	≥ 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

- P Risk-informed alternatives may include elimination, modification and addition to the technical requirements
- P Implementation of risk-informed alternative is voluntary, therefore, no need for backfit
- P Licensees can do their own cost-benefit assessment before volunteering
- P 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

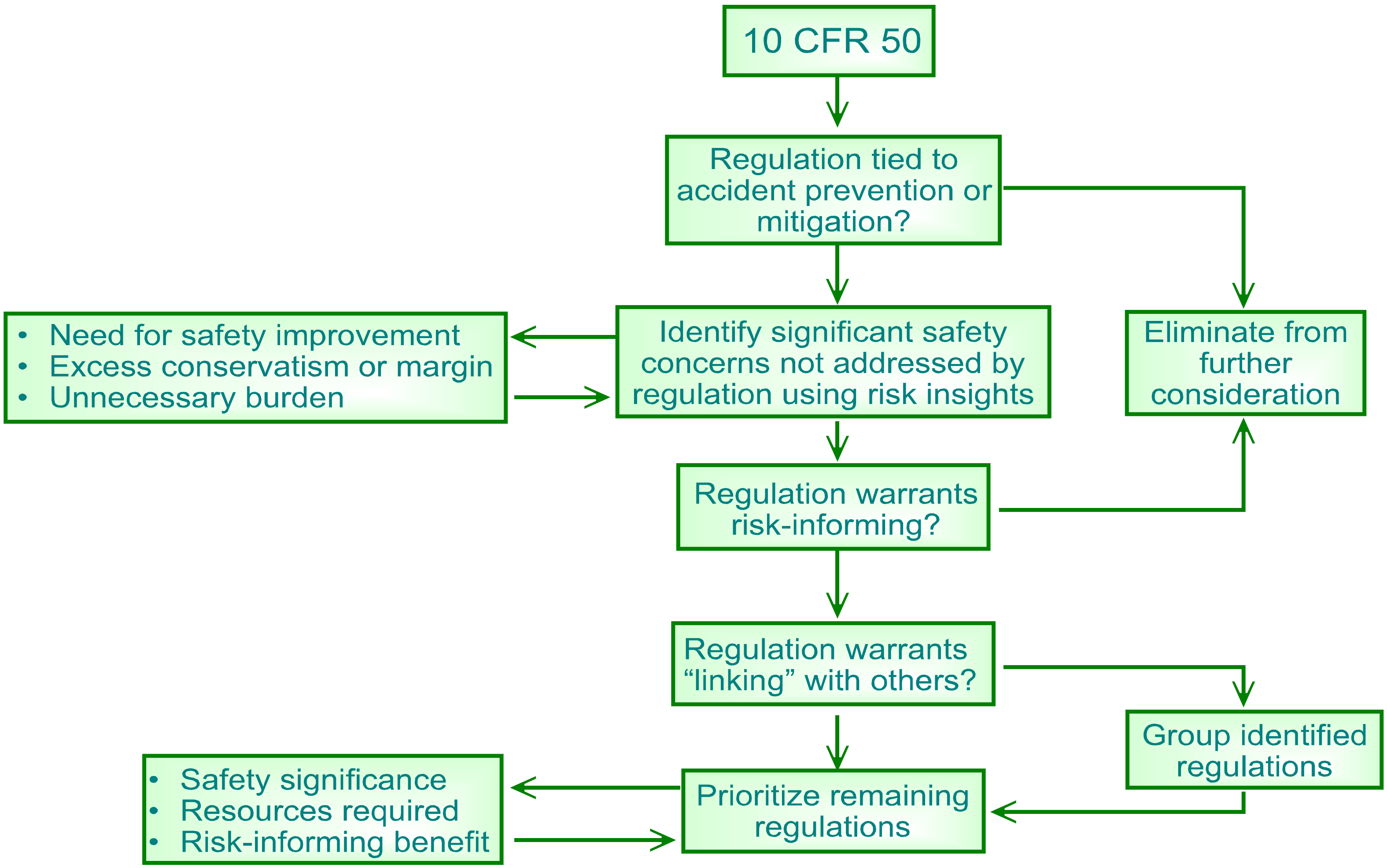
- Implies a measure of the conservatism employed in a design or process to assure a high degree of confidence that it will work to perform a needed function
- Excessive conservatism (i.e., safety margin) can lead to incorrect safety conclusions
- Safety margin imposed to account for uncertainties in 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 95th 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

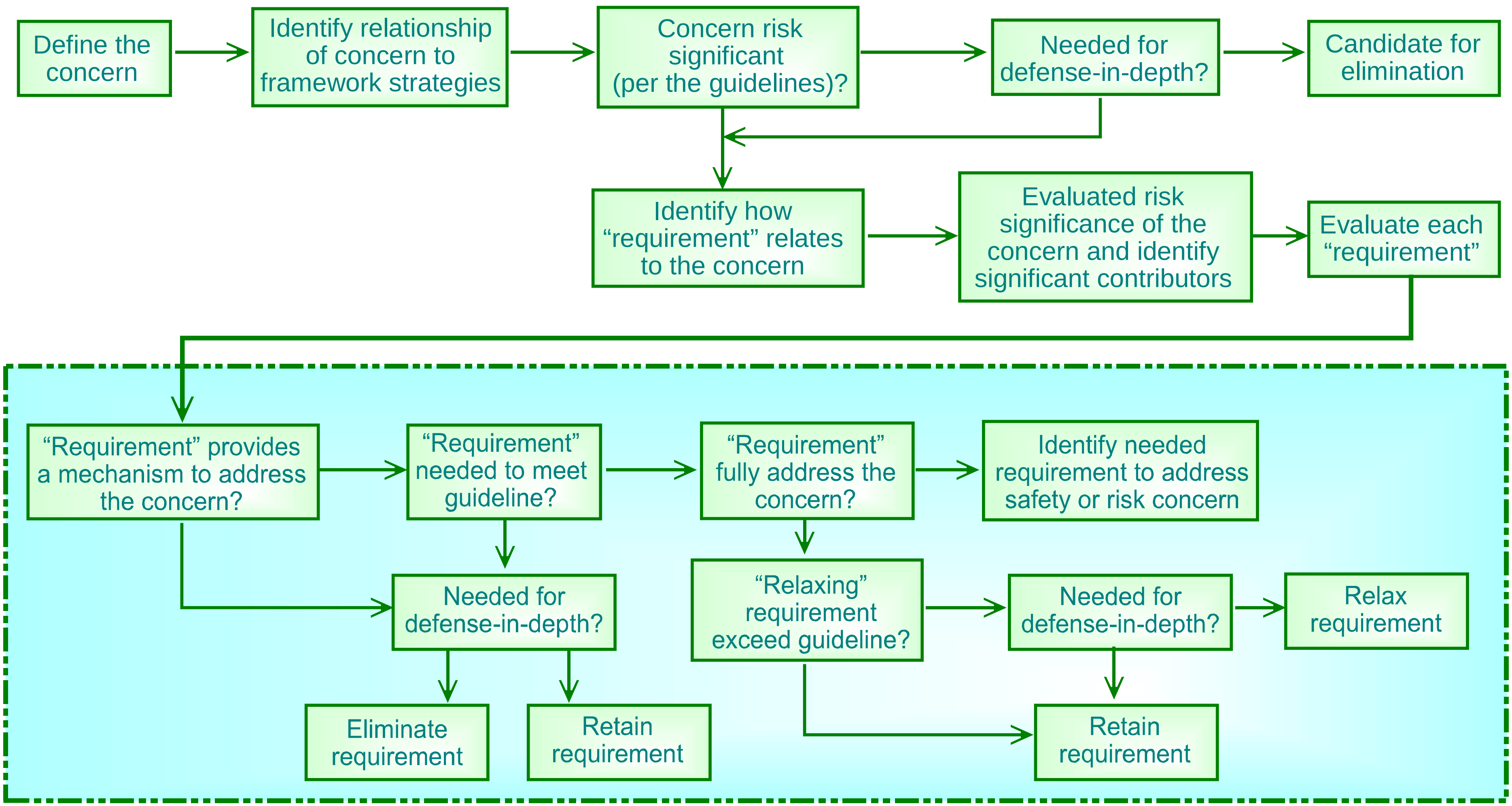


FRAMEWORK — DEVELOPMENT OF RISK-INFORMED ALTERNATIVE

- 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

EVALUATION OF POSSIBLE REVISIONS TO REQUIRED FEATURES OF 10 CFR 50 APPENDIX K

G. Norman Lauben
Safety Margins and Systems Analysis Branch
Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research

Workshop on Risk-Informing 10 CFR 50.46
10/02/00

EVALUATION OF POSSIBLE REVISIONS TO REQUIRED FEATURES OF APPENDIX K

BACKGROUND

1. **RES proposed to develop a set of sensitivity studies to determine how much of the conservatism in Appendix K was tied up in the decay heat requirement. It was expected to be design specific and dependent on the specific licensee or applicant's current approved ECCS model. Another important reason for these studies was to assure that there were no surprising interactions with other features contained in various approved Appendix K models.**

2. **NRR requested RES to also evaluate more realistic decay heat models.**

3. **RES stated that it would make maximum use of existing studies. Where needed RES performed additional sensitivities.**

4. **10 CFR 50.46 allows two options for calculating ECCS performance using either:**

   (1) A model in conformance with the required and acceptable features of Appendix K (1974 ECCS rule), or:

   (2) A realistic model with evaluation of modeling and input uncertainties so that there is a high level of probability that the ECCS criteria would not be exceeded (1989 ECCS rule change). It should be noted that by electing to use the realistic option, an applicant/ licensee/ vendor can choose any defensible realistic models, evaluate and combine uncertainties and obtain the maximum benefit.

   (3) A third option (described in SECY-83-472) is available which allows reduction in non-Appendix K specified conservatisms as long as a "best estimate" analysis is also provided to justify the proposed reduction. GE and W took advantage of this option for several plants.

RES EVALUATION PROCESS

- **RES evaluated the effect of allowing more realistic models for decay heat and metal water reaction in Appendix K analyses. In particular:**

  1. **Would the model changes result in any significant risk changes?**
  2. **What is the reduction in <u>conservatism</u> associated with separate or combined model changes?**
  3. **What is the retained conservatism as a result of the changes?**
  4. **Are there any surprising interactions with other features contained in ECCS evaluation models?**

- **Any modification to Appendix K should use simple decay heat and/or metal water reaction models with an appropriate uncertainty for each model.**

- **RES has chosen to evaluate the 1979 ANS decay heat standard and the Cathcart-Pawel metal water reaction model, since they are referenced as acceptable models in Reg. Guide 1.157 (Best Estimate Calculations of ECCS Performance,1989).**

- **Other decay heat and metal water models would be possible candidates, but they are similar in magnitude to the selected models.**

REDUCTION IN CONSERVATISM AND RETAINED CONSERVATISM

- **To evaluate the reduction in conservatism and the retained conservatism of using more realistic models for a sufficient sampling of plant types the information should include:**

  1. **A current Appendix K calculation,**

  2. **One or more Appendix K calculations using the more realistic decay heat and/or metal water models. For NRC and contractor analyses, those models include appropriate uncertainties, and**

  3. **A best estimate calculation that meets the requirements for the realistic option of 50.46.**

- **The difference in results between 1 and 2 is a measure of reduction in conservatism achieved by using less conservative models.**

- **The difference in results between 2 and 3 is a measure of retained conservatism.**

- **Some additional analyses will also be performed to estimate the increase in thermal power available by utilizing more realistic decay heat and metal water reaction models.**

- **RES solicited information from industry to facilitate this effort.**

SUMMARY OF RESULTS

1. The average benefit in 5 "Appendix K" large break comparison studies of using a best estimate decay heat model compared to ANS71 X 1.2 was about 358F. The numbers ranged from 168F to 462F and depends on the time of PCT. In 2 "best estimate" permutations with early PCT the benefits were 41F and 48F.

2. The average decay heat benefit in 5 small break studies was 432F with numbers ranging from 248F to 712F.

3. The average benefit in 4 large break studies of using Cathcart-Pawel vs. Baker-Just metal-water reaction model was 59F with numbers ranging from 45F to 73F. The effect is extremely temperature dependent.

4. If a decay heat credit is already taken, one large break calculation showed a metal water reaction model benefit of only 2F because the temperature was already low.

5. Little margin exists between Appendix K calculations and approved BE analyses for two W PWRs (~220F). Since the average Appendix K large break decay heat benefit was 358F, the margin could disappear, if this benefit were allowed.

6. The margin appears to be larger for BWRs, but NRR has applied a 600F PCT penalty to the GE 83-472 model (SAFER/GESTR).

IMPLEMENTING THE 1979 ANS DECAY HEAT STANDARD

- **The 1979 standard is more accurate but more complex than the 1973 standard.**
- **The two standards can be represented by the following equation:**

$$P = \frac{P_t}{Q}\left[M \times G(t, T, \Psi) \times \Sigma F_i(t, T) + F_{HE}(t, T, R) \right]$$

Where:

P	**=**	**total decay power**	**(Megawatts)**
P_t	**=**	**maximum reactor power during operation**	**(Megawatts)**
Q	**=**	**recoverable energy per fission**	**(Mev/fission)**
M	**=**	**uncertainty multiplier**	
G	**=**	**neutron capture factor**	
t	**=**	**shutdown time**	**(Seconds)**
T	**=**	**operating time**	**(Seconds)**
Ψ	**=**	**fissions per initial fissile atom**	
F_i	**=**	**decay power per fissionable nuclide**	**(Mev/fission)**
F_{HE}	**=**	**decay power for actinides**	**(Mev/fission)**
R	**=**	**atoms of U239 produced per fission**	

IMPLEMENTING THE 1979 ANS DECAY HEAT STANDARD (CONTINUED)

- Following are values used for the current Appendix K or suggested for the revised version

Parameter	73 standard	Appendix K	79 standard	Suggested Appendix K
P_t	total	total	segmented	maximum (example 3)
Q	200	200	user justified	200 (example 3)
M	1.2 (table 1)	1.2 (table 1)	Eq. 12 & 13	1.1 (example3)
G	N/A	1.0	Eq. 11	Eq. 11
T	∞	∞	user justified	4 years (example 3)
Ψ	N/A	N/A	user justified	1.0 (example 3)
F_i	F_{235}	F_{235}	1,2 or 3 isotopes	F_{235} (example 3, table 7)
F_{HE}	Eq. 3,4	Eq. 3,4	Eq. 14,15,16	Eq. 14,15,16
R	user	?	User justified	1.0 (RELAP5 default)

- All material and suggested values (except R) are referenced in the two consensus standards.
- Spatial and temporal core segmentation is discussed and allowed in the 79 standard based on reload patterns and power history, but adds complexity to calculation.
- IN 96-39 recognized complexity and variations in implementation and sometimes significant variation in results.
- It is suggested that burnup be limited to 80 Gwt/Mtu on the peak rod.

PRELIMINARY SUGGESTIONS AND OBSERVATIONS

1. **Other proposals described in this workshop for risk-informing 50.46 are closely related to removing conservatisms in Appendix K. The results of this study should be factored into the overall process of risk-informing Part 50.**

2. **Potential replacements for the Baker-Just metal-water reaction model will have only a small effect (about 15% as much as the decay heat benefit). Substantial modification as a result of the high burn-up review are also likely. A change might also require a change in the 17% metal-water limit.**

3. **A methodology should be considered that addresses known non-conservatisms in Appendix K models and provides for assessment of the remaining overall conservatism after Appendix K model changes are made.**

4. **Piecemeal modification to Appendix K was rejected in 1986-7 and does not encourage the development and use of best estimate analysis methods, which, in the long run, are much more consistent with the philosophy of risk-informed realistic regulation.**

5. **The suggested modification to the decay heat standard attempts to maintain a similar philosophy of safety and conservatism that was thought to exist in the original Appendix K while allowing for improvement in the state-of-the-art.**

6. **Feedback would be helpful regarding all suggestions especially items 2 and 3 on this slide and all of the parameter selections on the decay heat implementation slide.**

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 Leak detection Pipe break LBB	10 CFR Sections	Regulatory Requirements
2	ECC GDC-35 50.46 LOCA	SRP/FSAR Sections Regulatory Guides Branch Technical Positions Task Action Plan Items	Regulatory Guidance
3	GDC-38 GDC-41 GDC-50	Unresolved Safety Issues 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 Performance 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

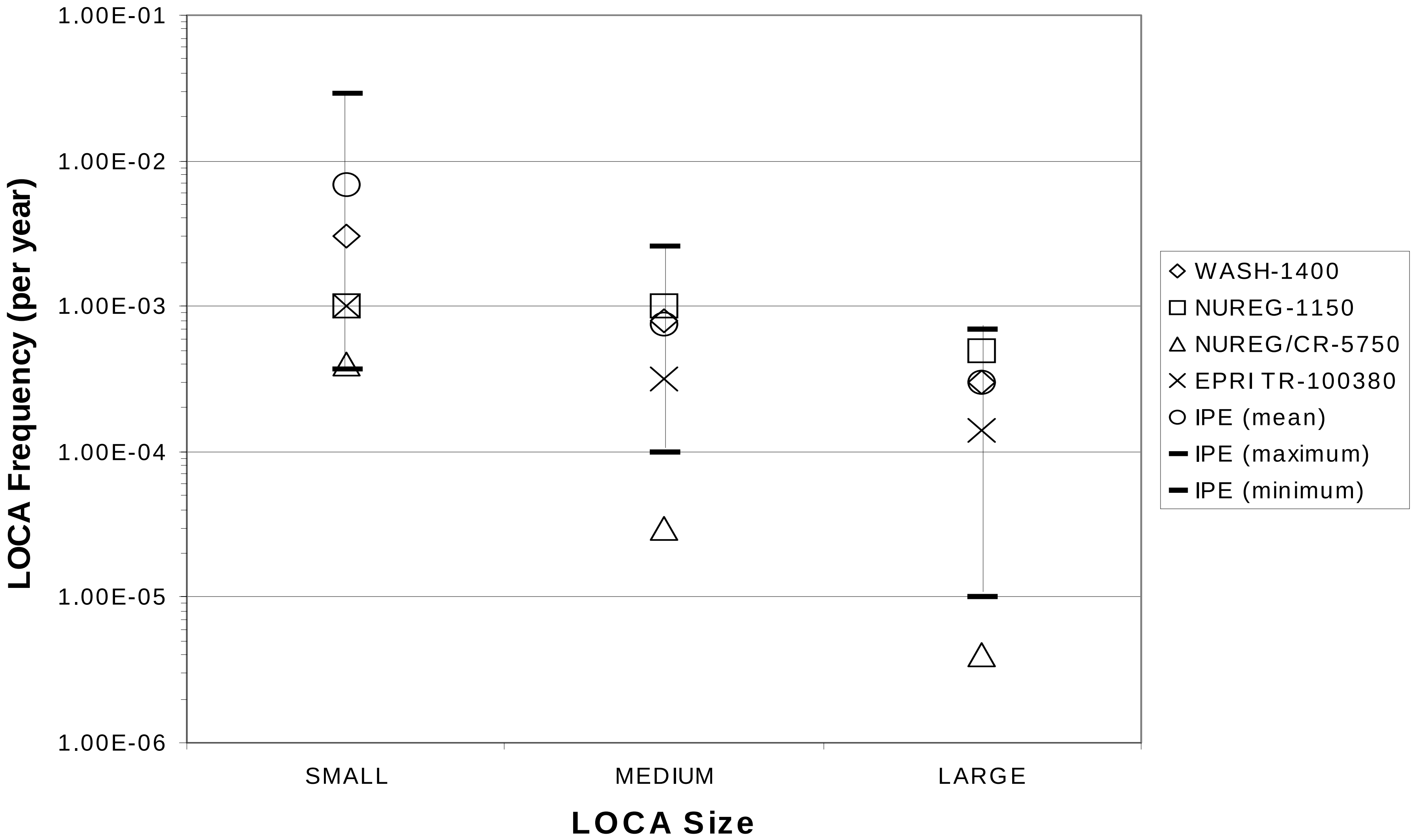
NUREG/CR-5750 METHODS FOR ESTIMATING PIPE-BREAK LOCA FREQUENCIES

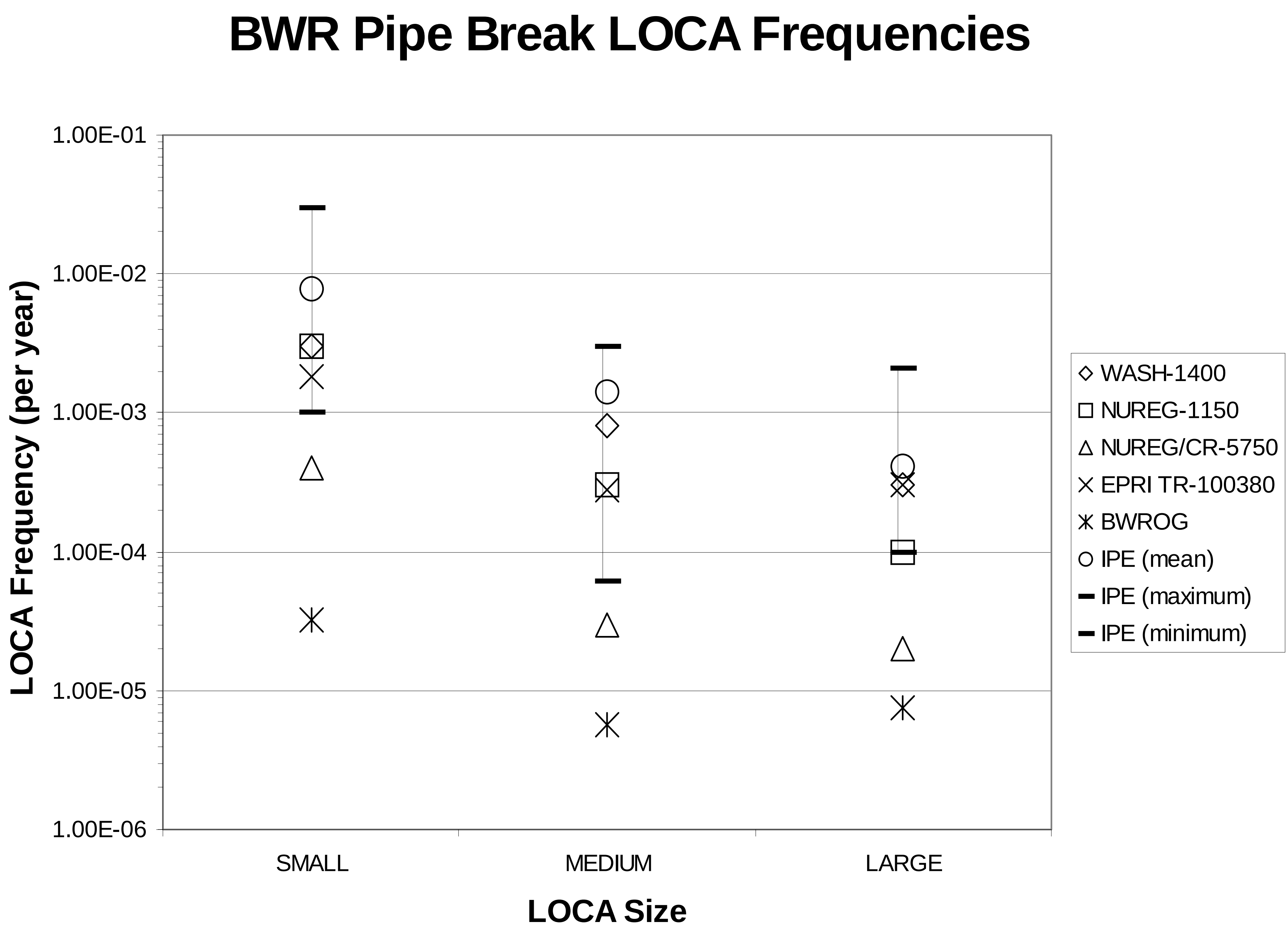
- Estimate through-wall crack frequency based on data
- Adjust downward for IGSCC mitigation (BWRs)
- Multiply by conservative estimate of probability of rupture given a through-wall crack ($P_{R:TW)}$ based on
  - Technical review of information on fracture mechanics
  - Data on high-energy pipe failures 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
LOCA Frequency (per year)
1.00E-01
1.00E-02
1.00E-03
1.00E-04
1.00E-05
1.00E-06
SMALL
MEDIUM
LARGE
LOCA Size
WASH-1400
NUREG-1150
NUREG/CR-5750
EPRI TR-100380
BWROG
IPE (mean)
IPE (maximum)
IPE (minimum)

DOUBLE-ENDED GUILLOTINE BREAKS (DEGB)

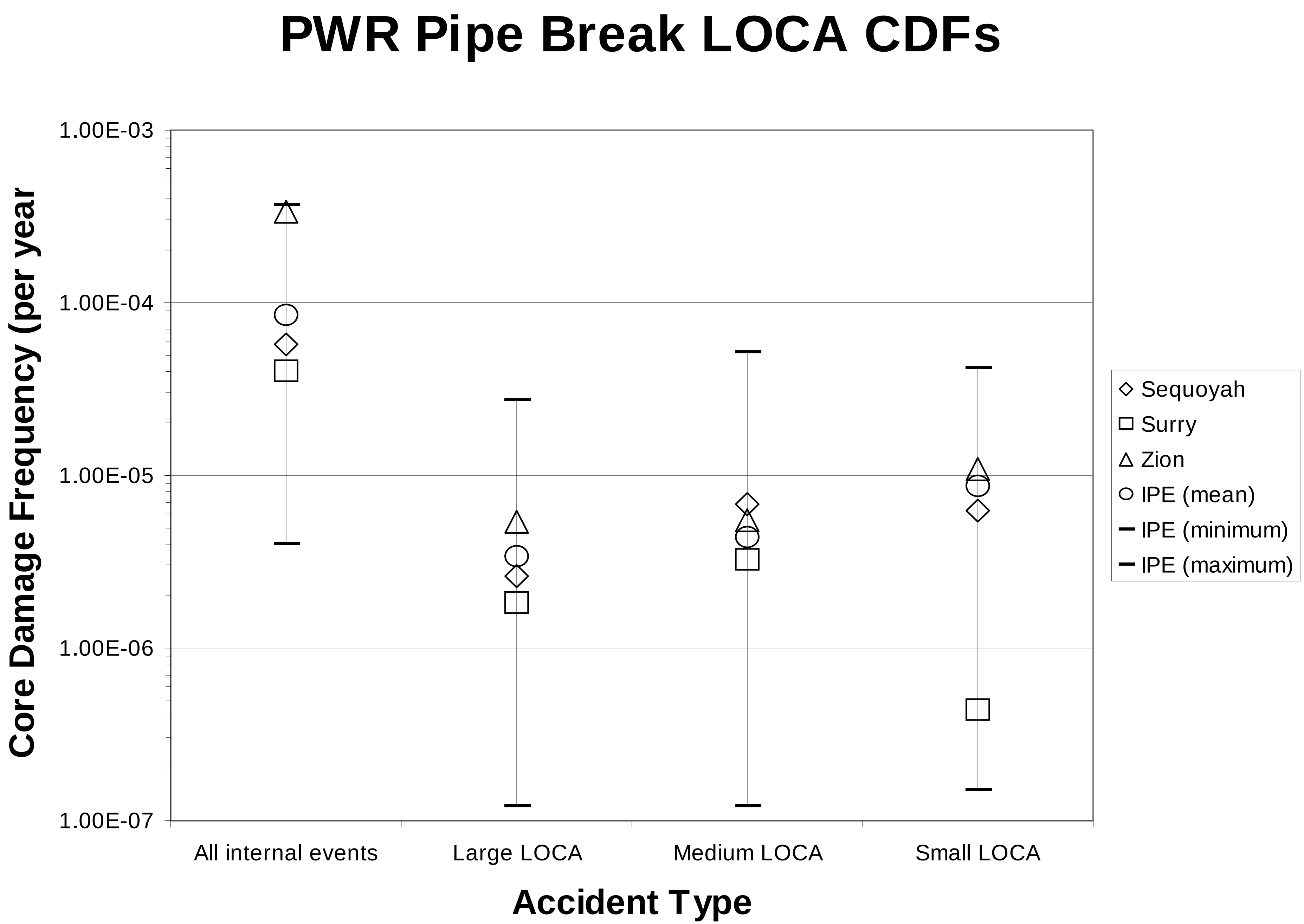
- Frequency of DEGBs have been estimated using fracture mechanics as part of LBB assessment (NUREG-1061 and NUREG/CR-4792)
- For CE plants,
  - Point estimates range from 6E-14/yr to 5E-13/yr
  - 90 percentile values range from 4E-12/yr to 7E-11
- For Westinghouse plants,
  - Median frequencies range 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

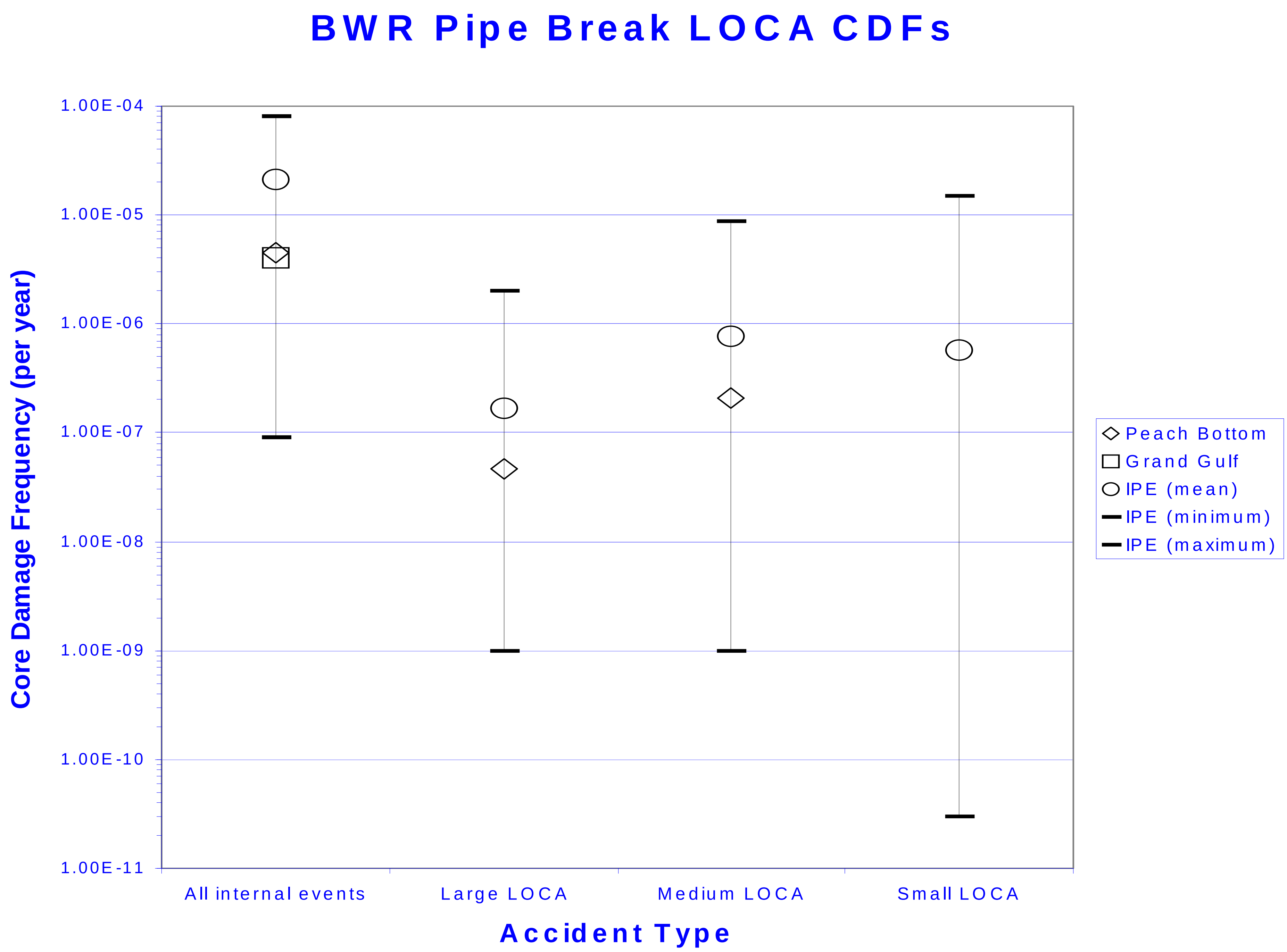
- Estimates using fracture mechanics provided in LBB analysis in NUREG/CR-3660 and NUREG/CR-3663
  - Frequency of direct seismic-induced pipe break LOCAs are 1 to 3 orders of magnitude lower than random LOCAs
  - Frequency of indirect seismic-induced DEGBs was significantly higher than frequency of direct 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)

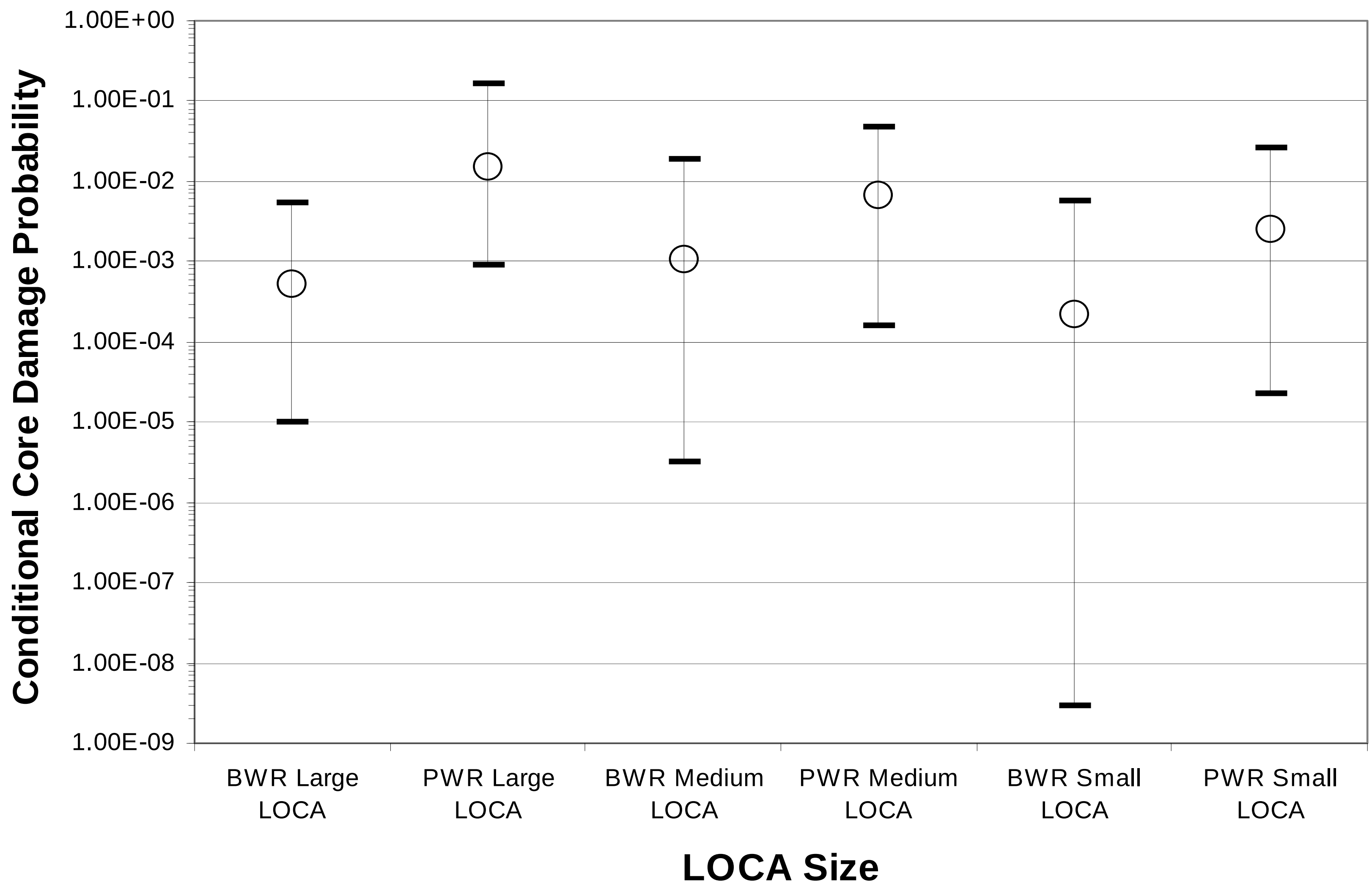


PWR Pipe Break LOCA CDFs
1.00E-03
1.00E-04
1.00E-05
1.00E-06
1.00E-07
Core Damage Frequency (per year
All internal events
Large LOCA
Medium LOCA
Small LOCA
Accident Type
Sequoyah
Surry
Zion
IPE (mean)
IPE (minimum)
IPE (maximum)



BWR Pipe Break LOCA CDFs
Core Damage Frequency (per year)
1.00E-04
1.00E-05
1.00E-06
1.00E-07
1.00E-08
1.00E-09
1.00E-10
1.00E-11
All internal events
Large LOCA
Medium LOCA
Small LOCA
Accident Type
Peach Bottom
Grand Gulf
IPE (mean)
IPE (minimum)
IPE (maximum)

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		

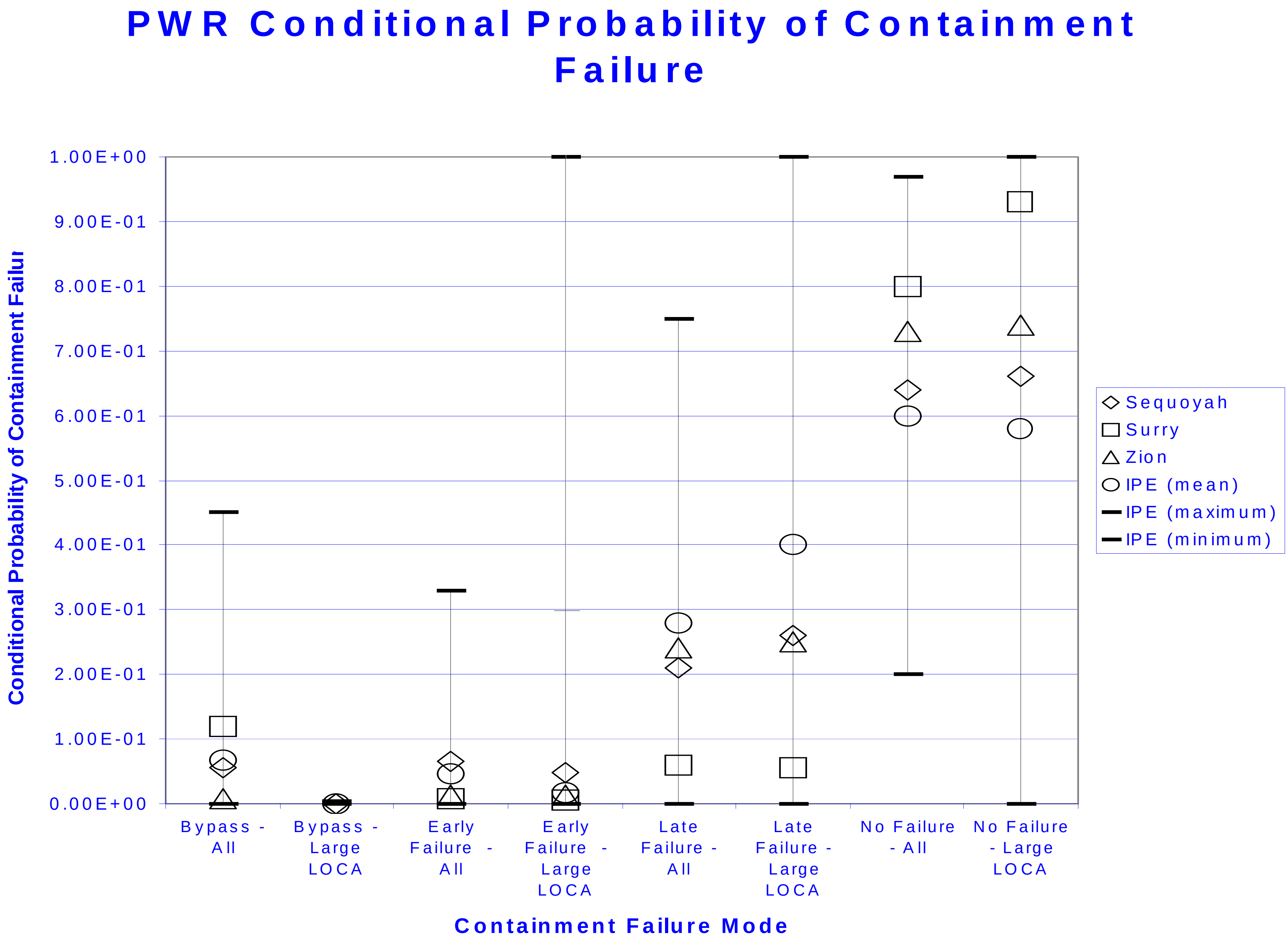
CDF FROM SEISMIC-INDUCED LOCAs

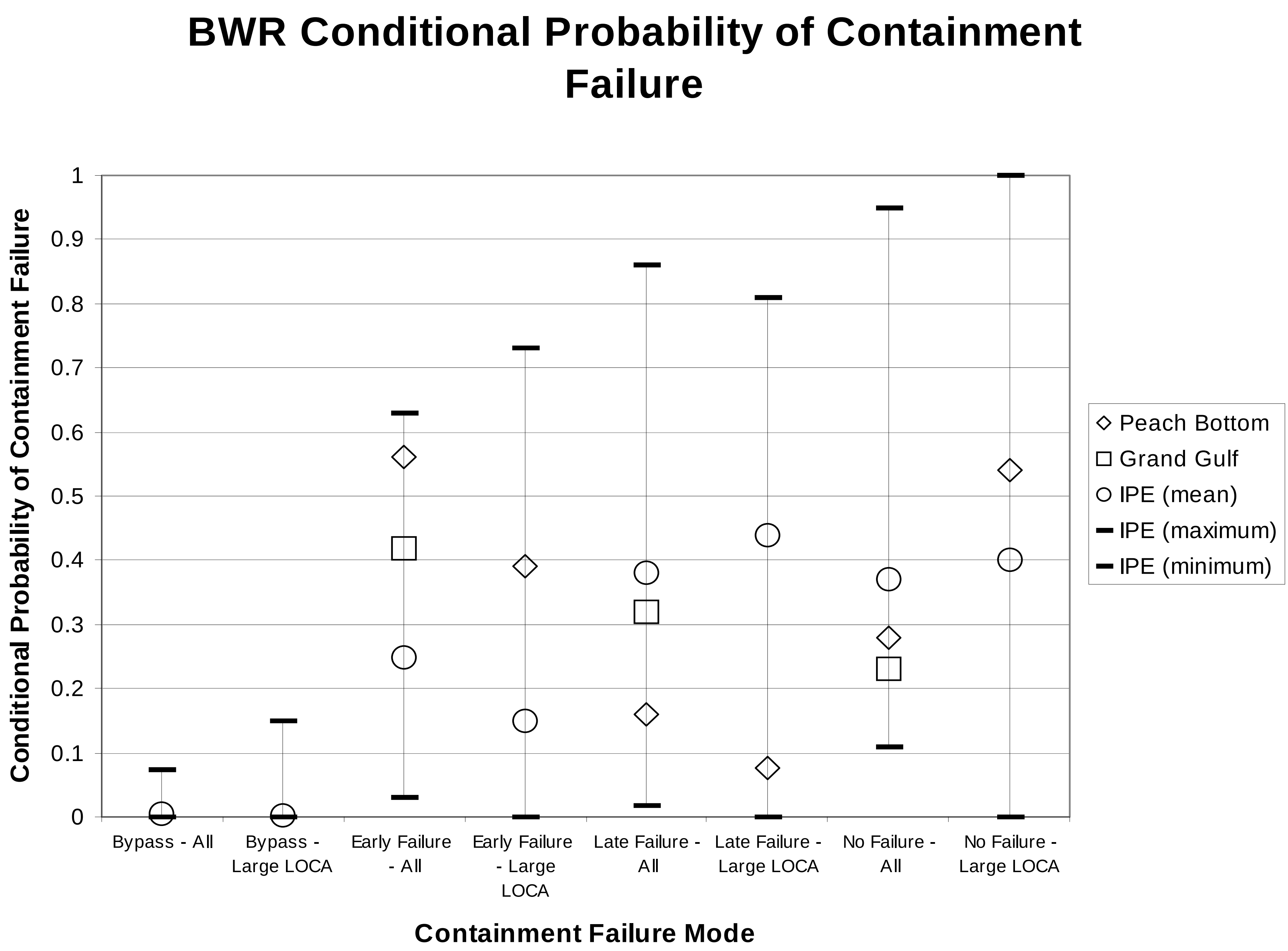
Accident Type	Mean Core Damage Frequency	
	LLNL Hazard Curve	EPRI Hazard Curve
Large LOCA		
Peach Bottom (NUREG-1150)	1.9E-05	6.8E-07
Surry (NUREG-1150)	7.7E-06	1.3E-06
Medium LOCA		
Peach Bottom (NUREG-1150)	7.4E-06	2.1E-07
Surry (NUREG-1150)	1.5E-06	1.7E-07
Small LOCA		
Peach Bottom (NUREG-1150)	1.5E-06	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
Conditional Probability of Containment Failur
1.00E+00
9.00E-01
8.00E-01
7.00E-01
6.00E-01
5.00E-01
4.00E-01
3.00E-01
2.00E-01
1.00E-01
0.00E+00
Bypass - All
Bypass - Large LOCA
Early Failure - All
Early Failure - Large LOCA
Late Failure - All
Late Failure - Large LOCA
No Failure - All
No Failure - Large LOCA
Containment Failure Mode
Sequoyah
Surry
Zion
IPE (mean)
IPE (maximum)
IPE (minimum)



BWR Conditional Probability of Containment Failure
Conditional Probability of Containment Failure
1
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
Bypass - All
Bypass - Large LOCA
Early Failure - All
Early Failure - Large LOCA
Late Failure - All
Late Failure - Large LOCA
No Failure - All
No Failure - Large LOCA
Containment Failure Mode
Peach Bottom
Grand Gulf
IPE (mean)
IPE (maximum)
IPE (minimum)

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-Infomed 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

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APPENDIX D. PUBLIC PRESENTATION MATERIAL

Public Workshop

Risk-Informing the Technical Requirements of 10CFR50

October 2, 2000
NRC Auditorium - One White Flint

Bob Christie

Performance Technology
P. O. Box 51663
Knoxville, TN 37950-1663
(865) 588-1444
FAX (865) 584-3043
performtech@compuserve.com

RJC
10/1/00

SUMMARY
Change Hydrogen Control Regulations as of 10/2/00

10CFR50.12 Exemption Requests	Petition for Rulemaking	SECY-98-300 Option 3 Framework	SECY-98-300 Option 3 10CFR50.44
Submitted:	**Recent action:**	**Recent action:**	**Agreement:**
San Onofre 2&3 - 9/10/98 Approved 9/3/99	Letter - Christie to Mike Snodderly (NRC), 7/3/00	ACRS 7/11/00	Delete post LOCA hydrogen requirements including delete measuring hydrogen concentration in short term.
Oconee - 7/26/00	ACRS 7/12/00	Letter - Christie to Ashok Thadani (NRC), 7/19/00	Containment air mixing unchanged
Three Mile Island 9/20/00	Letter - Christie to Sam Collins (NRC), 7/14/00	Letter - Christie to Ashok Thadani (NRC), 8/24/00	Reactor Coolant System high point vents unchanged
	Letter - Christie to Cynthia Carpenter (NRC), 7/20/00	ACRS 8/29/00	Mark I's and Mark II's inerted unchanged
	ACRS 8/29/00	SECY-00-0198	**Disagreement:**
	Letter - Carpenter (NRC) to Christie, 8/30/00		NRC staff wants to add specific combustible gas source term for severe accidents
	Letter - Christie to Cynthia Carpenter (NRC), 9/7/00		NRC wants igniters operable during Station Blackout for Mark IIIs and ice condensers
			Add long term hydrogen control in Severe Accident Management Guidelines
Future action:	**Future action:**	**Future action:**	**Future action:**
Other submittals in preparation	Mike Snodderly (NRC) working on open purge valve - severe accident	NRC proceeding while waiting for direction from Commissioners	NRC proceeding while waiting for direction from Commissioners

2/3

RJC
10/1/00

Combustible Gas Control

The utilities recommended a program (Whole Plant Study) in 1997 to change the existing requirements of 10CFR50 to make the regulations more effective and efficient through the use of Probabilistic Risk Assessment. Combustible gas control was chosen as the regulation to demonstrate feasibility of the program (Task Zero).

Task Zero for Arkansas Nuclear One and San Onofre were approved by the NRC staff in September 1998 and September 1999 respectively on the basis of enhanced safety and burden reduction. A petition for rulemaking was submitted based on the results of Task Zero. It is now 13 months since San Onofre Task Zero approval with no change at any nuclear electric power unit to enhance safety and reduce burden with respect to combustible gas control and no progress on the petition for rulemaking.

The utilities are now submitting exemption requests in order to change their nuclear plants to enhance safety and reduce burden with respect to combustible gas control. Why is this necessary?

Westinghouse Owners Group
Large Break (LB) LOCA Redefinition Program

Bob Osterrieder
Westinghouse Electric Company

NRC Workshop
October 2, 2000

Presentation Overview

- Background
- Program Approach
- Safety Benefits
- Examples of Potential Benefits
- Program Plan
- Safety Margin
- Conclusions

Westinghouse Owners Group
LBLOCA Redefinition Program

Background

- November 20, 1998 WOG Letter to Commissioner Diaz
  - Identified LBLOCA as Highest Priority Regulatory Issue
- January 12, 2000 NEI Letter to Chairman Meserve
  - Identified 10CFR50.46 and Appendix K to Part 50 as Prime Candidates for Assessment and Change
- WOG Considered Options and Concluded that LBLOCA Redefinition is Best Option to Pursue
- NRC Options Presented at February 2000 Option 3 Workshop

Westinghouse Owners Group
LBLOCA Redefinition Program

Program Approach

- Licensing - Consistent with Risk-Informing Part 50
  - SECY-99-264 (Proposed Staff Plan for Risk-Informing Technical Requirements in 10 CFR Part 50)
- Risk-Informing the Technical Requirements Complements Risk-Informing the Special Treatment Requirements and Adds Clarity/Consistency to the Regulations
- Changes to 10 CFR Part 50
  - 50.46 Acceptance Criteria for ECCS
  - Appendix A GDC (LOCA definition)
  - Appendix K (I.C.1)

Westinghouse Owners Group
LBLOCA Redefinition Program

Program Approach (continued)

- Redefine the Design Basis LOCA
  - Size
  - Attendant Consequences
- Maintain an Acceptable Margin of Safety
- Technical Justification
  - Use Risk-Informed Technology to Show Low Risk of LBLOCA
    - Utilize the Framework Contained in Regulatory Guide 1.174
  - Use Leak Before Break Analysis to Justify Break Size
    - Define New Limiting Break Size & Reanalyze, as Necessary, to Obtain Benefits
    - Intent is to Reduce Break Size Down to SBLOCA Range

Westinghouse Owners Group
LBLOCA Redefinition Program

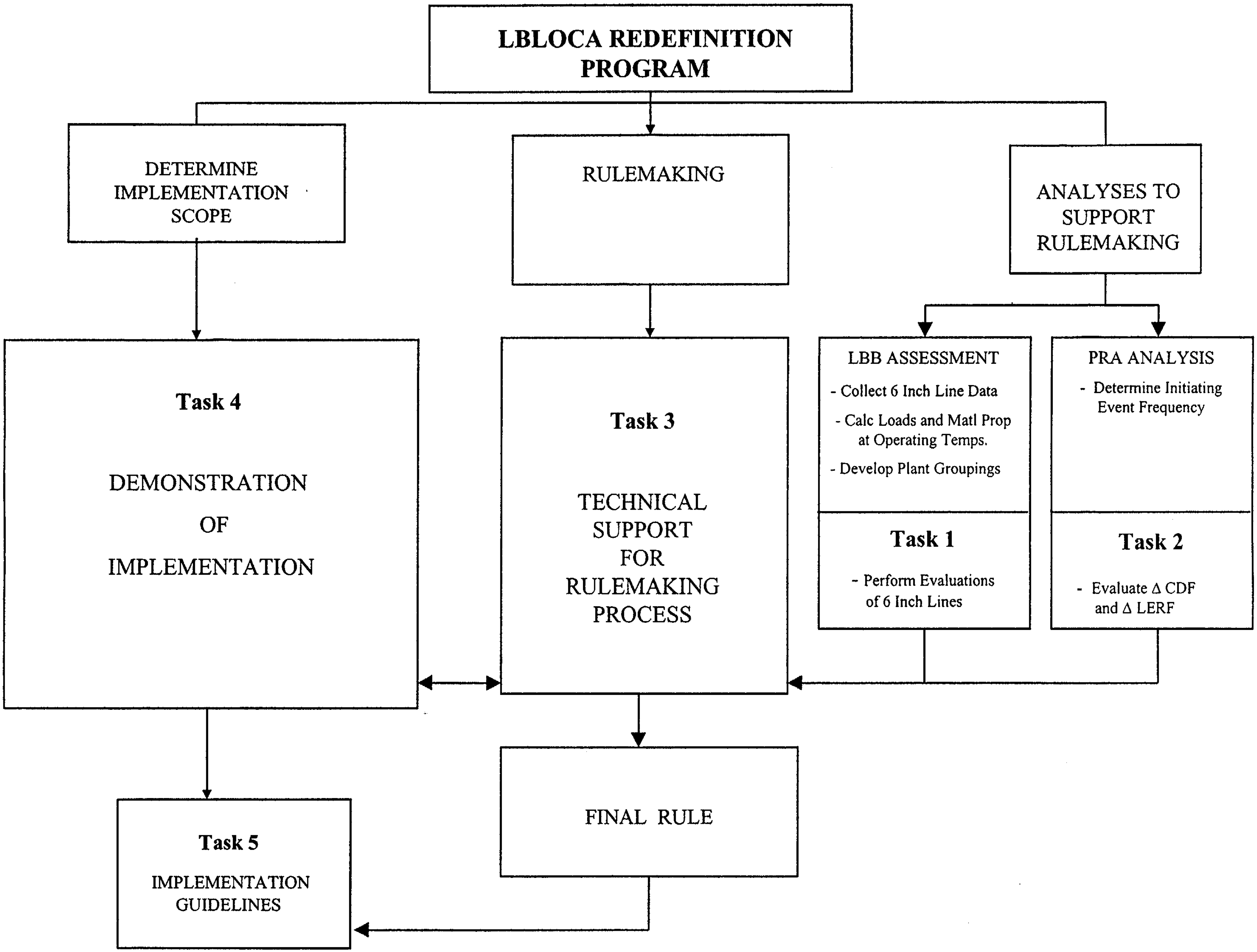
Safety Benefits

- Focus Resources on Activities of Greater Risk Significance
- Reduce Burden of Revising and Maintaining LBLOCA Design/Licensing Basis
- Consistency within the Regulations
- Consistency in Various Individual Analytical Applications
- Provides a More Realistic Basis for Design Evaluations
- Promotes Realistic Equipment Testing Requirements

Westinghouse Owners Group
LBLOCA Redefinition Program

Examples of Potential Benefits

- Avoided LBLOCA-Related Regulatory Issues
  - Reduced Scope of Generic Issues
- Tech Spec Relaxations
  - Diesel Generator Start Time
  - Accumulators
- Reduced Analytical Cost/Maintenance
  - Post LOCA Issues
    - Control Rod Insertion
    - Hot Leg Switchover
- Peaking Factor Increases
- Power Uprates
- Reactor Vessel Internals
  - Barrel Baffle Bolts
- Support Resolution of PWR Sump Issues


LBLOCA REDEFINITION PROGRAM
DETERMINE IMPLEMENTATION SCOPE
RULEMAKING
ANALYSES TO SUPPORT RULEMAKING
LBB ASSESSMENT
- Collect 6 Inch Line Data
- Calc Loads and Matl Prop at Operating Temps.
- Develop Plant Groupings
Task 1
- Perform Evaluations of 6 Inch Lines
PRA ANALYSIS
- Determine Initiating Event Frequency
Task 2
- Evaluate Δ CDF and Δ LERF
Task 4
DEMONSTRATION OF IMPLEMENTATION
Task 3
TECHNICAL SUPPORT FOR RULEMAKING PROCESS
FINAL RULE
Task 5
IMPLEMENTATION GUIDELINES

Demonstration of Implementation (Examples)

- Decrease Number of Accumulators in Tech Specs (N-1) or Relax Accumulator Parameter Requirements (Boron, Pressure, Water Volume)
- Increase Diesel Generator Start Time
- Lower LOCA Peak Containment Pressure
- Relax Ultimate Heat Sink Requirements
- Relax Containment Fan Cooler Requirements
- Relax ECCS Flow Balancing Requirements

Westinghouse Owners Group
LBLOCA Redefinition Program

Rule-making

- Utilize Rule-Making Based on SECY-98-300 Option 3
- Rule-Making with Option of Retaining Current Licensing Basis
- Regulatory Impact
  - 10 CFR 50.46
  - 10 CFR 50 Appendix K
  - 10 CFR 50 Appendix A General Design Criteria
- Ideally, the Regulations Should Define what the Implementing Methodology must Achieve, Rather than how the Requirements must be Satisfied
- Provides Broadest Range of Benefits
- Minimizes Staff Resources versus Exemption Requests

Westinghouse Owners Group
LBLOCA Redefinition Program

Safety Margin

- Risk-Informed Approach Maintains the Defense-In-Depth Philosophy that Underlies the Safety Regulations (SECY-98-300)
- CDF and LERF Ensure Margins for Health and Safety of the Public
- Redefinition of LBLOCA, Retains LOCA as a DBE
- Other DBAs Continue to Maintain Adequate Margin
- Focus Resources on Activities of Greater Risk Significance

Westinghouse Owners Group
LBLOCA Redefinition Program

Industry Support

- Multiple Industry Meetings Held to Discuss LBLOCA Redefinition Program
- Most Recent Meeting was August 29, 2000 Coordinated by NEI
- All NSSS Owners Groups See Benefit in Coordination of OG Activities Associated with Risk Informing 10CFR50.46
- WOG to Continue as Lead Entity for LBLOCA Redefinition

Westinghouse Owners Group
LBLOCA Redefinition Program

Conclusions

- Maintains Acceptable Margin of Safety
- Key Part of Risk-Informing Part 50
- Consistent with Intent of Reactor Oversight Process
- Single Approach to Reduce Burdens Imposed by an Unrealistic Event
  - Complementary to the 1987 GDC-4 Rule Change
  - Integrates Pursuit of Individual Applications in a Cohesive One Time Rule Change
- Consistent with Technological Advances/Knowledge
- Reduces both Staff and Industry Resources Associated with LBLOCA
- Increases Efficiency and Effectiveness
- Industry Supports Program
- **LBLOCA Redefinition is Best Option to Pursue**

Presentation on Behave of the B&W Owner's Group

by Bert M. Dunn
Framatome Technologies

* The BWOG Supports the Effort to Risk Inform Regulations Related to Large Break LOCA
* BWOG is Pursuing Support for This Activity As an Industry Program

Evaluation of Differing Programs to Risk Inform the Regulations

* Maintain Current Regulations
  - No Gain or Risk Enhanced Direction of Industry Efforts

* Change of Evaluation Models
  1. Relax Appendix K
     - Benefits Limited to Fuel Performance
  2. Easier Best Estimate LOCA
     - Benefits Limited to Fuel Performance

* Change Criteria
  - Benefits Mostly Limited to Fuel Performance
  - Future Possibility for Small Breaks if Large Breaks Eliminated

* Change Design Basis Event
  1. Relax Simultaneous Failures
     - Benefits Mostly Limited to Fuel Performance
  2. Eliminate Large Break LOCA
     - Provides Benefits Beyond 10CFR50.46
     - Not as Subject to Ever Increasing Complexity
     - Most Concrete and Decisive Approach
     - Supported by Risk and Leak-Before-Break Technologies
     - Favored by BWOG

Implementation Guidance

* Revision Must Consider Both Current Designs, Redesigns and New Designs
* Clear Limits on the Extent (Systems, Operations, or Procedural Changes) of Implementation of This Regulatory Revision Should Exist
* The BWOG Will Rely on the Concepts of Defense-in-Depth and Safety Margin to Recognize These Limitations
* The Existing Industry Consortium Has Agreed to Documentation of Considerations of Defense-in-Depth and Safety Margin and the Implications of These Concepts