PUBLIC MEETING ON POTENTIAL CHANGES TO LOCA-RELATED REGULATORY REQUIREMENTS

November 16, 2000

OFFICE OF NUCLEAR REGULATORY RESEARCH
U.S. NUCLEAR REGULATORY COMMISSION

AGENDA FOR PUBLIC MEETING ON POTENTIAL CHANGES TO LOCA-RELATED REGULATORY REQUIREMENTS

November 16, 2000

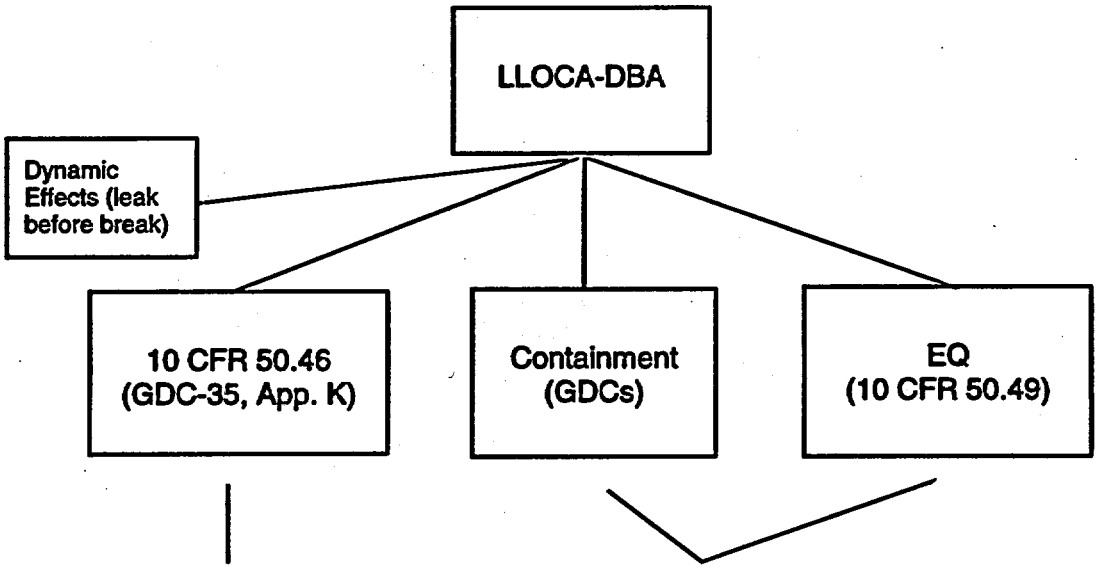
T-10A1

8:30am-9:10am	NRC Overview – Initial thoughts: scope, approach, implementation – Schedule
9:10am-2:05pm	Technical Issues Associated With LBLOCA Redefinition[1] – WOG-developed draft LOCA frequencies (~2 hrs) – Break (15 min) – LOCA-related risk data and insights (30 min) – Risk-impact of LBLOCA redefinition (45 min) – Lunch (45 min) – Potential benefits of LBLOCA redefinition (20 min) – Impact of LBLOCA redefinition on implementing documents and safety issues (20 min)
2:05pm-3:00pm	Other ECCS-Performance Options
3:00pm-3:15pm	Break
3:15pm-4:00pm	Other ECCS-Performance Options (Continued)
4:00pm-5:00pm	Approach and Implementation

[1] The technical issues associated with LBLOCA redefinition are listed in order of priority. If insufficient time is available, then not all of these issues will be discussed.

Risk-Informing Design Basis Large Break LOCA Requirements

SCOPE:



Options feasible

Recommend options and why

Schedule to work out explicit recommendations

- Complexity
- Schedule to identify options

DISCUSSION TOPICS FOR PUBLIC MEETING ON POTENTIAL CHANGES TO LOCA-RELATED REGULATORY REQUIREMENTS (NOVEMBER 16, 2000)

1 NRC Overview

1.1 Initial thoughts/scope

1.2 Schedule

2 Technical Issues Associated With LBLOCA Redefinition

2.1 <u>WOG-developed draft LOCA frequencies</u>

2.1.1 Discussion of WOG-developed draft LOCA frequencies based on fracture mechanics.

2.1.2 How will the WOG synthesize all available supporting information in their overall evaluation to demonstrate specific, or threshold, values for the frequency of large bore piping rupture (including the assessment of uncertainty in the values)?

<u>Background:</u> The draft framework (Attachment 1 to SECY-00-0198) for risk-informing regulatory requirements under Option 3 classifies a set of initiating events as rare if their frequency is demonstrably less than 1E-6 per year. The necessity of considering uncertainties in making such a determination is discussed in Section 4.4.2 of the draft framework document. NRC presentations at prior public meetings have identified several methods that could potentially be used to characterize the current state of knowledge regarding RCS pipe rupture frequencies. These include methods based on:

- the observed frequency of throughwall cracks coupled with models of the probability of pipe rupture given a throughwall crack
- the number of pipe segments and welds (EPRI)
- fracture-mechanics analyses (NUREG-1061, Volume 3)

2.2 <u>LOCA-related risk data and insights</u>

2.2.1 Are there industry comments on the LOCA-related risk data (pipe break frequencies, LOCA-related CDF, conditional containment failure probability [CCFP], and LERF, etc.) and insights presented at the public workshop on October 2, 2000?

2.2.2 What is the basis/reference for the probabilities of pipe break LOCA with simultaneous loss of offsite power presented by the staff at the October 2, 2000, public workshop?

2.2.3 Are there updated industry data on LOCA-related event frequencies and probabilities that industry would like to provide to the NRC? What are the nature and bases of this data?

2.2.4 Industry has stated that IPE results have been updated for most plants, and that more recent PRA data should be evaluated. In what form, and by what mechanism, should this updated data be provided to the staff?

2.3 Risk-impact of LBLOCA redefinition

Before a set of rare initiating events can be removed from the current set of design basis initiating events, the risk impact of the plant changes that may result needs to meet the criteria set in the framework document. It has been argued by some that new design-basis accidents (DBAs) may be required because the large break LOCA (LBLOCA) has traditionally been used to bound the impact of other potential accidents. Others argue that formal risk assessment methods should be applied to demonstrate acceptable impacts on risk measures such as core damage frequency (CDF) and large early release frequency (LERF).

2.3.1 How are LOCAs resulting from failures other than pipe breaks considered, and what are these other failures?

2.3.2 The NRC has stated that analysis of LBLOCAs has eliminated the need to analyze other events, which are bounded by LBLOCA. What are the events the NRC considers to be in this category?

2.3.3 Given that smaller LOCAs will still be part of the design basis and these along with other current DBAs can set the system requirements, is there a need to identify alternative design-basis accident(s) for ECCS (e.g. a draindown event), for containment, or for equipment qualification? Why or why not?

2.3.4 If alternative DBAs are needed, should there be a different set of DBAs for ECCS, containment and EQ, or could there be a set applicable to all three? Why or why not?

2.3.5 How would plant changes implemented as a result of LBLOCA redefinition impact external (e.g., seismic) risks, including consideration of indirect LOCAs?

2.3.6 How would plant changes implemented as a result of LBLOCA redefinition impact low power and shutdown risks?

2.4 Potential benefits of LBLOCA redefinition

2.4.1 What are the broadly applicable safety benefits of LBLOCA redefinition?

Background: Safety benefits associated with LBLOCA redefinition that would potentially be realized by all plants include:

- Reallocation of licensee and NRC resources from meeting LBLOCA requirements to better addressing risk-dominant accident types,
- Consistency with respect to the design-basis for dynamic effects, ECCS, containment, and equipment qualification,
- More realism in accident progression analyses. Conservative Appendix K calculations of design-basis LBLOCAs could be eliminated. Treatment of remaining design-basis LOCAs could be based on realistic models. Residual attention to LBLOCAs would be in the context of risk assessments, and could also be based on realistic models but without the need for costly uncertainty quantification.

2.4.2 What are the safety benefits of LBLOCA redefinition that may be realized by all or some specific plants? (Please explain/illustrate each benefit, indicating where the benefit would apply to specific NSSS or containment types.)

Background: Potential safety benefits associated with LBLOCA redefinition that might be realized by some but not all plants include:

- Lengthening the required emergency diesel start time would reduce wear associated with start testing and assist in preserving the reliability of the emergency diesel generators
- Lower ECCS set points might result in reliability improvements for other equipment
- Higher fuel peaking limits would permit fuel configurations that yield less radial neutron leakage thereby reducing the threat of pressurized thermal shock
- It may be possible to better optimize ECCS to deal with risk-dominant accident types (care should be taken to assure that the risk of draindown accidents is not increased).
- The increased time-window for switching to hot-leg injection or recirculation for design-basis accidents might lower failure probabilities associated with these switching actions for risk dominant accidents.

2.4.3 How many plants would be expected to realize each of the potential safety benefits identified in response to the preceding question? (Where relevant, please provide a breakdown by NSSS and/or containment type.)

2.4.4 What criteria are used by licensees to decide on barrel baffle bolt replacement?

2.4.5 What impact would LBLOCA redefinition have on plant life extension?

2.4.6 How many Westinghouse plants would be expected to realize each of the potential unnecessary burden reduction benefits listed in the May 18, 2000, WOG presentation?

2.4.7 What are the potential unnecessary burden reduction benefits of LBLOCA redefinition for B&W and CE plants?

2.5 Impact of LBLOCA redefinition on implementing documents and safety issues

At the public meeting on March 17, 2000, the WOG listed SRP/FSAR Sections, Regulatory Guides, Unresolved Safety Issues, and Generic Safety Issues that could potentially be affected by LBLOCA redefinition.

2.5.1 How does the WOG envision the listed SRP/FSAR Sections, Regulatory Guides, Unresolved Safety Issues, and Generic Safety Issues would be affected by LBLOCA redefinition?

2.5.2 Does the WOG believe that changes to the implementation guidance regarding leak detection (Regulatory Guide 1.145) will be required for LBLOCA redefinition?

2.5.3 What are the staff's plans for developing changes to the implementing regulatory requirements, i.e., Standard Review Plans, Regulatory Guides, etc., necessary to implement the rule change?

3 Other ECCS-Performance Options

3.1 What are the advantages and disadvantages of each of the following other ECCS-performance options:

3.1.1 Relax (risk-inform) simultaneous LOOP assumption
3.1.2 Relax Appendix K conservatisms
3.1.3 Make realistic modeling option less burdensome
3.1.4 Permit uncertainty in break size to be propagated with other uncertainties in the 10 CFR 50.46 best estimate option
3.1.5 Modify (risk-inform) single failure criterion
3.1.6 Relax or eliminate 10 CFR 50.46 reporting requirement
3.1.7 Demonstrate low risk associated with ECCS failure or inadequacy
3.1.8 Develop process for selecting DBLOCA initiators and coincident failures

3.2 What is the feedback from industry, if any, on the need to address known non-conservatisms in 50.46 evaluation models?

3.3 What is the feedback from industry, if any, on the NRC-suggested model selections and values for implementation of the 1979 ANS decay heat standard in Appendix K?

3.4 What is the feedback from industry, if any, on priorities for potential changes to the technical requirements of 10 CFR 50.46, GDC-35 and Appendix K?

3.5 Are there any potential safety or economic benefits to BWRs from a revised risk-informed 10 CFR 50.46, GDC-35 and Appendix K, and what would be the nature of these revisions?

4 Approach and Implementation

4.1 <u>Westinghouse Owner's Group program for LBLOCA redefinition</u>

4.1.1 What are the mechanisms (plant-specific submittals, bounding topical reports, etc.) by which the WOG anticipates obtaining NRC staff approval of leak-before-break (LBB) analyses to cover any piping systems which have not been previously approved for LBB?

4.1.2 What is the WOG schedule for submitting analyses for NRC review? Will the analyses contain proprietary information? If so, what information will be claimed as proprietary?

4.2 <u>NRC program to risk-inform the technical requirements of 10 CFR Part 50 (Option 3)</u>

4.2.1 From previous discussions industry understands that the NRC staff will recommend to the Commission pursuing changes to 10 CFR 50.46. It is assumed that such an activity, and its associated rulemaking, would encompass conforming changes to other regulations, such as Appendix K and Appendix A to Part 50, and not be limited to just ECCS. Are these assumptions correct?

A suggestion has been made that the staff may want to take a step by step approach to 10 CFR 50.46, with redefinition of the LOCA break size being the first step. What are the subsequent steps? Is a plan for 10 CFR 50.46 being included with the recommendations that are being made to the Commission in the June 2001 time-frame? What would trigger the staff to go to each of the next steps, and what would be the overall schedule for getting through to the final rules?

4.2.2 What does the staff plan to recommend to the Commissioners in the June 2001 time-frame with respect to risk-informing 10 CFR Part 50 (RIP50), Option 3?

4.2.3 Is one of the RIP50, Option 3, recommendations to the Commissioners going to include the pursuit of changes to 50.46, such as changing the 1971 decay heat model?

4.2.4 What is the basis for the quantitative guidelines for the Option 3 framework, how are they similar to or different from RG 1.174, and which would take precedence given a plant change related to LBLOCA (e.g., rare initiator frequency < 1E-05/year)?

4.2.5 The Option 3 framework document contains many generalizations. How would this document be specifically applied in attempting to address LBLOCA redefinition?

OPTION 1:

Risk-inform (relax) simultaneous loss of offsite power assumption

PROS:

- This option should result in lengthened DG start times and the associated safety benefit with respect to DG reliability
- Relatively straightforward change to existing regulations and implementing documents
- Likely applicable to PWRs and BWRs

CONS:

- This option would not result in reduced neutron fluence to the reactor vessel and the associated safety benefit with respect to pressurized thermal shock (PTS)
- Aside from lengenthing the DG start time, this option would have little impact on industry-identified areas of potential unnecessary burden

OPTION 2:

Risk-inform (relax) single failure criterion

PROS:

- May permit relaxed technical specifications (e.g., for low pressure injection systems)
- Likely applicable to PWRs and BWRs

CONS:

- Regulatory analysis and resulting changes to regulations and implementing documents would be complicated by fact that single failure criterion has broader scope than LBLOCA
- Significant safety benefits are not assured; that is, it is not clear that this option would result in:
  - lengthened DG start times and associated benefit with respect to DG reliability
  - reduced neutron fluence to reactor vessel and associated benefit with respect to pressurized thermal shock (PTS)
- Significant unnecessary burden reduction is not assured

OPTION 3:

Relax Appendix K conservatisms (specifically, permit the use of a more realistic decay-heat model)

PROS:

- Conceptually straightforward
- Reduced conservatism could enable significant reductions in unnecessary burden for some plants
- Relatively straightforward change to existing regulations and implementing documents

CONS:

- Revised Appendix K models would have to be reviewed and approved by NRC. It may be difficult to demonstrate use of more realistic decay-heat model still compensates for known nonconservatisms in existing Appendix K models.
- Significant safety benefits are not assured; that is, it is not clear that this option would result in:
  - lengthened DG start times and associated benefit with respect to DG reliability
  - reduced neutron fluence to reactor vessel and associated benefit with respect to pressurized thermal shock (PTS)
- The extent of unnecessary burden reduction could be highly plant specific

OPTION 4:

Make use of best-estimate models with uncertainty quantification less burdensome

PROS:

- Only Regulatory Guide 1.157 would need to be revised
- Would encourage development and use of best-estimate analysis methods, which, in the long run, are much more consistent with the philosophy of risk-informed regulation
- Methods exist that could potentially reduce time and costs associated with best-estimate model reviews and best-estimate calculations
- Reduced conservatism could enable significant reductions in unnecessary burden for some plants

CONS:

- Developing, demonstrating, and gaining acceptance of methods to reduce approval/computational costs could be time consuming and resource intensive
- Significant safety benefits are not assured; that is, it is not clear that this option would result in:
  - lengthened DG start times and associated benefit with respect to DG reliability
  - reduced neutron fluence to reactor vessel and associated benefit with respect to pressurized thermal shock (PTS)
- Not all licensees would adopt such methods
- The extent of unnecessary burden reduction could be highly plant specific

OPTION 5:

Propagate uncertainty in break size with other uncertainties when using best-estimate models

PROS:

- Only Regulatory Guide 1.157 would need to be revised
- Might encourage development and use of best-estimate analysis methods, which, in the long run, are much more consistent with the philosophy of risk-informed regulation
- Reduced conservatism could be used to reduce unnecessary licensee burden in a plant-specific manner

CONS:

- Distribution of break size given LBLOCA would have to be developed and defended
- Significant safety benefits are not assured; that is, it is not clear that this option would result in:
  - lengthened DG start times and associated benefit with respect to DG reliability
  - reduced neutron fluence to reactor vessel and associated benefit with respect to pressurized thermal shock (PTS)
- Only ~20 PWRs use best-estimate models with uncertainty quantification

OPTION 6:

Demonstrate low risk associated with removing LBLOCA as a DBA by plant type

- Demonstrate that resulting plant changes could not cause
  - ΔCDF to be a substantial fraction of framework CDF guideline (1E-4/yr)
    OR
  - ΔLERF to be a substantial fraction of framework LERF guideline (1E-5/yr)

NOTES:

- Demonstrating a LBLOCA frequency less than 1E-6/yr may not be sufficient because plant changes could potentially impact frequencies of other core-damage accident classes
- Demonstration would be made as part of NRC regulatory analysis
- Rulemaking would result in changes to 50.46, Appendix K, and GDC
- Industry LBLOCA redefinition efforts could support this option

PROS:

- This option would permit potential safety benefits to be realized:
  - lengthened DG start times and associated benefit to DG reliability
  - reduced neutron fluence to reactor vessel and associated benefit with respect to pressurized thermal shock (PTS)
- Could permit unnecessary burden reduction in all of the areas identified by industry
- Would not require submittal/review of plant-specific CDF, ΔCDF, LERF, and ΔLERF

CONS:

- Framework goals are for full-scope Level-2 CDF and LERF
- Uncertainties and plant-to-plant variabilities must be considered
- Establishing acceptably low ΔCDF and ΔLERF by plant type may be difficult
- Alternative DBAs (e.g., draindown event) might be required

OPTION 7:

Demonstrate low plant-specific risk associated with removing LBLOCA as a DBA

- Demonstrate that resulting plant-specific risk changes (ΔCDF and ΔLERF) are consistent with Regulatory Guide 1.174

PROS:

- This option would permit potential safety benefits to be realized:
  - lengthened DG start times and associated benefit to DG reliability
  - reduced neutron fluence to reactor vessel and associated benefit with respect to pressurized thermal shock (PTS)

- Could permit unnecessary burden reduction in all of the areas identified by industry

CONS:

- Would require submittal/review of plant-specific CDF, ΔCDF, LERF, and ΔLERF as described in RG 1.174

DESIRED INFORMATION FROM REVISED IPE/PRAs ON LARGE BREAK LOCAS

Provide mean parameter estimates along with any uncertainty information for the following parameters where available:

Large Break LOCA Frequency

RCS water breaks:
- Size (e.g., >6 inches in diameter)
- Frequency
- Source of frequency data/method of calculation

RCS steam breaks (BWRs only):
- Size (e.g., >6 inches in diameter)
- Frequency
- Source of frequency data/method of calculation

Large Break LOCA CDF

Random pipe breaks (full-power operation):

- CDF from water breaks
- CDF from steam breaks (BWRs only)

LOCA CDF during low power/shutdown:

- CDF from pipe breaks (for each mode)
- CDF from drain down events (for each mode)

Seismic-induced LOCAs:

- Direct seismic-induced LOCA CDF (i.e., CDF from seismic-induced pipe breaks)
- Indirect seismic-induced LOCA CDF (i.e., CDF from falling objects that result in pipe breaks)

Containment Performance During Large LOCAs

Probability of different containment failure modes following a large LOCA (during full power and low power/shutdown, and during earthquakes):

- Conditional probability of containment bypass or isolation failure
- Conditional probability of early containment failure
- Conditional probability of late containment failure
- Conditional Probability of no containment failure

LERF from large LOCAs:

Definition of LERF
LERF

Results of Level 3 PRA assessment of large break LOCA

Early and latent fatality information

Conditional probability of a loss-of-offsite power during a Large LOCA

Probability (or probabilities used in analyses)
Bases for probabilities

Frequency, CDF and LERF information from other initiators that can cause large losses of primary coolant

For example, interfacing system LOCAs, vessel rupture, multiple open relief valves, and multiple steam generator tube ruptures.

Other

Results of any fracture mechanic evaluations on RCS piping
Data on through wall cracks found in RCS piping

Potential Causes and Frequencies of Loss of Primary Coolant

Source of Loss of Coolant	Frequency Estimates	References	Comments
Reactor Vessel			
Vessel rupture	3E-7/yr	WASH-1400	
CRDH housing failure	-		No frequency estimate available. Too small for LBLOCA.
Head closure seal leakage rupture	9E-5/yr 9E-7/yr	EGG-SSRE-9639 WSRC-TR-93-262	Reflects flange/gasket failure estimates. A rupture could be a LBLOCA.
Failure of instrumentation penetrations	-		No frequency estimate available. Too small for LBLOCA.
Inadvertently-open head vent valve	9E-4/yr	NUREG/CR-4550	Frequency of spuriously-open MOV, two valves may have to open. Too small for LBLOCA.
Pressurizer			
Shell leakage rupture	9E-5/yr 9E-7/yr	EGG-SSRE-9639	Reflects data recommended for a pressurized tank. Estimated based on 12 leakage events (1 in the PCS) and 2 rupture events (0 in PCS). A rupture could be a LBLOCA.
Surge line rupture	-		Assumed included in other PCS piping failure frequency.
Pressurizer spray line rupture	-		Assumed included in other PCS piping failure frequency.
Inadvertently-open PORV	1E-3/yr	NUREG/CR-5750	Too small for LBLOCA.
Stuck-open safety valves 1 SORV (BWR) 1 SORV (PWR) 2 or more SORVs	 5E-3/yr 4.6E-2/yr 3.2E-4/yr	NUREG/CR-5750	Three SORVs would be a LBLOCA.
Failure of instrumentation penetrations	-		No frequency estimate available. Too small for LBLOCA.

Potential Causes and Frequencies of Loss of Primary Coolant

Source of Loss of Coolant	Frequency Estimates	References	Comments
Steam Generator			
Single tube rupture Multiple tube ruptures	7E-3/yr -	NUREG/CR-5750	Based on 3 reported events. Multiple SGTRs would result in a LBLOCA. No frequency estimate identified.
Shell leakage rupture	9E-5/yr 9E-7/yr	EGG-SSRE-9639	Reflects estimates for heat exchangers. Estimates based on 2 leakage events and 0 rupture events in non-PCS HTXs. A rupture could be a LBLOCA.
Manway failure	-		No frequency estimate available.
Failure of instrumentation penetrations	-		No frequency estimate available. Too small for LBLOCA.
Reactor Coolant Loops			
Pump casing leakage rupture	3E-4/yr 3E-6/yr	EGG-SSRE-9639	Based on 50 reported incidences of external leakage (4 in PCS) and 2 rupture events (0 in PCS). A rupture could be a LBLOCA.
RCS pump seal failure	2.5E-3/yr	NUREG/CR-5750	Too small for LBLOCA.
Valve body leakage rupture	9E-5/yr 9E-7/yr	EGG-SSRE-9639	Estimated based on 170 incidences of reported leakage (29 in PCS) and 7 rupture events (none in the PCS). A rupture could be a LBLOCA.
Large pipe rupture BWR PWR	 2E-5/yr 4E-6/yr	NUREG/CR-5750	Frequencies are for BWR pipe sizes >5 inches in diameter and PWR pipe sizes >6 inches in diameter.
Accumulator leakage rupture	9E-5/yr 9E-7/yr	EGG-SSRE-9639	Reflects data recommended for a pressurized tank. Estimated based on 12 leakage events (1 in the PCS) and 2 rupture events (0 in PCS). Valve failures would also have to occur to result in loss of primary coolant.
Failure of instrumentation penetrations	-		No frequency estimate available. Too small for LBLOCA.

Potential Causes and Frequencies of Loss of Primary Coolant

Source of Loss of Coolant	Frequency Estimates	References	Comments
Interfacing System LOCAs			
BWRs	9.6E-4/yr	NUREG/CR-5750	This is the mean CDF of the distribution of IPE point estimates. Not all ISLOCAs may be LBLOCAs.
PWRs	5.1E-5/yr	NUREG/CR-5750	This is the mean CDF of the distribution of IPE point estimates. Not all ISLOCAs may be LBLOCAs.
Low Power/Shutdown Events			
Draindown events	6.1E-2/yr	NUREG/CR-5593	Frequency is for a BWR. The frequency does not include the fraction of time the plant is shutdown.
Inadvertent overpressurization (makeup greater than letdown or spurious ECCS actuation)	1.6E-3/yr 1.4E-2/yr	NUREG/CR-5593	Frequency is for a BWR. Applicable for cold shutdown only. The frequency does not include the fraction of time the plant is in cold shutdown.
ISLOCA (RHR)	1.6E-2/yr	NUREG/CR-5593	Frequency is for a BWR. The frequency does not include the fraction of time the plant is shutdown.

QUANTITATIVE GUIDELINES

Based on Commission's Safety Goals

— Quantitative Health Objectives —

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

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

	Limit frequency of accident initiating events	Limit probability of core damage given event	Limit radionuclide releases given core damage	Limit public health effects given release
	Initiator Frequency	Conditional core damage probability	Conditional large release probability	Conditional individual fatality probability
Frequent Initiators	≥ 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.

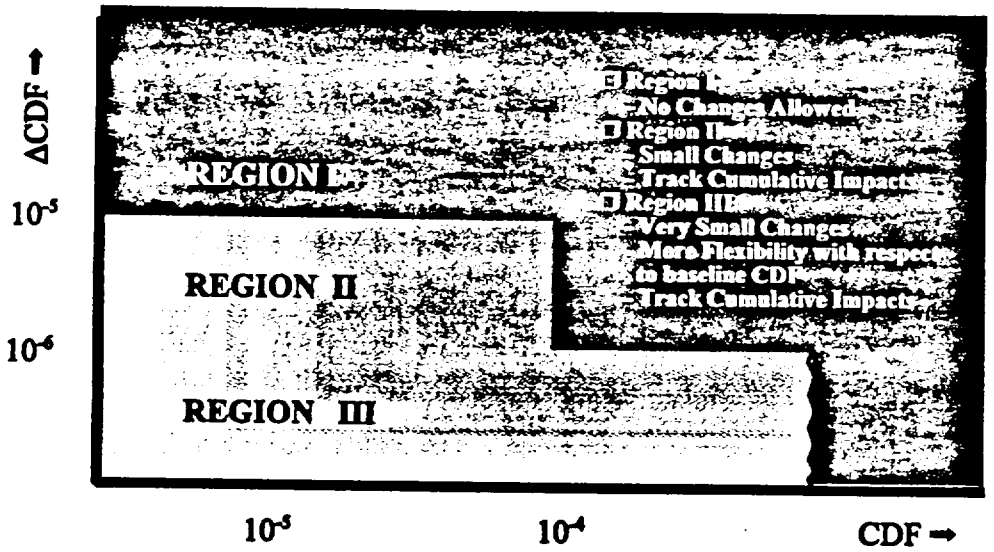


Figure 3 - Acceptance Guidelines* for Core Damage Frequency (CDF)

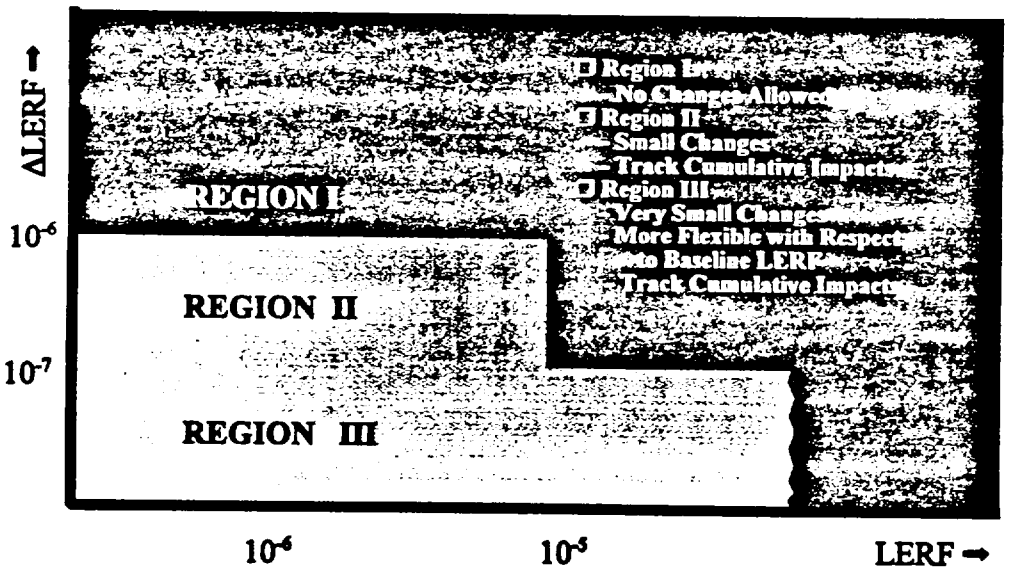


Figure 4 - Acceptance Guidelines* for Large Early Release Frequency (LERF)

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

Part 2
Risk Significance of LOCAs and ECCS

RCS PIPE BREAK LOCA FREQUENCIES

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

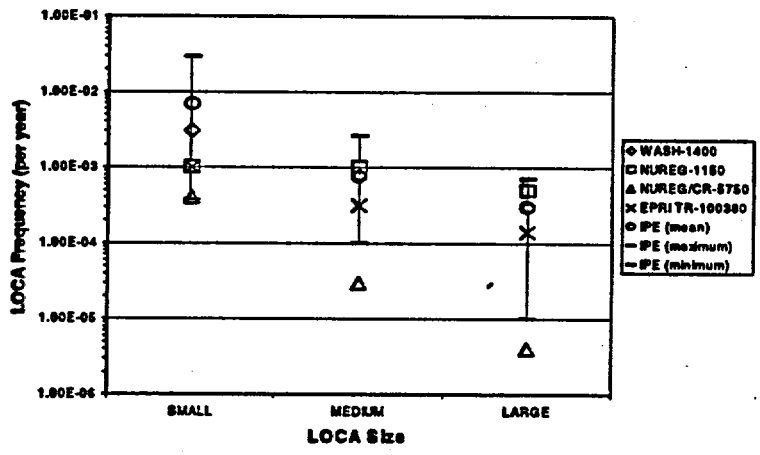
NUREG/CR-5750 METHODS FOR ESTIMATING PIPE-BREAK LOCA FREQUENCIES

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

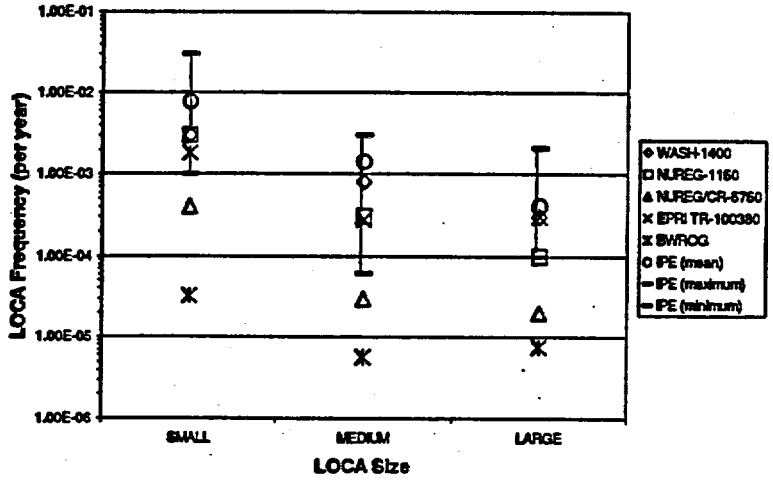
THROUGH-WALL CRACK DATA (NUREG/CR-5750)

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

PWR Pipe Break LOCA Frequencies



BWR Pipe Break LOCA Frequencies



DOUBLE-ENDED GUILLOTINE BREAKS (DEGB)

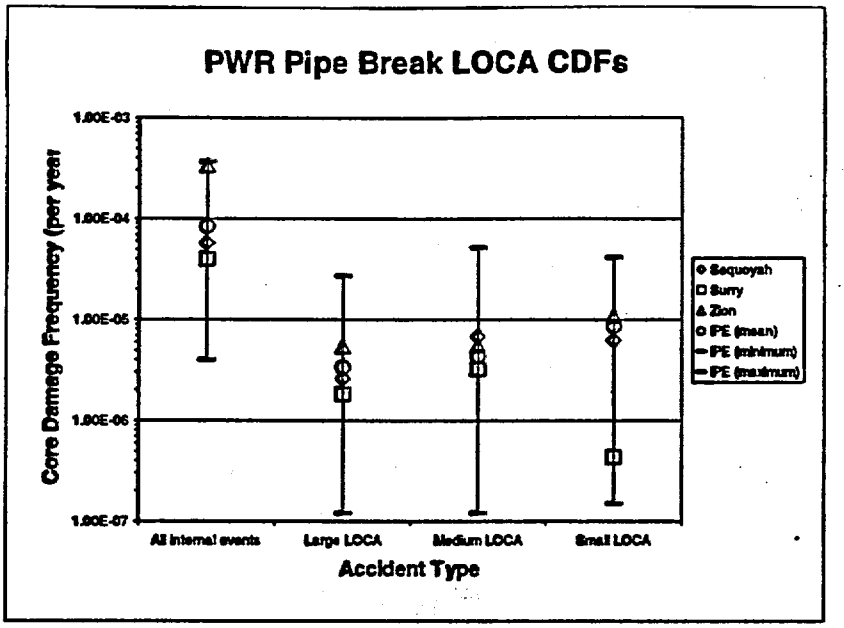
- Frequency of DEGBs have been estimated using fracture mechanics as part of LBB assessment (NUREG-1061 and NUREG/CR-4792)
- For CE plants,
  - Point estimates range from 6E-14/yr to 5E-13/yr
  - 90 percentile values range from 4E-12/yr to 7E-11
- For Westinghouse plants,
  - Median frequencies range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

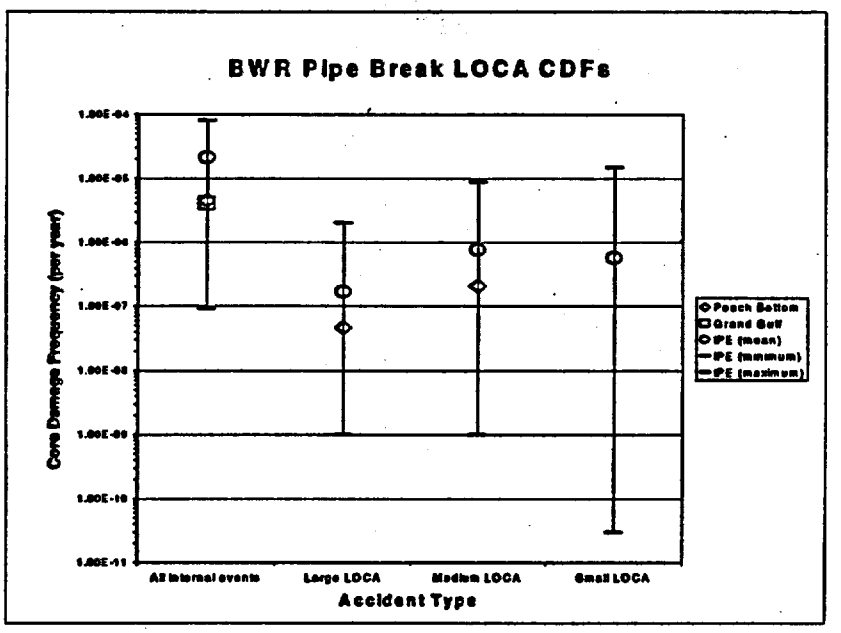
SEISMIC-INDUCED LOCA FREQUENCIES

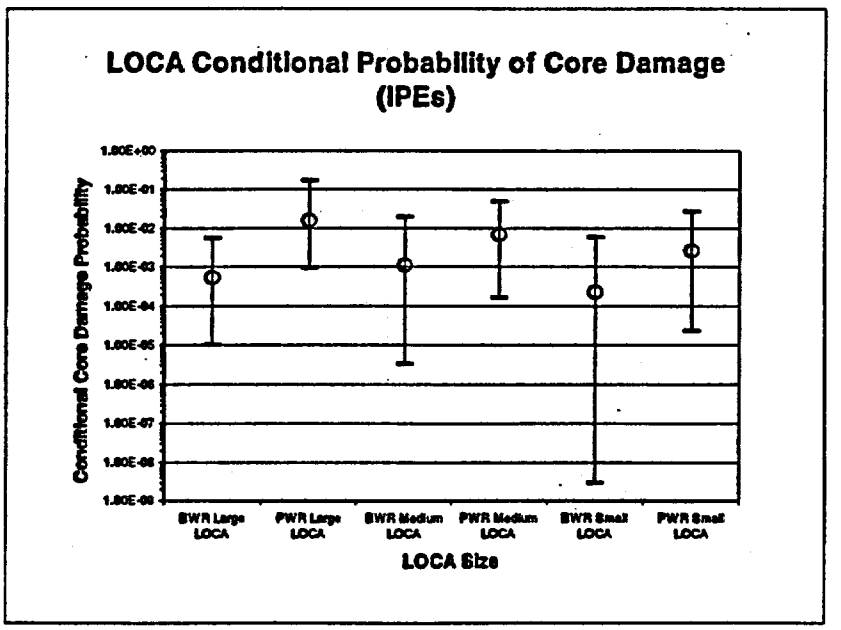
- Estimates using fracture mechanics provided in LBB analysis in NUREG/CR-3660 and NUREG/CR-3663
  - Frequency of direct seismic-induced pipe break LOCAs are 1 to 3 orders of magnitude lower than random LOCAs
  - Frequency of indirect seismic-induced DEGBs was significantly higher than frequency of direct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

LOCA CONTRIBUTION TO CDF

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


PWR Pipe Break LOCA CDFs
Core Damage Frequency (per year)
1.00E-03
1.00E-04
1.00E-05
1.00E-06
1.00E-07
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)
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)
Conditional Core Damage Probability
1.00E+00
1.00E-01
1.00E-02
1.00E-03
1.00E-04
1.00E-05
1.00E-06
1.00E-07
1.00E-08
1.00E-09
BWR Large LOCA
PWR Large LOCA
BWR Medium LOCA
PWR Medium LOCA
BWR Small LOCA
PWR Small LOCA
LOCA Size

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	8.1E-02	1.3E-07	1.3E-08	7.6E-09
LOCA in RHR	1.6E-02	2.1E-08	4.2E-07	8.7E-07
Total		1.3E-06	4.3E-07	3.8E-08
Surry				
Large LOCA		2.5E-6		

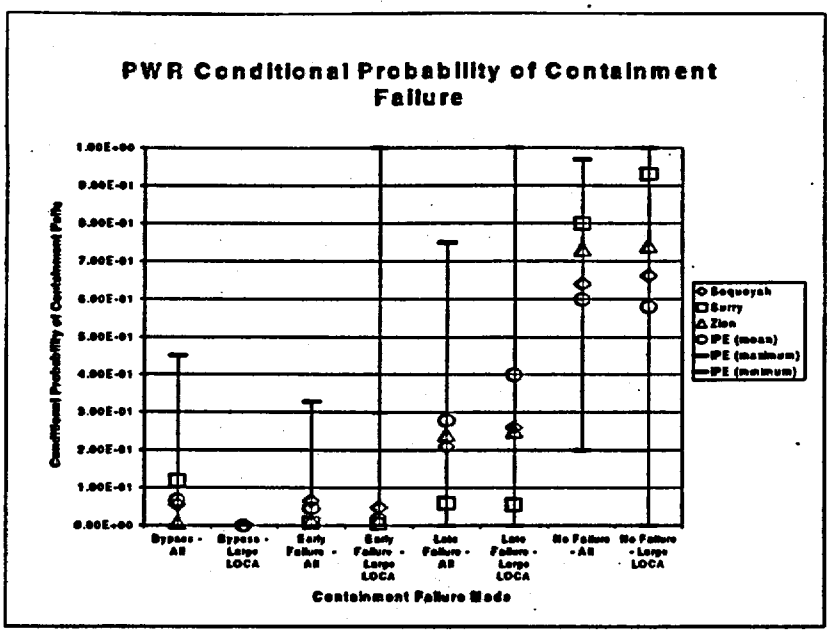
CDF FROM SEISMIC-INDUCED LOCAs

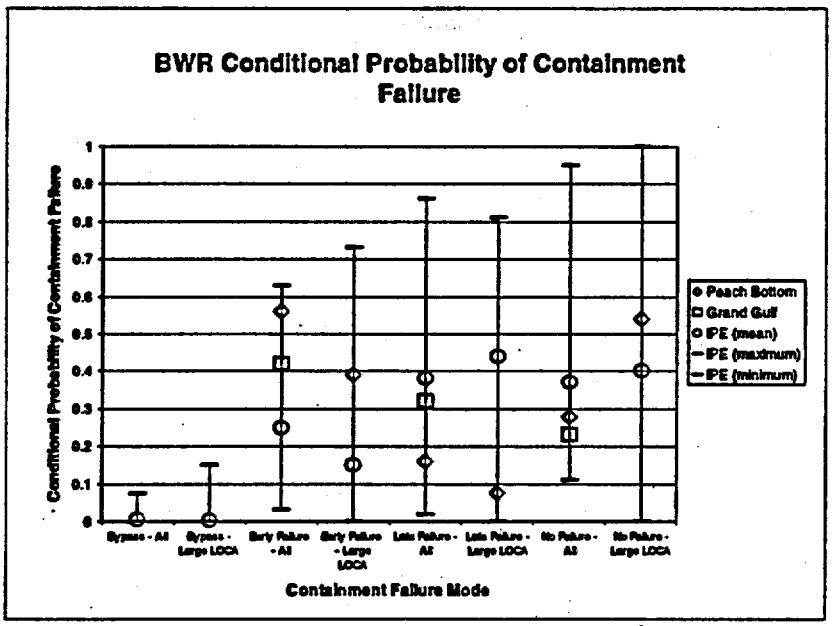
Accident Type	Mean Core Damage Frequency	
	LLNL Hazard Curve	EPRI Hazard Curve
Large LOCA		
Peach Bottom (NUREG-1150)	1.9E-05	6.8E-07
Surry (NUREG-1150)	7.7E-06	1.3E-06
Medium LOCA		
Peach Bottom (NUREG-1150)	7.4E-06	2.1E-07
Surry (NUREG-1150)	1.5E-06	1.7E-07
Small LOCA		
Peach Bottom (NUREG-1150)	1.5E-06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





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