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Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process

Draft Report for Comment

U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Washington, DC 20555-0001

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ABSTRACT

The emergency core cooling system (ECCS) requirements are contained in 10 CFR 50.46, Appendix K to part 50, and GDC 35. Consideration of an instantaneous break with a flow rate equivalent to a double-ended guillotine break (DEGB) of the largest primary system in the plant generally provides the limiting condition in the required ECCS analysis. However, the DEGB is widely recognized as an extremely unlikely event. Therefore, the NRC is developing a risk-informed revision of the design-basis break size requirements for operating commercial nuclear power plants. A central consideration in selecting a risk-informed design basis break size is an understanding of the LOCA frequency as a function of break size.

LOCA frequency estimates have been developed using an expert elicitation process to consolidate service history data and insights from PFM studies with knowledge of plant design, operation, and material performance. This elicitation process is well-recognized for quantifying phenomenological knowledge when data or modeling approaches are insufficient. Separate BWR and PWR piping and non-piping passive system LOCA frequency estimates have been developed as a function of effective break size and operating time through the end of license extension. The elicitation focused solely on determining event frequencies that initiate by unisolable primary system side failures that can be exacerbated by material degradation with age. The expert elicitation process employed in this study is an adaptation of the formal expert judgment process used in NUREG-1150. This current elicitation process included the decomposition of the complex technical issues which impact LOCA frequencies into fundamental elements in order to more easily assess these important contributing factors. The elicitation process required each member of the elicitation panel to qualitatively and quantitatively assess these LOCA contributing factors and also indicate their uncertainty in this assessment. This information was collected from each panelist in an individual elicitation session.

The qualitative insights provided by the panel members are reasonably consistent. Most panelists agreed that a complete break of a smaller pipe, or non-piping component, is more likely than an equivalent size opening in a larger pipe, or component. Many panelists thought that aging may have the greatest effect on intermediate diameter (6 to 14-inch diameter) piping systems due to the large number of components within this size range and the fact that this piping generally receives less attention than larger diameter piping and is harder to replace than the more degradation-prone smaller diameter piping.

Frequency estimates are not expected to change dramatically over the next fifteen years, or even the next thirty-five years. While aging will continue, the consensus is that mitigation procedures are in place, or will be implemented in a timely manner, to alleviate possible LOCA frequency increases.

The quantitative responses were analyzed separately for each panel member to develop individual BWR and PWR total LOCA frequency estimates of the mean, median, 5th and 95th percentiles. The LOCA frequencies for the individual panelists were then aggregated to obtain group LOCA frequency estimates, along with measures of panel diversity. While there was general qualitative agreement among the panelists about important technical issues and LOCA contributing factors, the individual quantitative estimates are much more variable. Additionally, as the LOCA size increased, the panel members generally expressed greater uncertainty in their predictions, and the variability among individual panelists' estimates increased. Both trends are expected given the underlying scientific uncertainty.

The elicitation LOCA frequency estimates are generally much less than the prior WASH-1400 estimates and more consistent with the NUREG/CR-5750 estimates. The small break (SB) LOCA frequency estimates are similar once the steam generator tube rupture frequencies are added to the NUREG/CR-5750 PWR results. The elicitation medium break LOCA estimates are higher than the NUREG/CR-5750 estimates by factors of approximately 4 and 20 for BWR and PWR plant types, respectively. These increases are partly due to PWSCC of piping and non-piping (CRDM) components, as well as the general aging concerns with piping in this size range. The NUREG/CR-5750 LB LOCA frequency estimates, most comparable to the elicitation LOCA Category 4, tend to be slightly higher (approximately a factor of 3) than the current elicitation results.

Sensitivity analyses were conducted to examine the robustness of the quantitative results to the analysis procedure. These sensitivity analyses investigated the effect of distribution shape on the mean, correlation structure, panelist overconfidence, panel diversity measure, and aggregation method on the estimated parameters. The mean calculation used a split lognormal distribution truncated at the 99.9th percentile to obtain reasonably conservative values. The correlation structure assumed maximal correlation, which is reasonably representative of the elicitation structure and provides conservative 95th percentile estimates. However, based on selected Monte Carlo simulations, assuming an independent correlation structure results in larger median and 5th percentile estimates. The means are unaffected by the correlation structure. The analysis procedure adjusted some panelists' responses that had relatively narrow uncertainty ranges to account for a known tendency for people to be overconfident when making subjective judgments. Sensitivity analyses examined the effects of other overconfidence adjustments as well as no adjustment of the nominal subjective confidence levels supplied by the panelists. While blanket overconfidence adjustments can result in large, unsupportable increases in the frequency estimates, no adjustment results in modest decreases in the mean and 95th percentile estimates. The analysis procedure used confidence intervals for the aggregated estimates as a measure of panel diversity to reflect variability in the individual estimates. An alternative approach used quartiles of the individual estimates, leading to comparable but narrower intervals.

Finally, there is a large sensitivity to the method used to aggregate the individual panelist estimates to obtain group estimates. A geometric mean aggregation, rather than an arithmetic mean or mixture distribution aggregation, was chosen in this case because it was a better representation of the consensus-type group estimate sought by this study.

There was a wide range of individual responses for the LOCA frequency expert elicitation. Use of the arithmetic mean or mixture distribution aggregation method in this case would produce group estimates which are dominated by the one or two highest individual estimates and would not represent a consensus-type group estimate, which was sought for this study.

For this study, the geometric mean aggregation develops frequency estimates which are a better representation of consensus-type results and better express the expert panel's state of knowledge regarding LOCA frequencies in a manner consistent with the elicitation process used to obtain the panel responses, the nature of these responses and the stated study objectives.

FOREWORD

Loss-of-coolant accident (LOCA) frequencies (i.e., the estimated frequencies of pipe rupture as a function of pipe size) are used in a variety of regulatory applications, including probabilistic risk assessment (PRA). Currently, the U.S. Nuclear Regulatory Commission (NRC) is using such information to develop a risk-informed alternative to the emergency core cooling system (ECCS) requirements in Title 10, Section 50.46, of the *Code of Federal Regulations* (10 CFR 50.46). Current requirements consider pipe breaks in the reactor coolant pressure boundary, up to and including breaks equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system. One aspect of this activity is to evaluate the technical adequacy of redefining the design-basis break size (the largest pipe size described in 10 CFR 50.46) to a smaller size that is consistent with the estimated frequency of pipe failures as a function of pipe size.

To provide the technical basis for a risk-informed definition of the design-basis break size, this study developed LOCA frequency estimates using an expert elicitation process. This process consolidated service history data and insights from probabilistic fracture mechanics studies with knowledge of plant design, operation, and material performance. This elicitation process is well-recognized for quantifying phenomenological knowledge when modeling approaches or data are insufficient. The process used in this study is an adaptation of the formal expert judgment process used in NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," dated December 1990.

The results from the expert elicitation provide separate LOCA frequency estimates for piping and non-piping passive systems, as a function of effective break size and operating time through the end of license extension, for both boiling-water and pressurized-water reactors. In addition, this study considered the sensitivity of the results to various analysis approaches. The greatest sensitivity, and therefore the greatest uncertainty, is a function of the method used to aggregate the individual panelists' estimates to obtain group estimates. The range of results from the sensitivity analyses have been used as a baseline for defining the transition break size in the proposed risk-informed alternative to 10 CFR 50.46.

Although the NRC initiated this report (and the underlying expert elicitation) to support a risk-informed alternative to 10 CFR 50.46, the data provided in this report may be useful for other activities requiring LOCA frequency estimates. Other activities should first consider the applicability of the data to the purposes and context of the specific application. Although this places an additional burden on users, those users are in the best position to assess the extent to which the study results can be used for their particular applications.

The NRC is publicly releasing this draft report to facilitate public comment. Interested parties should review the related *Federal Register* notice to submit comments in a timely manner. After resolving the public comments, the NRC will publish a revised final report.

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EXECUTIVE SUMMARY

The emergency core cooling system (ECCS) requirements are contained in 10 CFR 50.46, Appendix K to part 50, and GDC 35. Specifically, ECCS design, reliability, and operating requirements exist to ensure that the system can successfully mitigate postulated loss-of-coolant accidents (LOCAs). Consideration of an instantaneous break with a flow rate equivalent to a double-ended guillotine break (DEGB) of the largest primary piping system in the plant generally provides the limiting condition in the required 10 CFR Part 50, Appendix K analysis. However, the DEGB is widely recognized as an extremely unlikely event. Therefore, the staff is performing a risk-informed revision of the design-basis break size requirements for operating commercial nuclear power plants.

A central consideration in selecting a risk-informed design basis break size is an understanding of the LOCA frequency as a function of break size. The most recent NRC-sponsored study of pipe break failure frequencies is contained in NUREG/CR-5750. Unfortunately, these estimates are not sufficient for design basis break size selection because they do not address all current passive-system degradation concerns and they do not discriminate among breaks having effective diameters greater than 6 inch. There have been two approaches traditionally used to assess LOCA frequencies and their relationship to pipe size: (i) estimates based on statistical analysis of service experience data and (ii) probabilistic fracture mechanics (PFM) analysis of specific postulated failure mechanisms. Neither approach is particularly suited to evaluate LOCA event frequencies due to the rareness of these events and the modeling complexity. In this study, LOCA frequency estimates have been developed using an expert elicitation process to consolidate service history data and insights from PFM studies with knowledge of plant design, operation, and material performance. This process is well-recognized for quantifying phenomenological knowledge when data or modeling approaches are insufficient.

The objective of this study was to develop separate BWR and PWR piping and non-piping passive system LOCA frequency estimates as a function of effective break size and operating time through the end of license extension. The evaluation considered three distinct time periods: the current day, the end of plant licensing, and the end of license extension. The elicitation focused solely on determining event frequencies that initiate by unisolable primary system side failures that can be exacerbated by material degradation with age. The effects of safety culture on LOCA frequencies were also evaluated. Central LOCA frequency estimates are the primary quantitative objective sought by the elicitation process so that a consensus type result is achieved. However, quantifying the variability among the individual panel members is also an important objective.

The elicitation primarily assumed normal plant operational cycles and loading histories consistent with current internal event plant risk assessment (PRA) models. The loads include representative constant stresses (e.g. pressure, thermal, residual) and expected transient stresses (e.g. thermal striping, heat-up/cool-down, pressure transients) that occur over the extended licensing period. The effect of rarer transients, such as a large seismic event, on the LOCA frequencies was not considered directly as part of this effort. The elicitation implicitly considered all modes of operation per calendar year based on the loading or operational history associated with each piping system or non-piping component.

An important assumption implicit in the elicitation was that the plant construction and operation are in accordance with applicable codes and standards. In addition, plant operation, inspection, and maintenance were generally assumed to occur within the expected parameters allowable by the regulations and the technical specifications. Deviations from these practices, however, do represent some percentage of the events included in the operating experience data and extrapolation of this data implicitly assumes that similar future deviations will occur with similar periodicity. Also, future plant operating practice was assumed to be essentially consistent with past operating practice. The effects of operating profile changes were not considered because of the large uncertainty surrounding possible operational changes and the potentially wide-ranging ramifications.

The elicitation focused on developing generic, or average, values for the commercial fleet. Because the fleet average is not significantly affected by a few outliers, only factors which affect a large number of plants can significantly affect the average. Consequently, the panelists were instructed to account for broad plant-specific factors which influence the generic LOCA frequencies in providing uncertainty bounds, but not consider factors specific to any individual plants. Thus, the uncertainty bounds do not represent LOCA frequency estimates for individual plants that deviate from the generic values, i.e., they do not reflect plant-to-plant variability. This study partitioned LOCA sizes into three smaller categories which are consistent with historical small break (SB), medium break (MB), and large break (LB) flow rate definitions. Additionally, three larger LOCA categories were defined within the classical LB LOCA regime to examine trends with increasing break size, up to and including, a DEGB of the largest piping system in the plant.

The expert elicitation process employed in this study is an adaptation of the formal expert judgment processes used in NUREG-1150, to develop seismic hazard curves, and to assess the performance of radioactive waste repositories. The process consisted of a number of steps. To begin, the project staff identified the issues to be evaluated through a pilot elicitation. The panel members for this pilot elicitation were all NRC staff. A panel of twelve panel members was then selected for the formal elicitation. The staff gathered background material and prepared an initial formulation of the technical issues which was provided to the panel. At its initial meeting, the panel discussed the issues and, using the staff formulation as a starting point, developed a final formulation for the elicitation structure. This structure included the decomposition of the complex technical issues which impact LOCA frequencies into fundamental elements in order to more easily assess these important contributing factors. Piping and non-piping base cases were also defined for use in anchoring quantitative elicitation responses. The base cases represent a set of well-defined conditions which could cause a LOCA. A subset of the panel was created to develop quantitative estimates of LOCA frequencies associated with the base case conditions. At this initial meeting, the panel was also trained in subjective elicitation of numerical values through exercises and discussion of potential biases.

After this initial meeting, the staff prepared a draft elicitation questionnaire and iterated with the panel to obtain a final questionnaire. The panelists quantifying the base case conditions also developed initial estimates. A second meeting was held with the entire panel to review the base case results, review the elicitation questions, and finalize the formulation of remaining technical issues. At their home institutions, the individual panel members performed analyses and computations to develop answers to the elicitation questionnaire.

The elicitation questionnaire required panelists to assess the following technical areas: the base case evaluation effort, utility and regulatory safety culture effects on LOCA frequencies, piping system LOCA frequencies, and non-piping system LOCA frequencies. The utility and safety culture questions required the panelists to compare future safety culture with the existing culture and predict the effect on LOCA frequencies. These effects were considered separately from passive system degradation because the panelist judged these effects to be independent. The base case evaluation required the panelists to assess the accuracy and uncertainty in the base case analyses, and to also choose a particular base case approach for anchoring their elicitation responses. The piping and non-piping LOCA frequency questions required each panelist to first identify important LOCA contributing factors (i.e., piping systems, materials, degradation mechanisms, etc.) and select appropriate base case conditions for comparison. The panelists were then required to provide the relative ratios between their important contributing factors and the base case conditions based on their knowledge of passive system component failure. Each relative comparison required mid value, upper bound, and lower bound values. The mid value is defined such that, in the panelist's judgment, there is a 50% chance that the unknown true answer lays above the mid value. The upper and lower bounds are defined such that there is a 5% chance that the true answer lays above the upper bound or below the lower bound, respectively. Each panelist was also required to provide their qualitative rationale supporting their quantitative values.

A facilitation team consisting of substantive members, a normative member and two recorders met separately with each panel member in day-long individual elicitation sessions. At these sessions, each panel member provided answers to the elicitation questionnaire along with their supporting technical rationales. The panel members then returned to their home institutions where they refined their responses based on feedback from their elicitation session. (The detailed quantitative responses to the elicitation questions from each of the individual members of the elicitation panel are available with this report at: http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1829.) Upon receipt of the updated responses, the project staff compiled the panel's responses and developed preliminary estimates of the LOCA frequencies. Along with the rationales, these preliminary estimates were presented to the panel at a wrap-up meeting. Panel members were invited to fill in gaps in their questionnaire responses and, if desired, to modify any of their responses based on group discussion of important technical issues considered during the individual elicitations. Final estimates of the LOCA frequencies were then calculated and provided to the panel members for final review and quality assurance.

The qualitative insights provided by the panel are reasonably consistent. The panel members generally believe that the future safety culture will not differ dramatically from the current culture. Many panelists do believe that the effects of safety culture are cyclical and that safety culture can significantly affect LOCA frequencies at specific plants. However, these plant-specific issues do not affect the generic averages and it was determined that no specific adjustment to the LOCA frequency estimates would be required to explicitly account for safety culture effects. Many participants believe that the number of precursor events (e.g., cracks and leaks) is generally a good barometer of the LOCA susceptibility for the associated degradation mechanism. Welds are almost universally recognized as likely failure locations due to high residual stress, preferential attack of many mechanisms, and the increased defect likelihood.

Most panelists also agreed that a complete break of a smaller pipe, or non-piping component, is generally more likely than an equivalent size opening in a larger pipe, or component, because of the increased severity of fabrication or service cracking. Therefore, the biggest frequency contributors for each LOCA size tend to be systems having the smallest pipes, or component, which can lead to that size LOCA. The exception to this general rule is the BWR recirculation system, which is important at all LOCA sizes due to lingering IGSCC concerns. Many panelists thought that aging may have the greatest effect on intermediate diameter (6 to 14-inch diameter) piping systems due to the large number of components within this size range and the fact that this piping generally receives less attention than the larger diameter piping and is harder to replace than the more degradation-prone smaller diameter piping.

The participants generally identified thermal fatigue, stress corrosion cracking (SCC), flow accelerated corrosion (FAC), and mechanical fatigue as the degradation mechanisms that most significantly contribute to LOCA frequencies in BWR plants. Generally, the most important BWR degradation mechanism is intergranular SCC (IGSCC), although the panelist's recognize that mitigation has greatly reduced the susceptibility of BWR plants to this mechanism over the past 20 years. With the exception of FAC, similar degradation mechanisms and concerns were also deemed to be important in PWR plants. Specifically, primary water SCC (PWSCC) is a principal concern. Many panelists believe that PWSCC will be mitigated in PWR plants within the next 15 years, but effective mitigation has yet to be developed and implemented.

The quantitative responses were analyzed separately for each panel member to develop individual BWR and PWR total LOCA frequency estimates. Experts self-censored their responses to reflect their expertise and several only provided inputs for one plant type. A unified analysis format was developed to ensure consistency and commonalty in processing the panelists' inputs. The panelists' mid-value and upper bound and lower bound estimates were assumed to represent the median, 95th, and 5th percentiles, respectively, of their subjective uncertainty distributions for each elicitation response. The analysis structure is based on the assumption that all the responses correspond to percentiles of lognormal distributions. These distributions were then combined using a lognormal framework. The final output for each panelist is BWR and PWR-specific total LOCA frequency estimates of the mean, median, 5th and 95th percentiles.

While there was general qualitative agreement among the panelists about important technical issues and LOCA contributing factors, the individual quantitative estimates are much more variable. This is to be expected given the underlying scientific uncertainty. Estimated LOCA frequencies for the individual panelists were determined and then aggregated to obtain group LOCA frequency estimates, along with measures of panel diversity. The individual and group estimates for the means, medians and 5th and 95th percentiles of the LOCA frequency distributions were determined using the following six assumptions and choices:

(i) The mid-value, upper bound, and lower bound supplied by the panelists for each elicitation question are assumed to correspond to the median, 95th percentile and 5th percentile, respectively, of a split lognormal distribution, with the mean calculated assuming that the upper tail is truncated at the 99.9th percentile.

(ii) Some panelist estimates are adjusted to account for possible overconfidence in the elicited uncertainty ranges for the elicitation responses using an error factor adjustment scheme (Section 7.6.2.2).

(iii) Split lognormal distributions are summed by assuming perfect rank correlation among the individual terms.

(iv) Aggregation of the individual estimates into a group estimate is performed using the total LOCA frequency estimates determined for each panelist.

(v) The group estimate of the total LOCA frequency parameters (i.e., median, mean, 5th percentile, and 95th percentile) is defined using the geometric mean of the individual estimates.

(vi) Panel diversity is characterized by using a two-sided 95% confidence interval based on an assumed lognormal model for the individual estimates.

The resultant individual and group estimates are consistent with the elicitation objectives and structure and are reasonably representative of the panelists' quantitative judgments. In particular, they are not dominated by extreme results, either on the high or low end. In the report, the six assumptions and choices above define what are termed the *baseline* estimates, with one important difference. Instead of the overconfidence adjustment in (ii), the baseline estimates are determined without any such adjustment. However, the adjusted individual LOCA frequency estimates using (ii) above are deemed to result in improved group LOCA frequency estimates. These improved estimates will be referred to as *summary* estimates.

These LOCA frequency estimates for the current day and end of original license period are provided in Table 1 for both BWR and PWR plant types. The aggregated group estimates for the median, mean, 5th and 95th percentiles are summarized. Frequency estimates are not expected to change dramatically over the next fifteen years for any size LOCA, or even the next thirty-five years for LOCA Category 4 and smaller (see results in Appendix L). An order of magnitude increase in Category 5 and 6 LOCA frequencies is predicted over the next 35 years, but this increase largely stems from current uncertainty about future conditions. While aging will continue, the consensus is that mitigation procedures are in place, or will be implemented in a timely manner, to alleviate possible LOCA frequency increases. The current day median, mean, and 95th percentile *summary* estimates are graphically presented in Figure 1. The 95% confidence intervals calculated for these parameters are also illustrated in this figure. A measure of the individual uncertainty in Table 1 and Figure 1 is given by the difference between the median and 5th or 95th percentile estimates. Panel diversity is reflected in the confidence bounds in Figure 1. As the LOCA size increased, the panel members generally expressed greater uncertainty in their predictions, and the variability among individual panelists' estimates increased. This is to be expected because of the greater extrapolation required from available service data.

The elicitation LOCA frequency estimates are generally much lower than the prior WASH-1400 estimates and more consistent with the NUREG/CR-5750 estimates. The small break (SB) LOCA frequency estimates are similar once the steam generator tube rupture frequencies are added to the NUREG/CR-5750 PWR results. The elicitation medium break LOCA estimates are higher than the NUREG/CR-5750 estimates by factors of approximately 4 and 20 for BWR and PWR plant types, respectively. These increases are partly due concerns for PWSCC of piping and non-piping (CRDM) components as well as the general aging concerns with piping in this size range. The NUREG/CR-5750 LB LOCA frequency estimates are most comparable to the elicitation LOCA Category 4 estimates which tend to be slightly higher (less than a factor of 3) than the current elicitation results. The generally good agreement between the NUREG/CR-5750 and current elicitation estimates is somewhat surprising given the markedly different methodologies used to arrive at these results.

Table 1 Total BWR and PWR LOCA Frequencies (After Overconfidence Adjustment using Error Factor Scheme)

Plant Type	LOCA Size (GPM)	Eff. Break Size (inch)	Current Day Estimate (per cal. year) (25 yr fleet average operation)				Estimate at End of Plant License (per cal. yr.) (40 yr fleet average operation)			
			5th Per.	Median	Mean	95th Per.	5th Per.	Median	Mean	95th Per.
BWR	>100	½	3.1E-05	3.0E-04	6.4E-04	2.1E-03	2.6E-05	2.6E-04	6.0E-04	2.0E-03
	>1,500	1 7/8	2.7E-06	4.8E-05	1.2E-04	4.1E-04	2.2E-06	4.4E-05	1.1E-04	4.1E-04
	>5,000	3 ¼	5.6E-07	9.7E-06	2.8E-05	1.0E-04	4.9E-07	9.8E-06	3.2E-05	1.2E-04
	>25K	7	9.6E-08	2.2E-06	7.3E-06	2.7E-05	8.7E-08	2.3E-06	9.3E-06	3.4E-05
	>100K	18	7.2E-09	2.9E-07	1.5E-06	5.4E-06	6.2E-09	3.1E-07	2.1E-06	7.3E-06
	>500K	41	5.6E-12	3.0E-10	6.4E-09	1.6E-08	6.7E-12	4.0E-10	1.0E-08	2.5E-08
PWR	>100	½	6.0E-04	3.7E-03	6.4E-03	1.8E-02	3.5E-04	2.5E-03	4.7E-03	1.4E-02
	>1,500	1 5/8	7.0E-06	1.4E-04	6.2E-04	2.2E-03	7.6E-06	1.6E-04	7.6E-04	2.7E-03
	>5,000	3	2.0E-07	3.4E-06	1.6E-05	5.8E-05	4.5E-07	7.6E-06	3.6E-05	1.3E-04
	>25K	7	1.3E-08	3.1E-07	1.6E-06	5.7E-06	2.6E-08	6.5E-07	3.6E-06	1.3E-05
	>100K	14	3.8E-10	1.1E-08	1.9E-07	5.2E-07	9.2E-10	2.7E-08	4.6E-07	1.3E-06
	>500K	31	3.3E-11	1.2E-09	3.1E-08	7.8E-08	8.2E-11	2.9E-09	8.1E-08	2.0E-07

Sensitivity analyses were conducted to examine the robustness of the quantitative results to the analysis procedure. These sensitivity analyses investigated the effect of distribution shape on the means, and the effects of correlation structure, panelist overconfidence, panel diversity measure, and aggregation method on the estimated parameters. The mean calculation used a split lognormal distribution truncated at the 99.9th percentile to obtain reasonably conservative values. The correlation structure assumed maximal correlation, which is reasonably representative of the elicitation structure and provides conservative 95th percentile estimates. However, based on selected Monte Carlo simulations, assuming an independent correlation structure results in larger median and 5th percentile estimates (the means are unaffected). The analysis procedure adjusted some panelists' responses that had relatively narrow uncertainty bands to account for a known tendency for people to be overconfident when making subjective judgments. Sensitivity analyses examined the effects of other overconfidence adjustments as well as no adjustment of the nominal subjective confidence levels supplied by the panelists. While blanket overconfidence adjustments can result in large, unsupportable increases in the frequency estimates, no adjustment results in modest decreases in the mean and 95th percentile estimates. The analysis procedure used confidence intervals for the aggregated estimates as a measure of panel diversity to reflect variability in the individual estimates. An alternative approach used quartiles of the individual estimates, leading to comparable, but narrower intervals.

Finally, there is a large sensitivity to the method used to aggregate the individual panelist estimates to obtain group estimates. A geometric mean aggregation, rather than an arithmetic mean or mixture distribution aggregation, was chosen in this case because it was a better representation of the consensus-type group estimate sought by this study.

There was a wide range of individual responses for the LOCA frequency expert elicitation. Use of the arithmetic mean or mixture distribution aggregation method in this case would produce group estimates which are dominated by the one or two highest individual estimates and would not represent a consensus-type group estimate, which was sought for this study.

For this study, the geometric mean aggregation develops frequency estimates which are a better representation of consensus-type results and better express the expert panel's state of knowledge regarding LOCA frequencies in a manner consistent with the elicitation process used to obtain the panel responses, the nature of these responses and the stated study objectives.

This study does not make a recommendation as to whether the LOCA frequency estimates using the baseline analysis procedure or using a particular sensitivity analysis should be used in any particular application. For the reasons stated above, the geometric mean aggregated results after using the error factor scheme to individually adjust for overconfidence (i.e., the *summary* estimates) are believed to be a reasonable representation of the expert panel's current state of knowledge regarding LOCA frequencies for the stated study objectives. However, sensitivity studies demonstrate that alternative analyses can lead to significantly different LOCA frequency estimates. Therefore, the purposes and context of any application must be considered when determining the applicability of any set of study results. While this places an additional burden on the users of the results, those users are in the best position to judge the extent to which the study results can be used for their particular applications.

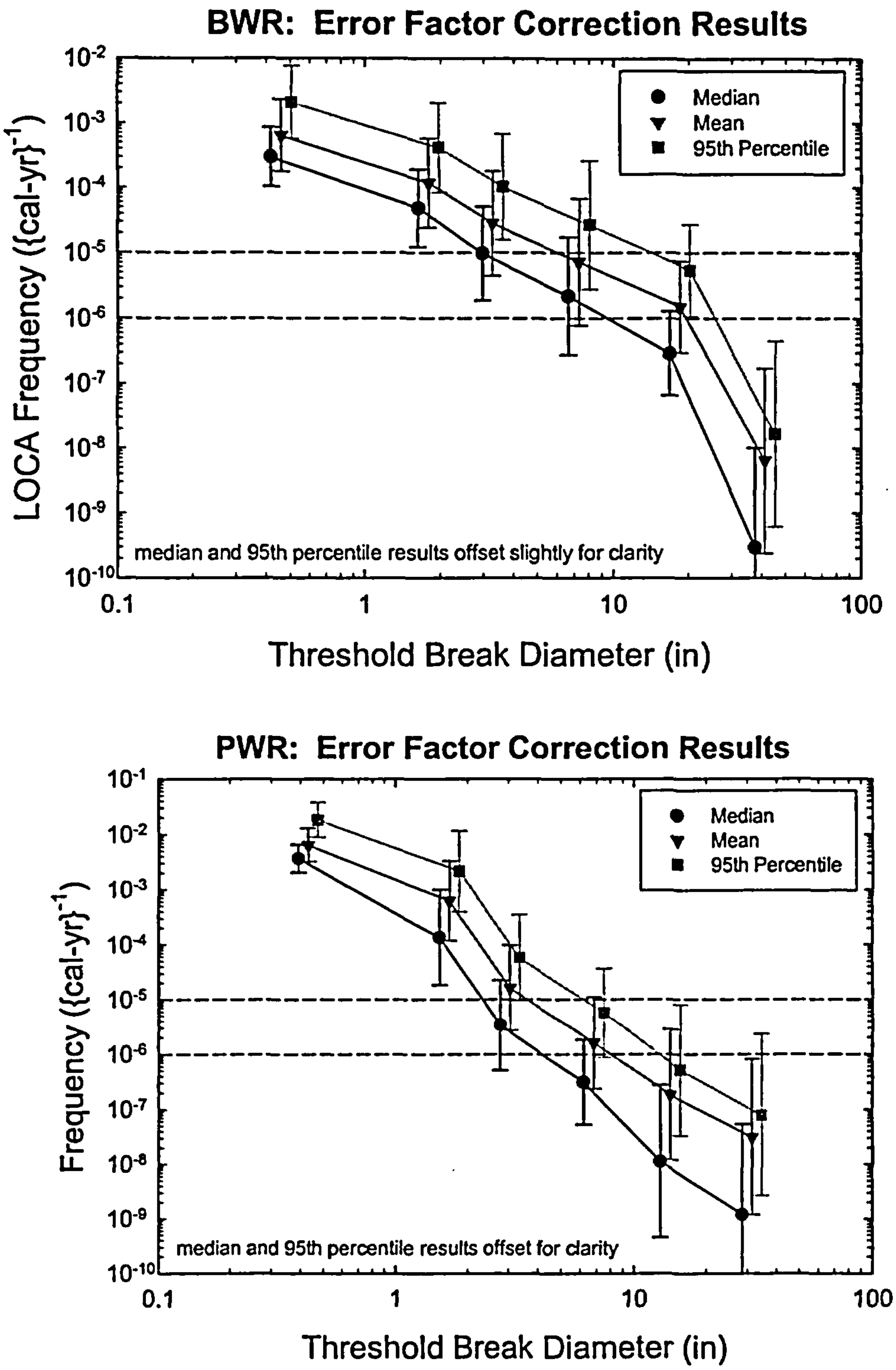

Figure 1 BWR and PWR Error Factor Adjusted LOCA Frequency Estimates

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• Dr. Karen Gott	Swedish Nuclear Power Inspectorate
• Dr. David Harris	Engineering Mechanics Technology, Inc.
• Mr. Bengt Lydell	ERIN® Engineering and Research, Inc.
• Dr. Peter Riccardella	Structural Integrity Associates, Inc.
• Mr. Helmut Schulz	Gesellschaft für Reaktorsicherheit (GRS) mbh
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NOMENCLATURE

1. Symbols

a	Flaw depth
a^*	Crack depth having 50% chance of being detected
b	Value of the upper bound supplied by a panelist
b'	Value of the lower bound supplied by a panelist
$b_p(Y)$	pth percentile of distribution Y
C	Parameter in fatigue crack growth relationship
C_1, C_2	Coefficients used in probability of detection curve definition
D	Diameter
DN	Nominal pipe diameter
E	Young's modulus
$E(Y)$	Expected value of distribution Y
$EF(Y)$	Error factor of distribution Y
f	Fraction of welds inspected for cracks or flaws
f	Inspection coverage/scope
g	Median of the lognormal distribution U
g^*	Group estimate of g
J	J-integral fracture parameter
J_D	Deformation J
J_{Ic}	Plane strain J at crack initiation by ASTM813
J_M	Modified J
J-R	J-resistance
k_a	p_ath percentile of the standard lognormal distribution
k_p	pth percentile of the standard lognormal distribution
K	Stress intensity factor
K_I	Mode I stress intensity factor
K_{Ic}	Plane strain stress intensity factor at crack initiation
f_{POD}	Probability of detection function
I	ISI effectiveness factor
L	Length
L_{BC}	Likelihood of a leak due to any degradation mechanism (base case)
L_{PL}	Likelihood of a perceptible leak
L_{TSL}	Likelihood of a technical specification leak
L_{50}	Likelihood of a crack 50% through wall deep
m	Median value
m	Parameter in fatigue crack growth relationship
$m(Y)$	Mean of distribution Y
n	Number of panelists
n_c	Number of cracks or flaws
n_f	Number of failure events
N	Normal operating stress
N	Number of components
N	Number of stress cycles
p	Percentile
P	Pressure
p_a	Value of the percentile in the adjusted distribution corresponding to the original error factor
P_{BC}	Conditional failure probability for the chosen seismic piping base case

P_{FD}	Probability inspected welds will find existing flaw
P_{ND}	Probability of not detecting a crack
P_{PL}	Conditional failure probability of a crack that has just formed a perceptible leak
P_{TSL}	Conditional failure probability of a crack leaking at the technical specification limit
$P_{TSL@SLB}$	Conditional failure probability of a crack leaking at the technical specification limit assuming a Service Level B load
$P_{TSL@SLD}$	Conditional failure probability of a crack leaking at the technical specification limit assuming a Service Level D load
P_{50}	Conditional failure probability of a crack with a maximum depth of 50% of the wall thickness
p'	Value of the percentile of the panelist's assumed lognormal distribution corresponding to b'
r	Error factor
r_p	Error factor of the panelist's adjusted lognormal distribution
R	Stress ratio
$R_{C/F}$	Number of non-through-wall cracks per leak event
S	Sum of cyclic stress
S	Seismic
S^2	Sample variance
SD(Y)	Standard deviation of distribution Y
t	Wall thickness
T	Thermal
T	Tearing modulus
T	Total time
V(Y)	Variance of distribution Y
Z_1	Anchoring factor
Z_2	Adjustment ratios
ϵ	Strain
ϵ	Probability of not detecting a crack regardless of depth
λ	Pipe failure frequency (through-wall crack)
μ	Mean
υ	Parameter controlling slope of P_{ND} curve
ϕ	Total frequency (cracks and leaks)
σ	Stress
σ	Standard deviation
σ^2	Variance
σ_{DW}	Dead weight stress
σ_{NO}	Normal operating stress
σ_{te}	Thermal expansion stress

2. Acronyms and Initialisms

AM	Arithmetic mean
ANL	Argonne National Laboratories
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
BINP	Battelle Integrity of Nuclear Piping
BWR	Boiling water reactor
B&W	Babcock and Wilcox

CBP	Conditional break probability
CE	Combustion Engineering
CFR	Code of Federal Regulations
CFP	Conditional failure probability
COD	Crack opening displacement
CRD	Control Rod Drive
CRDM	Control Rod Drive Mechanism
CS	Carbon steel
CV	Correct value
CVCS	Chemical Volume and Control System
DEGB	Double ended guillotine break
DVI	Direct Volume Injection
DW	Dead weight
ECCS	Emergency Core Cooling System
ECSCC	External chloride stress corrosion cracking
FDR	Fabrication defects and repairs
EDY	Effective degradation years
EF	Error factor
EFPY	Effective full power years
Emc^2	Engineering Mechanics Corporation of Columbus
EMT	Engineering Mechanics Technology
EPRI	Electric Power Research Institute
EQ	Elicitation question
FAC	Flow accelerated corrosion
FAD	Failure assessment diagram
FDR	Fabrication defect and repair
FS	Flow sensitive
FW	Feed water
GALL	Generic aging lessons learned
GC	General corrosion
GDC	General Design Criterion
GE	General Electric
GL	Generic letter
GM	Geometric mean
GRS	Gesellschaft für Reactorsicherheit
HAZ	Heat affected zone
HPCS	High Pressure Core Spray
HPI/MU	High Pressure Injection/Make-up
HWC	Hydrogen water chemistry
IAEA	International Atomic Energy Agency
ICI	In-core Instrumentation
ID	Inside diameter
IGSCC	Intergranular stress corrosion cracking
IHSI	Induction heat stress improvements
INEEL	Idaho National Engineering and Environmental Laboratory
IPE	Individual plant evaluation
IPIRG	International Piping Integrity Research Group
IQR	Interquartile range
ISI	In-service inspection
IS LOCA	Interfacing system loss of coolant accident
LAS	Low alloy steel

LB	Large break
LB	Lower bound
LBB	Leak before break
LC	Localized corrosion
LER	Licensee Event Report
LERF	Large early release frequency
LIV	Loop Isolation Valve
LOCA	Loss of coolant accident
LOOP	Loss of offsite power
LPCS	Low Pressure Core Spray
LQ	Lower quartile
LTOP	Low temperature over pressurization
MA	Material aging
MB	Medium break
MF	Mechanical fatigue
MITI	Ministry of International Trade and Industry (Japan)
MRP	Materials Reliability Program
MSIV	Main Steam Isolation Valve
MV	Mid value
NB	Nickel base weld
NDE	Non-destructive examination
NG	Nuclear grade
NPP	Nuclear power plant
NPS	Nominal pipe size
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NUPEC	Nuclear Power Engineering Test Center (Japan)
NWC	Normal water chemistry
OBE	Operational basis earthquake
OD	Outside diameter
ORNL	Oak Ridge National Laboratories
PFM	Probabilistic fracture mechanics
PIFRAC	Pipe FRACture mechanics material property database
PIV	Pressurizer isolation valve
PNNL	Pacific Northwest National Laboratories
POD	Probability of detection
PORV	Power operated relief valve
PRA	Probabilistic risk assessment
PSA	Probabilistic safety assessment
PSI	Pre-service inspection
PSL	Pressurizer Spray Line
PTS	Pressurized thermal shock
PVP	Pressure Vessel and Piping
PWHT	Post weld heat treatment
PWR	Pressurized water reactor
PWSCC	Primary water stress corrosion cracking
RCIC	Reactor Core Isolation Cooling
RCP	Reactor Cooling Piping
RCPB	Reactor Coolant Primary Boundary
RCS	Reactor cooling system
RES	Office of Nuclear Regulatory Research

RH	Reactor head
RHR	Residual Heat Removal
RI-ISI	Risk informed in-service inspection
RPV	Reactor Pressure Vessel
RR	Rolls Royce
RS	Residual stress
RV	Random variable
RWCU	Reactor Water Cleanup
SAM	Seismic anchor motion
SB	Small break
SCC	Stress corrosion cracking
SCSS	Sequence Coding and Search System
SG	Steam generator
SIS	Safety Injection System
SKI	Swedish Nuclear Inspectorate
SLB	Service Level B
SLC	Standby Liquid Control
SLD	Service Level D
SRM	Staff requirements memorandum
SRV	Safety relief valve
SQUIRT	Seepage Quantification of Upsets in Reactor Tubes
SS	Stainless steel
SSE	Safe shutdown earthquake
TF	Thermal fatigue
TGM	Trimmed geometric mean
TGSCC	Transgranular stress corrosion cracking
TMI	Three Mile Island
TS	Thermal stratification
TWC	Through-wall crack
UA	Unanticipated mechanism
UB	Upper bound
UQ	Upper quartile
US	United States
USNRC	United States Nuclear Regulatory Commission
VTC	Video Teleconference
WH	Water hammer
WO	Weld overlay
WOG	Westinghouse Owners Group

1. BACKGROUND

1.1 Motivation

The emergency core cooling system (ECCS) requirements are contained in 10 CFR 50.46, Appendix K to Part 50, and GDC 35. Specifically, ECCS design, reliability, and operating requirements exist to ensure that the system can successfully mitigate postulated loss-of-coolant accidents (LOCAs). Loss-of-coolant accidents are defined in 10 CFR 50.46(c) as hypothetical or postulated *"accidents that would result from the loss of reactor coolant, at a rate in excess of the capability of the reactor coolant makeup system, from breaks in pipes in the reactor coolant pressure boundary up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system"*. In addition, 10 CFR Part 50, Appendix K, paragraph (I)(C)(1), states that *"In analyses of hypothetical loss-of-coolant accidents, a spectrum of possible pipe breaks shall be considered. This spectrum shall include instantaneous double-ended breaks ranging in cross-sectional area up to and including that of the largest pipe in the primary coolant system."* The LOCA definition in 10 CFR Part 50, Appendix A, expands this definition to consider "breaks in the reactor coolant pressure boundary, up to and including a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system." The consideration of instantaneous breaks with a flow rate equivalent to a double-ended guillotine break (DEGB) of the largest primary system piping generally provides the limiting condition in the required 10 CFR Part 50, Appendix K analysis.

A DEGB of the largest primary system piping is widely recognized as an extremely unlikely event. Furthermore, the consideration of this event in nuclear plant design and operation requires resources beyond what is commensurate with the associated risk. Focusing resources on more risk-significant events in a manner consistent with Regulatory Guide 1.174 [1.1] should ultimately improve plant safety while also eliminating overly burdensome restrictions. In an effort spur a risk-informed reevaluation of the regulatory requirements, the Commission provided NRC staff with direction to proceed with a study of risk-informing the technical requirements of 10 CFR Part 50 [1.2]. The staff provided its plan and schedule for this work in SECY-99-264, "Proposed Staff Plan for Risk-Informing Technical Requirements in 10 CFR Part 50," dated November 8, 1999. In this plan, the risk-informed reevaluation of 10 CFR 50.46 requirements was prioritized. The Commission approved this plan on February 3, 2000 [1.3]. Since that time, the staff has provided five status reports to the Commission: SECY-00-0086, SECY-00-0198, memorandum to the Commission dated February 5, 2001, SECY-01-0133, and SECY-02-0057 [1.4]. A staff requirements memorandum (SRM) was provided on March 31, 2003 [1.5] addressing the latest status report, SECY-02-0057. This SRM provides explicit staff direction on risk-informing 10 CFR 50.46, Appendix K, and GDC 35. Most relevantly, the SRM directed the staff to consider a risk-informed revision of the design-basis break size requirements and certain non-functional changes to the design basis of operating commercial nuclear power plants.

A central consideration in selecting a risk-informed design basis break size is an understanding of the LOCA frequency as a function of break size. The most recent NRC-sponsored study of pipe break failure frequencies is contained in NUREG/CR-5750 [1.6]. These frequencies are currently the basis for initiating event frequencies in many plants' probabilistic risk assessment (PRA) models. However, there are several concerns with utilizing the NUREG/CR-5750 LOCA frequency estimates for the risk-informed reevaluation of 10 CFR 50.46. First, several degradation mechanisms have emerged at plants within the last few years which were not previously evident within the service period covered by the NUREG/CR-5750 estimates. These include primary-water stress-corrosion cracking (PWSCC) of pressurized water reactor (PWR) alloy 82/182 welds, PWR vessel head degradation at Davis Bessie, hydrogen combustion failures of the type experienced at the Hamaoka and Brunsbüttel plants, and control-rod-drive mechanism (CRDM) cracking.

Second, LOCAs can occur from failure of non-pipe break passive failures (e.g. CRDM nozzles, valve bodies, vessel head degradation, and steam generator tubes). The NUREG/CR-5750 estimates only focused on piping failures while LOCA contributions from other passive system failures must also be considered. Third, forward-looking LOCA frequency estimates are required to understand the future ramifications associated with possible changes to the regulation up to the end of the license-extension period for approved plants. The NUREG/CR-5750 estimates are unavoidably based solely on the historical operating performance.

Finally, the NUREG/CR-5750 estimates defined LOCA break sizes in a manner consistent with current PRA classification using small break (SB), medium break (MB), and large break (LB) LOCA categories that are loosely based on plant mitigation requirements for each break size. The large break category represents the cumulative frequency of a rupture with an equivalent diameter greater than 6 inches. These frequencies will not be representative of the failure frequency of a DEGB in the largest primary system piping which is approximately 30 inches in diameter for most PWR plants. Therefore, frequency estimates of breaks larger than an equivalent 6 inch diameter pipe need to be considered when risk-informing 10 CFR 50.46.

A review of the NUREG/CR-5750 and other historical LOCA frequency estimation techniques follows. The strengths and weaknesses of each technique are highlighted with respect to many of the concerns enumerated above. The proper consideration of the role of mitigation in LOCA frequency estimates is discussed as well as a description of non-historical failure modes that are important to consider. The relative merits of expert elicitation are also outlined and are the basis for its use in this estimation of LOCA frequencies.

1.2 Previous Approaches for Estimating LOCA Frequencies

There have been two approaches traditionally used to assess LOCA frequencies as a function of pipe size: estimates based on statistical analysis of service experience data and probabilistic fracture mechanics (PFM) analysis of specific postulated failure mechanisms. Both these approaches have unique strengths and weaknesses for determining LOCA frequencies. In many ways, the two methods are complementary although combined or comparative analyses utilizing both approaches are rare.

1.2.1 Service-Based LOCA Estimates

There are several distinct advantages to using service experience approaches to directly calculate LOCA frequencies. Service history experience identifies the degradation mechanisms in piping systems and non-piping components which have led to defect repair or material replacement under operating plant conditions. Service data can also provide, in the long term, information on the effectiveness of specific mitigation techniques and some indication of the likelihood of precursor events (e.g., cracks and leaks) prior to failure for specific degradation mechanisms. Detailed service history data can also be used to define the role of operating conditions, service environment, design characteristics, or fabrication techniques which could lead to degradation. All this information is valuable for inferring LOCA frequencies.

However, some natural deficiencies make it challenging to directly calculate LOCA frequencies from operating data. First, precursor failure information comes from a variety of sources and it is difficult to construct a comprehensive database. Because pipe break LOCAs have not occurred, capturing all precursor failure events is important when estimating actual passive-system failure rates. Additionally, the precursor events in the operating experience must be conditionally related to LOCA frequencies, and not all LOCAs evolve from detectable precursor events (e.g., hydrogen explosions). The relationship between precursor and non-precursor events and associated LOCA failure frequencies is a function of the aging mechanism. This relationship, for LOCA determination, cannot be explicitly developed from the operating experience.

Past service experience is also not necessarily representative of future performance. Therefore, the effect of future aging mechanisms that have yet to be detected is difficult to explicitly consider using service data. When new aging mechanisms do emerge in service, there is a lag before their consequences are fully understood. Aging mechanisms can require significant incubation time before causing significant precursor degradation or failure. However, once the incubation period is over, degradation can occur relatively rapidly and can lead to rapidly increasing failure propensity with time. This pattern was experienced with intergranular stress corrosion cracking (IGSCC) failures in the early 1980s [1.7]. Recent primary water stress corrosion cracking (PWSCC) [1.8] is another aging mechanism with similar characteristics. Therefore, it is crucial to consider the effect that these, and other aging mechanisms, could have on historical piping precursor and failure rates.

1.2.1.1 WASH-1400 - The first systematic study of piping failure in the nuclear industry is contained within WASH-1400 which was completed in 1975 [1.9]. At the time, the combined years of reactor service experience was less than 200. Therefore, the pipe LOCA frequencies were based on experience within other industries. WASH-1400 examined data from the naval nuclear reactor experience, experimental reactors, United Kingdom military information, commercial power plants, and the oil and gas transmission pipeline industry. The most comprehensive data was obtained from the oil and gas pipeline industry and formed the basis of the WASH-1400 LOCA frequency estimates after proper normalization to account for nuclear plant pipe lengths.

It was certainly realized in the WASH-1400 study that transmission pipeline materials, quality assurance, in-service inspection, operating conditions, failure modes, and environments were vastly different from, and in most cases inferior to, commercial nuclear requirements. Therefore, the WASH-1400 analysis was considered to represent an appropriately conservative estimate of the pipe break LOCA frequency given the relatively scant nuclear experience available at the time of this study. The WASH-1400 pipe break estimates formed the basis of NUREG-1150 estimates in 1987 [1.10]. The NUREG-1150 estimates were calculated by performing a Bayesian update of the WASH-1400 estimates with the information that no additional LOCAs failures had occurred in the interim.

1.2.1.2 NUREG/CR-5750 - The next NRC-sponsored evaluation of pipe break LOCA frequencies occurred within Appendix J of NUREG/CR-5750, "Rates of Initiating Events at U.S. Nuclear Power Plants: 1987 - 1995" [1.6]. The authors evaluated nuclear piping failures in this study and separate frequencies were determined for BWR and PWR reactors. For BWR plants, only U.S. experience was considered for a total of 710 reactor calendar years. The PWR database combined U.S. and "Western-style" PWR data from international experience for a total of 3,362 reactor calendar years. The authors utilized distinct methods to calculate pipe break frequencies as a function of break size. The SB LOCA estimates were calculated using a Bayesian update of the WASH-1400 SB LOCA estimates with the information that no SB LOCAs occurred between the WASH-1400 and NUREG/CR- 5750, Appendix J studies. This is analogous to the earlier update for the NUREG-1150 estimates.

The MB and LB LOCA frequencies were derived from precursor leak frequencies determined from service experience. The leak frequency was multiplied by a Beliczey and Schulz conditional pipe break probability (CBP) which is inversely related to pipe diameter [1.11]:

$$CBP = 2.5/DN \tag{1.1}$$

In Equation 1.1, DN is the nominal pipe diameter in millimeters. The NUREG/CR-5750, Appendix J analysis capped this expression at 0.01 for piping greater than 250 mm to impart some conservatism to the estimates. The advantage of this approach is that there had been several reported leaks of primary pressure boundary piping, but no failures. Therefore, service history experience could be utilized directly to determine the pipe leak frequency and only the CBP given a precursor leak needed to be estimated. One concern is that the Beliczey and Schulz expression was developed for fatigue crack failures only. It is not expected to be applicable for other aging mechanisms. A bigger concern with this precursor approach is that it ignores failure contributions from existing flaws or degradation that does not result in a leak. There are many potential initiating events which do not exhibit a precursor leak. Recent hydrogen combustion failures of residual heat removal piping at Hamaoka [1.12] and auxiliary coolant system piping at Brunsbüttel [1.13] represent one such mechanism. Flow accelerated corrosion (FAC), which induced a rupture of an 18" diameter feedwater suction pipe elbow at Surry 2 in 1986 [1.14], is another mechanism which can lead to rupture prior to precursor leaking.

The NUREG/CR-5750, Appendix J approach also applied a mitigation factor of 1/20 to the BWR leak rate data to account for the effectiveness of IGSCC mitigation strategies. The IGSCC mechanism was the prevalent BWR piping failure mechanism within the precursor database. This mitigation factor was justified in light of analysis of the operating experience data [1.15] and quantitative estimates of the improvement in reliability for all mitigation strategies [1.16]. There are several different IGSCC mitigation techniques that have been applied including: hydrogen water chemistry addition, weld overlay, increased inspection, and material replacement with a less susceptible austenitic grade of piping (L or NG grades) [1.16]. Some plants have applied single or multiple mitigation strategies and the effectiveness of each particular strategy will obviously vary and may not be well-characterized by a single mitigation factor. Additionally, the Germans have experienced IGSCC in some of the low carbon, less susceptible steels [1.17]. Therefore, this mitigation factor may also not be applicable to describe the future effectiveness of IGSCC mitigation measures up to the end of the license renewal period.

1.2.1.3 Barsebäck-1 Estimation - In the same time period as the NUREG/CR-5750 evaluation, the Swedish Nuclear Inspectorate (SKI) initiated an effort to develop an international piping failure database [1.18]. This differs from earlier studies in several fundamental ways. Most importantly, it was the first study to concentrate solely on pipe failure and it includes class 2 and class 3 piping failures as well as class 1, or primary system piping. Secondly, several data sources were utilized to corroborate each event and determine the metallographic root failure cause whenever possible. Thirdly, it was constructed to allow specific queries by material engineers and PRA practitioners. The database has been employed to develop initiating event frequencies, evaluate emerging failure trends, and judge the effectiveness of mitigation strategies.

The SKI effort culminated in 1997 with a guide for using the database to evaluate piping reliability in terms of important influence and attribute factors [1.15]. The report stresses the need to consider each failure mechanism separately and develop estimates from the database with respect to the influence factors for that mechanism. For instance, flow accelerated corrosion (FAC) is most severe in carbon steel piping at pipe tees and elbows. Pipe failure frequencies associated with FAC are then developed per reactor year and per the number of carbon steel pipe tees and elbows in the plant using the database. Then, the relevant attributes of a specific plant are determined (i.e. the number of carbon steel pipe tees and elbows) and plant-specific event frequencies can be developed. This process is repeated for all potential failure mechanisms (e.g., IGSCC, vibration fatigue, water hammer, etc.). Finally, all the relevant mechanisms are combined to develop total LOCA frequencies [1.15].

This framework was followed to develop failure rates in the Barsebäck-1 reactor coolant pressure boundary piping [1.19]. Barsebäck-1 is a third generation boiling water reactor by ABB-Atom that was closed in 1999. The study collected all the plant specific attributes of the Barsebäck-1 plant. The attributes include the pipe materials, geometry, pipe length, number of pipe welds, number of pipe connections, and the connection type. Only medium and large LOCA frequencies were determined by considering those failure mechanisms represented within the database.

While the initial SKI-effort database developments and the Barsebäck study were completed in 1998, the database has been maintained and updated through the current period. There is currently a three-year OECD-sponsored Piping Database Exchange (OPDE) project involving the U.S. and twelve other European and Asian countries that is expanding the coverage of the database for events occurring during the 1990's, adding current events, and improving the accuracy and completeness of all included events. Specific attention is focused on including indications of non-leaking flaws discovered during pre-service and in-service inspections. The 1998 version of the database included 1,880 U.S. piping failures and a total of 2,416 failures worldwide. Currently, in June 2004, the database contains nearly 3,200 U.S. piping failures and 4,600 worldwide failures. Therefore, the database completeness has been enhanced by the inclusion of over 70% more US events and almost double the number of international events since 1998.

1.2.1.4 Comparison of Service-Based LOCA Estimates - The pipe break LOCA frequencies developed from the WASH-1400, NUREG/CR-5750, Appendix J, and the Barsebäck-1 study using the SKI pipe database are compared in Table 1.1. As previously noted, each of these assessments differed dramatically in applied rigor, the completeness and relevance of the underlying database, and the time period of study. The WASH-1400 estimates are understandably higher because the oil and gas transmission industry does not have the same rigorous design practice, quality assurance, and inspection as the nuclear industry. Also, pipe failure from external damage (e.g., mechanical damage from construction equipment, external corrosion, etc.) represents the largest contributing failure mechanism in this industry. The WASH-1400 estimates also reflect the conservatism necessarily based on the dearth of operating experience at the time of this study.

The application of actual service data in NUREG/CR-5750, Appendix J and the SKI-pipe database in the Barsebäck-1 study reduced the expected break frequencies by an order of magnitude compared to WASH-1400 (Table 1.1). There is roughly a factor of two difference between these two studies, with the Barsebäck study predicting more MB LOCA events and less LB LOCA events. This difference is primarily due to the differences in the analysis methodology and the application of plant specific data for Barsebäck-1. The BWR frequencies in both the NUREG/CR-5750 and Barsebäck-1 account for the successful mitigation strategies adopted to combat IGSCC in the early and mid-1980's.

Table 1.1 Comparison of Mean Results from Previous Studies (per Calendar Years)

Reactor Type	Analysis	SB LOCA ($x 10^{-5}$)	MB LOCA ($x 10^{-5}$)	LB LOCA ($x 10^{-5}$)
BWR	WASH-1400	300	80	30
	NUREG/CR-5750	40	2.6	2.4
	Barsebäck-1	NA	3.4	0.8
PWR	WASH-1400	300	80	30
	NUREG/CR-5750	40	3.0	0.36

1.2.2 PFM LOCA Estimates

The PFM models are attractive because they can parametrically assess the effects of possible mitigation strategies, including inspection, on future piping system performance for particular degradation mechanisms. This is a valuable trait for assessing the possible severity of emerging degradation mechanisms where there is both a dearth of service history experience and a lack of understanding of mitigation effects. However, it is difficult to utilize PFM models for developing absolute LOCA frequency estimations because many of the input variables and model assumptions are overly simplistic and may not adequately represent true plant conditions. The simplicity also causes variability in the predicted results as a function of the input uncertainties and specific assumptions. It is also difficult, and uncommon, to benchmark PFM models using the sparse piping failure information. It is often necessary to benchmark PFM models using limited existing data for large non-leaking flaws or for very small leaks. The model can then be used to predict the probabilities of much larger leaks or complete failures as a function of operating time. The PFM failure rate predictions for future operation can only be realistic if estimations for current precursor or failure rates match the service experience.

The international community (e.g., GRS in Germany and SKI in Sweden) has sponsored several PFM-based predictions of passive system failure rates [1.20, 1.21]. The NRC and the nuclear industry have also sponsored PFM-research over the last twenty years in an attempt to develop LOCA predictions from first principals including the prediction of pipe failure probability in Westinghouse PWR reactor coolant loops [1.22] and the reliability analysis of stiff versus flexible piping [1.23]. Additionally, joint international research and investigation of primary piping system integrity has been conducted in the International Piping Integrity Research Group programs (IPIRG1 and IPIRG2) [1.24, 1.25] as well as the Battelle Integrity of Nuclear Piping program (BINP) [1.26].

There was also a large body of research sponsored during the 1980's to evaluate the leak-before-break propensity of BWR and PWR plants [1.27-1.30]. These studies evaluated the frequency of double-ended guillotine breaks (DEGB) in BWR main steam, feedwater, and reactor recirculation piping, and in PWR reactor coolant loop piping. Also, the frequency of seismic-induced direct and indirect DEGB in PWR plants was studied. Direct DEGB is induced by fatigue crack growth due to the combined effects of thermal, pressure, seismic, and other cyclic loads. Indirect DEGB occurs when the failure of other passive components causes a consequential failure in the primary piping system. An example is the failure of PWR steam generator supports, leading to additional primary coolant piping loads and possible failure.

Table 1.2 is a summary of the seismic-induced DEGB median and 90th percentile estimates of the associated frequencies.

Table 1.2 Seismic-Induced DEGB Frequencies for PWRs

Plant Type	Frequency of Seismic-Induced DEGB [1/Reactor Year]			
	Direct DEGB		Indirect DEGB	
	Median	90th Percentile	Median	90th Percentile
CE PWRs	6.0E-14 to 5.0E-13	4.0E-12 to 7.0E-11	5.0E-17 to 6.0E-6	3.0E-14 to 5.0E-5
Westinghouse PWRs	2.0E-13 to 3.0E-11	8.0E-10 to 1.0E-9	5.0E-8 to 5.0E-6	1.0E-6 to 5.0E-5
B&W PWRs	< 1.0E-10		6.0E-11 to 2.0E-7	8.0E-9 to 1.0E-5

Several of the more well-known US computer codes arising from these various efforts include PRAISE [1.31], SRRA [1.32], PSQUIRT [1.33], and PROLBB [1.34]. Each code has distinct capabilities, and there are marked differences in structure, approach, modeling assumptions, and the degree of benchmarking with failure data. The PRAISE models have been benchmarked with the through-wall fatigue cracks in a PWR feedwater line near the steam generator and with the small leak in a BWR recirculation line inlet nozzle safe end [1.35] and with the observed leak probabilities due to IGSCC [1.36]. The SRRA code has been benchmarked with observed repair data for flow-assisted corrosion and directly compared with the PRAISE results for fatigue [1.32]. Crack initiation, sub-critical crack growth, leak rate calculation, and fracture/failure analysis are the four major components required for life prediction using PFM, yet the various codes handle these modules using vastly different approaches, or not at all.

For instance, the SRRA code uses a limit-load failure criterion while the PROLBB code uses an elastic-plastic fracture mechanics based criterion. This single difference can result in tremendous output variability in the pipe rupture probability. Figure 1.1 depicts the conditional pipe failure probability for a hypothetical through-wall crack. The crack size is increased systematically and the associated leak rate and conditional failure probability are calculated. The figure on the left assumes that the pipe fails upon reaching limit load, i.e., the crack is in high toughness wrought stainless steel. The figure on the right assumes the crack is in lower toughness stainless steel weld metal such that failure occurs below the pipe limit stress. This single difference results in up to a five order of magnitude difference in the conditional break probability predictions. For a conservatively assumed rupture loading event frequency of 10^{-3} per year, the expected failure frequency would be about 10^{-10} {per calendar year} for the base metal and about 10^{-7} {per calendar year} for the weld metal assuming loading at 50% of service level A and a 1 gpm (3.8 lpm) leakage detection threshold. This threshold is approximately the level required to locate unidentified leakage within nuclear plants. Either analysis could be realistic for a given weld joint depending on the actual pipe material, fabrication practice, and crack location.

The PFM analysis must also model each potential aging mechanism of concern (FAC, IGSCC, PWSCC, corrosion fatigue, vibration fatigue, etc.) to develop comprehensive LOCA estimates. Each mechanism potentially requires a different degradation/cracking model and corresponding set of material properties to accurately assess the LOCA likelihood. Each of the above PFM codes also utilize different approximations for input parameters such as the applied load magnitude and spectrum, pipe boundary conditions, residual stress contribution, and initial flaw distribution. While the codes are complex, many simplifying assumptions are necessary and the model of actual plant conditions is, at best, approximate. Given these modeling realities, and the inherent sensitivity of fatigue and crack growth calculations to the initial conditions and modeling assumptions, it is not surprising that the results from the various models can vary dramatically.

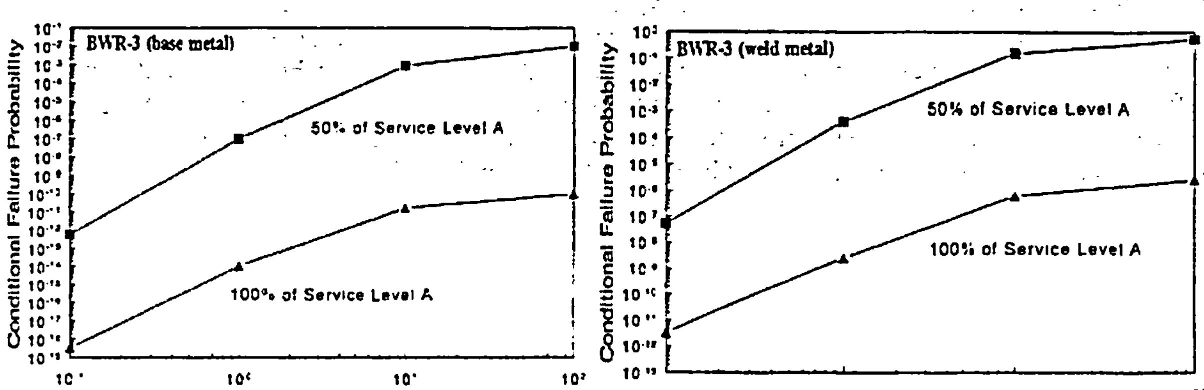

Figure 1.1 Sensitivity of Conditional Failure Probability to Failure Mode Assumption

1.3 Modeling Mitigation Effects

Once degradation mechanisms are discovered and assessed, it is expected that mitigation measures will be developed and applied in order to decrease the passive system degradation rate to historical levels, or lower. Mitigation includes such measures as inspection, material replacement, repair, and changes in operating conditions. Therefore, the expected impact of a new degradation mechanism on LOCA frequencies is a short term increase followed by a decrease due to mitigation after some time period. This effect, and the benefits of successful mitigation, are demonstrated by the IGSCC cracking phenomena in the early-1980s (Figure 1.2). There was a period of two to three years when the cracking mechanism and its prevalence was discovered in susceptible systems. There was another two to three year period before mitigation was developed and fully implemented. During the approximately four year period of diagnosing and solving the IGSCC problem, the instantaneous discovery rate was approximately one order of magnitude higher than historical levels. However, since mitigation has been employed, the occurrence of IGSCC is less than pre-1980 values [1.19].

It can be difficult to account for the effects of mitigation in either service-based or PFM LOCA frequency estimates. The pre-mitigation data is part of the service history and must be properly screened when predicting frequencies for plants which have implemented mitigation measures. While the general effect of mitigation on IGSCC is apparent, the effect of individual mitigation measures at specific plants is less obvious. This is relevant because several different IGSCC mitigation options are practiced, often in combination, and it is difficult to quantify the effects of individual measures from the data alone. The frequencies developed within the NUREG/CR-5750 and the Barsebäck-1 analysis did take into account the general effect of IGSCC mitigation when calculating frequencies (Table 1.1). However, both these efforts had the benefit of several years of mitigation experience as a basis for their analysis at the time of the calculation.

In probabilistic fracture mechanics-based analysis, a more explicit treatment of individual mitigation measures is possible to evaluate the effects on the degradation rate. The PFM approach can be especially valuable for comparing the relative differences between pre- and post-mitigation failure probabilities. An ASME-sponsored effort [1.37] examines the effects of mitigation techniques such as stress improvement and weld overlay for BWR recirculation piping subject to IGSCC. However, mitigation measures are often simplistically modeled and absolute predictions of post-mitigation degradation rates, and postulated failure frequencies, are subject to the same limitations discussed earlier for PFM models. A common mitigation technique usually considered in PFM analysis is the effect of inspection on the failure frequencies. The impact of inspection is a function of the periodicity, the accuracy of the technique, and operator skill. The accuracy and operator skill is typically considered by a single relationship between the probability of detection (POD) and defect size. Unfortunately, for many techniques, this relationship is highly variable and the curves themselves are often developed under idealized laboratory conditions. Therefore, they may not be representative of actual field experience. The POD relationship chosen for the models can significantly influence the predicted failure rates. The possible impact associated with inspection accuracies have been documented by an ASME Research Task Force [1.37].

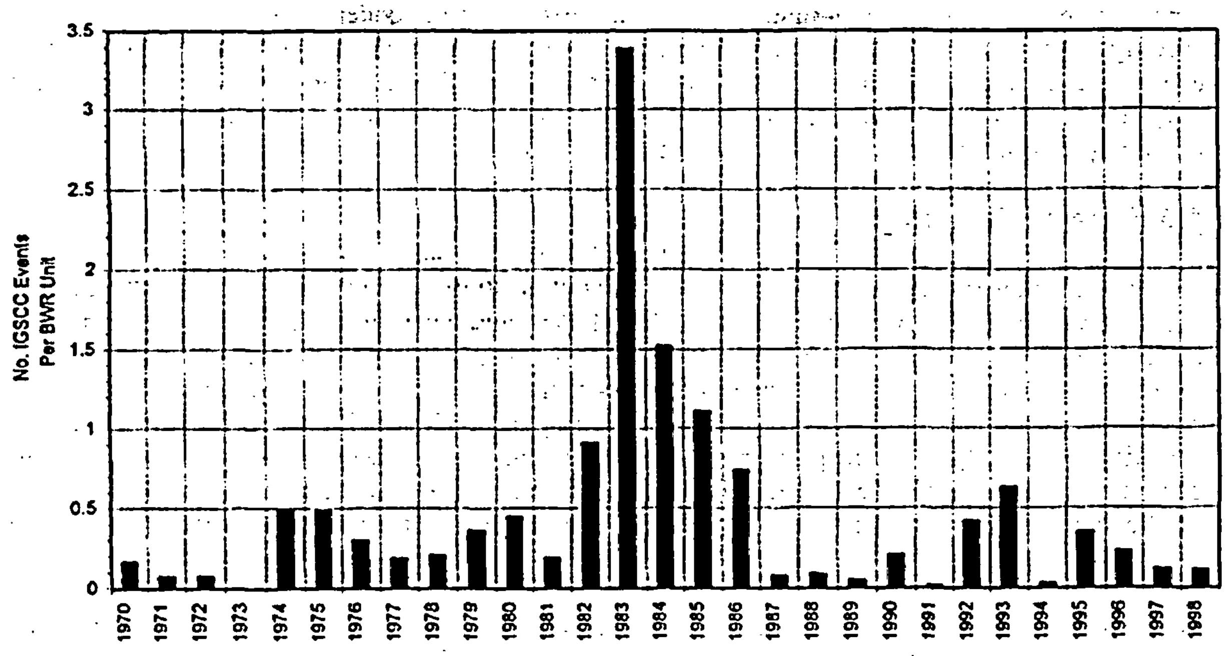

Figure 1.2 IGSCC Events by Year

1.4 Other LOCA Initiating Events

The LOCA frequencies are also influenced by active component failure. Active components are defined as those components subject to periodic maintenance and those that contain moving parts which are subject to either wear or failure (e.g., pump seals, SRV, PORV, etc.). Active component failure was not considered within this study, but these frequencies should be combined with the passive system LOCA frequencies to understand total system risk. These failures occur with enough regularity, that they can be estimated from the operating experience database as in [1.6].

Steam generator tube rupture is a passive component failure which was considered in the elicitation. Tube ruptures have also occurred with enough regularity that they are represented in the operating experience database. Historical rupture frequencies can therefore be established as in [1.6]. However, the applicability of the operating experience to the future steam generator failure rates is unclear. For instance, it is important to understand how steam generator replacement, secondary and primary side environmental changes, and other factors may affect the future failure rates. For these reasons, these failures were also considered in the elicitation.

There is an entire class of other non-piping, passive system failures which have recently emerged that have not been considered in previous LOCA frequency estimates. These failure rates must be combined with historical pipe failure estimates to determine more comprehensive LOCA frequencies. Recent experience has also demonstrated that degradation, and therefore LOCAs, can occur in non-piping components such as CRDM housings, BWR stub tubes (Hamaoka), and the RPV reactor head (Davis Besse)]. These degradation mechanisms and failure locations have not been explicitly represented in historical piping-based service history databases. However, the impact of these emerging LOCA initiators is an important consideration in determining total LOCA risk.

Finally, in both piping and non-piping components, it is important to identify and assess the LOCA severity of degradation mechanisms with long incubation times which have yet to surface in the operating experience and may not be modeled in PFM predictions. The identification of these future mechanisms is only possible by understanding the long term interrelationship between the passive component materials, operating environment, and loading conditions.

1.5 Expert Elicitation

Expert elicitation has attributes which can build on the strengths and compensate for the weaknesses associated with purely service-based or PFM-based approaches to estimate LOCA frequencies. These attributes make this technique a natural choice for estimating LOCA frequencies. Expert elicitation is a formal process for providing quantitative estimates of the frequencies of physical phenomena when the required data is sparse and when the subject is too complex to adequately model. Furthermore, the scientific uncertainty about the phenomena is so large that, without adequate data, validated models or computer codes cannot be developed. If the issue is also important, i.e., it has significant regulatory implications and may also be controversial, then devoting the substantial resources required for an expert elicitation may be justified.

Formal elicitation is a structured process which enhances its accuracy, consistency, credibility, and thus acceptability compared to informal, less-structured processes. The emphasis on a structured decomposition of the issues improves accuracy and credibility, thus making the results more acceptable to the stakeholders. Formal elicitation reduces the likelihood of bias and enhances the consistency and comparability of the results. The emphasis on documentation leads to improved scrutiny and acceptance of the results. The main drawbacks in using a formal expert elicitation process are the increased time and resources required. Because of the structure, there is also reduced flexibility to make changes as the process proceeds.

Formal elicitation is a well-established technique [1.38] which has been used on a number of occasions to evaluate technical issues related to nuclear safety. Examples include: NUREG-1150 [1.10], the determination of flaw density and size distributions in reactor pressure vessels [1.39], the evaluation of the high level radioactive waste repository [1.40, 1.41], and in probabilistic seismic hazard curve analysis [1.42].

Formal elicitation was selected to develop passive system LOCA frequency estimates because data sparseness and subject complexity are characteristic of pipe break LOCA frequency estimation. Data sparseness is evidenced by the fact that no Class 1 nuclear pipe break LOCA events have occurred. Existing pipe break LOCA frequency estimates from NUREG/CR-5750, Appendix J, vary from 4×10^{-4} per calendar-year for SB LOCAs to 4×10^{-6} per calendar-year for PWR LB LOCAs. On the average, this translates into one SB LOCA every 2,500 years per plant and one PWR LB LOCA every 250,000 years per plant.

Complexity is evidenced by the large number of pipe system variables which must be considered to accurately evaluate the full spectrum of pipe breaks using the PFM or statistical methods discussed in previous sections. Variables include piping design and layout; piping fabrication; materials; degradation mechanisms; stress; service environment; application of codes and standards; inspection type, quality and schedule; and the plant operating history. These variables serve as input for both service-history and PFM models. The importance of a particular degradation mechanism depends on the convergence of appropriate combinations of these variables.

As previously discussed, the number and interaction of these variables and the sparseness of data severely hinders accurate frequency assessment using service-history approaches (Section 1.2.1). The PFM approach (Section 1.2.2) suffers because small changes in the input variable assumptions can dramatically affect the predicted piping reliability. Also, the underlying physical modeling of degradation and failure mechanisms (crack initiation, leak rates, complex crack growth, and piping instability) is unavoidably simplistic. These limitations lead to a great deal of relative uncertainty when utilizing solely service-history or PFM analysis to evaluate LOCA frequency estimates. Expert elicitation to develop these estimates fosters a more inclusive approach by using individuals with relevant technical expertise to identify and focus on important variables which affect LOCA frequencies. The experts then are required to combine insights gained from service data, PFM modeling, and other considerations to develop comprehensive LOCA estimates.

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2. OBJECTIVE AND SCOPE

The objective of the expert elicitation process was to develop piping and non-piping passive system LOCA frequency estimates as a function of effective break size and operating time through the end of license extension. The evaluation considered three distinct time periods: the current day, the end of plant licensing, and the end of license extension. For the purposes of the elicitation, these time periods are quantified as 25 (approximate current fleet average), 40, and 60 years, respectively, after plant operation commences. The time periods were not specifically defined in the elicitation to represent either point estimates of LOCA frequencies at 25, 40, and 60 years of operation or frequency averages between 0 – 25 years, 25 – 40 years, and 40 – 60 years of plant operation. Each panelist was asked to provide their responses and then invited to discuss possible differences between the average, point estimate, and maximum frequencies within the defined operational time periods of generic plant operations.

The elicitation was solely focused on determining event frequencies that initiate by unisolable primary system side failures that can be exacerbated by material degradation with age. Therefore, active system failures (e.g., stuck open valve, pump seals, interfacing system LOCAs), consequential primary pressure boundary failures due to either secondary side failures, or failures of other plant structures (e.g., crane drops) were not considered. Such frequency contributions may be an important consideration when evaluating total plant risk and determining total LOCA frequency estimates, however, assessment of these risk contributions is beyond the expertise of the assembled panel members.

The LOCA frequency estimates are represented by the estimated median, mean, 5th and 95th percentiles parameters. Separate LOCA frequency distributions have been determined associated with boiling water reactor (BWR) piping, BWR non-piping, pressurized water reactor (PWR) piping, and PWR non-piping failures. These piping and non-piping frequencies have been combined to estimate total passive system LOCA frequencies for BWR and PWR plants for each individual panelist. Then, these individual results have been aggregated to provide a central estimate. These central estimates represent a type of consensus result for the group. Additionally, confidence bounds have been determined to reflect the variability among the panel members for these central estimates. The central estimates for each LOCA parameter are the primary quantitative results that are sought by the elicitation process.

The elicitation focused on developing generic, or average, values for the commercial fleet and uncertainty bounds of this generic average. The BWR and PWR LOCA frequencies have not been partitioned further to describe differences related to design class, vendor, or specific plant operating characteristics. These features can influence LOCA frequencies and it is expected that frequency distributions for specific plants differ from these generic distributions. Elicitation panelists were instructed to account for broad plant-specific factors which influence both the generic LOCA frequencies and especially the uncertainty bound estimations, but not to consider differences that only exist at a few plants. For instance, if a particular plant vendor design is more LOCA-sensitive and encompasses 20 PWRs, then the panelist should consider this in the generic frequency determination and also the upper bound estimation. If the same plant design only applies to two plants, then this should not be considered in the estimates.

This exercise developed LOCA frequencies consistent with historical small break (SB), medium break (MB), and large break (LB) flow-rate definitions. Additionally, three larger LOCA categories were defined in the elicitation within the classical LB LOCA regime. The purpose of these additional categories was to examine trends with increasing break size, up to and including, a DEGB of the largest piping in the plant. The consideration of the consequences of such a break is a requirement of current 10 CFR 50.46, and associated Appendix K and GDC 35 requirements. The SB, MB, and LB LOCA categories have historically been defined on the basis of flow rate. Simple correlations were developed to relate the rupture size to the expected flow rate required for the ECCS make-up system. The correlations developed herein are different from those used in the past, but they provide a mechanism to compare these current LOCA estimates with previous benchmarks.

The intent is that these LOCA frequencies will also be amenable to future evaluation of core damage frequency (CDF) and large early release frequency (LERF) metrics using both current and advanced probabilistic risk assessment (PRA) tools. Therefore, the elicitation primarily considers normal plant operational cycles and loading histories consistent with current internal event PRA analyses. Separate frequencies for each unique mode of plant operation have not been determined. Rather, the frequencies developed implicitly consider all modes of operation per calendar year per reactor based on the loading or operational history associated with each piping system or non-piping component.

Consideration of normal plant operational cycles and loading histories was limited to representative constant stresses (e.g., pressure, thermal, and residual) and expected transient stresses (e.g., thermal striping, heat-up/cool-down, and pressure transients) that occur over the extended licensing period. Therefore only loading events with a frequency of greater than approximately 0.01 $\{\text{cal-yr}\}^{-1}$ were explicitly addressed. Rare event loading from seismic, severe water hammer, and other sources was not considered in this generic evaluation because of their strong dependency on plant specific factors. The assessment of these risk contributions was also beyond the expertise of the assembled panel members. As with other consequential LOCAs, the LOCA frequency estimation from the rare event loading may be an important consideration when evaluating total plant risk and determining total LOCA risk contributions.

An important assumption implicit in the elicitation is that the plant construction and operation comply with applicable codes and standards. Therefore, all LOCA-sensitive passive-system components were designed and fabricated using approved materials in accordance with ASME or similarly applicable requirements. Component inspections are also conducted as per generic ASME Section XI periodicity and quality requirements. Other additional inspections required by the NRC to evaluate specific degradation mechanisms (e.g., IGSCC or CRDM inspections) are also assumed to occur as directed. Plant operation is generally assumed to occur within the expected parameters allowable by the regulations and the technical specifications. The specific impact of counterfeit materials and fabrication techniques, avoidance of inspection, and blatant deviation from approved operating criteria on the LOCA frequencies has not been determined.

Another implicit assumption in the elicitation is that the future plant operating characteristics are assumed to be essentially consistent with past operating practice. The effects of operating profile changes have not been considered because of the large uncertainty surrounding possible operational changes and the potentially wide-ranging ramifications of significant changes on the underlying LOCA frequencies. For instance, significant power upgrade allowances may change plant performance and relevant operating characteristics (e.g., temperature, environment, flow rate, etc.) to a degree which significantly impacts future LOCA frequencies. Because operating experience provides fundamental information used to determine the LOCA frequency estimates, these types of changes undermine the applicability of the operating experience data and also the LOCA frequency estimates. The interrelationship between plant operating profile and LOCA frequencies should be considered when evaluating possible risk-informed changes based on these elicitation results.

3. ELICITATION APPROACH

The expert elicitation process used with the expert panel is an adaptation of the formal expert judgment processes used in NUREG-1150 [3.1] to estimate core damage frequencies [3.2] and to assess the performance of radioactive waste repositories [3.3]. The expert elicitation process used for this project consisted of a number of steps (Figure 3.1). To begin, the project staff identified the issues to be evaluated through a pilot elicitation (Block 2 in Figure 3.1) and selected a panel of twelve members (Block 1). The staff then gathered background material and prepared an initial formulation of the technical issues (Block 3) that was provided to the panel. At its initial meeting, the panel discussed the issues and, using the staff formulation as a starting point, developed a final formulation for the elicitation (Block 3). This formulation included an elicitation structure (Block 5) for decomposing the technical issues and the development of base cases (Block 6) which are used in the subsequent anchoring of all the elicitation responses. A base case team was established (Block 7) as a subset of the entire panel to estimate the LOCA frequencies associated with the base case conditions. At this initial meeting, elicitation training (Block 4) was also conducted using exercises and a discussion of biases to educate the panel about the subjective elicitation of numerical values. After the first meeting, the base case team developed preliminary estimates for the base case frequencies. The staff also prepared a draft elicitation questionnaire (Block 8) which was reviewed by the panelists (Block 9) and revised based on comments received (Block 8). A second meeting was held to review the base case estimates, review the elicitation questions, and finalize remaining technical issue formulation issues. The elicitation questions and base case estimates were finalized based on feedback received at that meeting (Blocks 8 and 10).

At their home institutions, the individual panel members performed analyses and computations to develop their answers to the elicitation questionnaire. A facilitation team consisting of substantive experts, a normative expert and two recorders met separately with each panel member in day-long elicitation sessions (Block 11). At these sessions, each panel member provided answers to the elicitation questionnaire along with their supporting technical rationales (Block 12). The panel members then returned to their home institutions where they refined their responses based on feedback from the elicitation session. Upon receipt of the updated responses, the project staff compiled the panel's responses and developed preliminary estimates of the LOCA frequencies (Block 13). Along with the rationales, the preliminary estimates were presented to the panel at a third meeting (Block 14). Panel members were invited to fill in gaps in their questionnaire responses and, if desired, to modify any of their responses after this meeting (Block 15). Based on these updates, final estimates of the LOCA frequencies were calculated and provided to the panel members for review (Block 16). The project staff developed a draft report on the elicitation process and results (Block 17). A fourth meeting via video teleconferencing was held to discuss the draft report which was revised based on feedback received during this meeting (Blocks 19 and 20). Separately, an external peer review of the analysis of the elicitation responses from the panelists (Sections 5 and 6) was conducted (Block 18). In addition, public comment is being solicited (Block 21) prior to the development of the final LOCA frequencies to be used in the redefinition of the emergency core cooling system requirements in 10 CFR 50.46 (Block 22). More detail on each of these general steps is provided below.

3.1 Pilot Elicitation

The study was initiated with a pilot elicitation conducted using only NRC staff from the Office of Nuclear Regulatory Research (RES) and the Office of Nuclear Reactor Regulation (NRR) (Block 2). The primary objectives of this pilot study were to identify technical issues for consideration

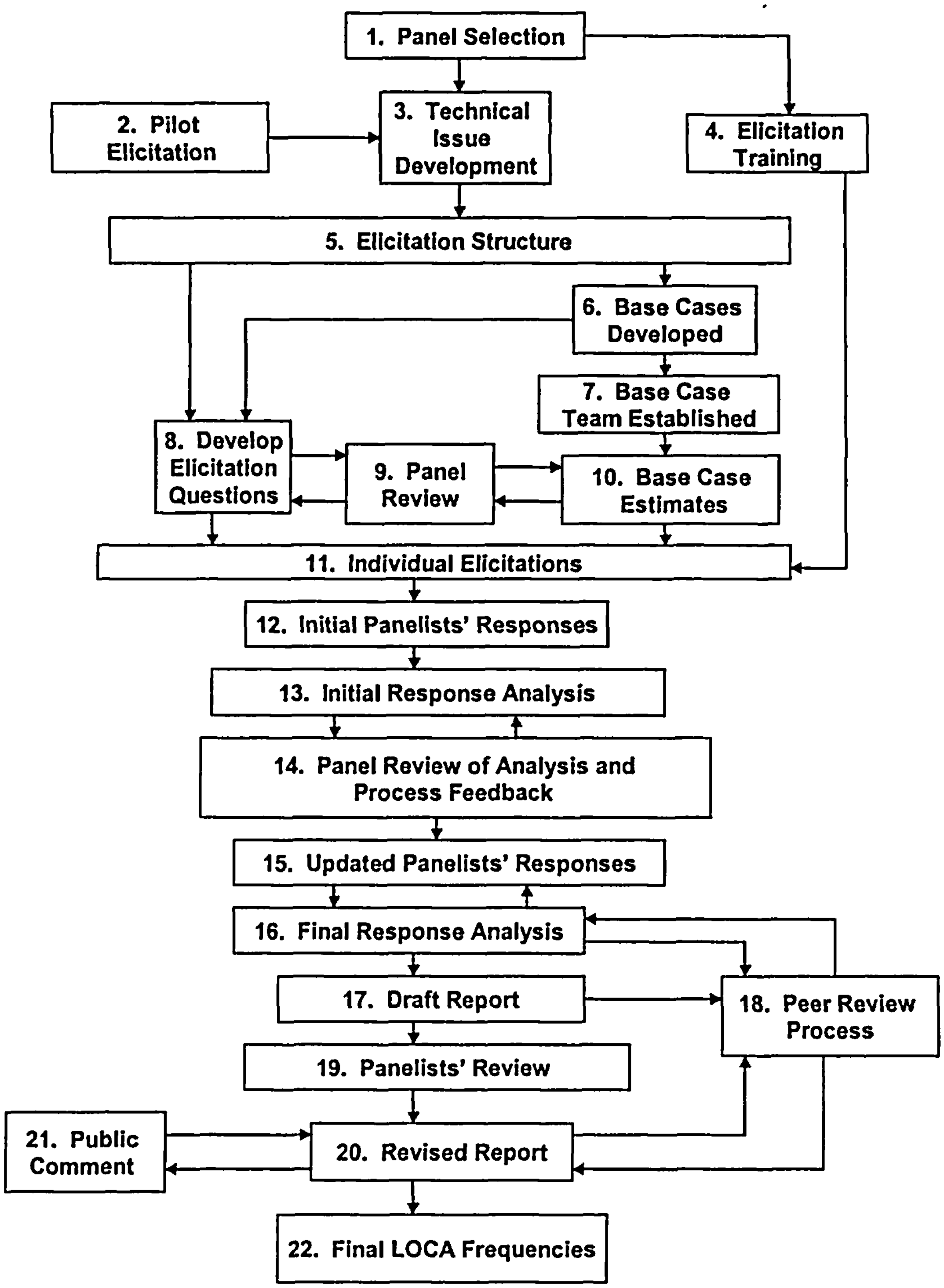

Figure 3.1 Flowchart of the Overall Elicitation Process

during the subsequent formal elicitation and test a possible elicitation framework. Additionally, interim LOCA frequency estimates were developed to support the study conducted by RES on the feasibility of risk-informing 10 CFR 50.46, Appendix K, and GDC 35. Specifically, estimates were sought to explore the potential of eliminating the design requirements to mitigate a simultaneous LOCA and loss-of-offsite-power (LOOP) event. The technical issues identified by the staff and discussed during this pilot study provided the foundation for subsequent discussion during the elicitation. The pilot elicitation was also valuable for identifying strengths and weaknesses in the process. The RES staff used the lessons learned from this pilot study to help formulate a suggested formal elicitation structure for subsequent consideration by the elicitation panelists. The results of this feasibility study and the staff's pilot elicitation have been reported [3.4], but a synopsis of salient points is provided below.

The pilot elicitation was structured similarly to the formal elicitation. Panelists were chosen to provide expertise in relevant technical areas. A kick-off meeting was held to discuss objectives and provide background information. An issue development meeting was then conducted to define the problem, identify important technical issues for consideration, develop elicitation questions, conduct elicitation training, and identify baseline LOCA frequencies for subsequent adjustment during the elicitation. The elicitation questionnaire was then independently answered by each panel member, the analyzed results were presented to the panelists, and the panelists provided feedback of the elicitation process.

As part of the issue development, the panelists developed a structure of classifying piping systems and non-piping components separately and identified corresponding materials, degradation mechanisms, and mitigation measures. The panelists then eliminated piping systems, non-piping components, and degradation mechanisms from consideration based on consensus opinion of those issues that were not important. The panelists utilized historical LOCA definitions and size classification from NUREG/CR-5750 [3.5] for consistency and agreed that the NUREG/CR-5750 LOCA frequencies would be the baseline for the exercise. The group evaluated the time period from the current period (2002) up to the end of the license extension period. The panelists were asked to determine the highest frequencies over this time period, but these maximums were typically associated with the end of the license extension period. Separate BWR and PWR estimates were developed. It should also be noted that the pilot study considered both active and passive system LOCA frequencies simultaneously.

The elicitation questions were structured so that each piping system or non-piping component was evaluated separately. There were two types of questions that provided redundant information. One type of question asked each participant to provide the relative percentage change between the NUREG/CR-5750 frequency estimates and the frequencies associated with each piping system or non-piping component. The other type of question asked for the relative ratio between MB and SB LOCAs and between LB and MB LOCAs for each piping system or non-piping component. There were also six global questions posed that were not related to any particular piping system. These included a consideration of the importance of the following issues on future LOCA frequencies: current IGSCC mitigation; hydrogen-combustion failures; future mitigation and degradation mechanisms; leak-detection-system accuracy; contribution of non-precursor degradation; and future in-service inspection (ISI) techniques. All questions were posed in a questionnaire and the results were analyzed and summarized.

There were several features of the pilot elicitation structure that were identified as advantageous for use during the formal elicitation. First, it was evident that classifying issues with respect to piping systems and non-piping components was appropriate. Next, it was valuable to identify those variable classes (e.g., material and degradation mechanisms) that affect the LOCA frequencies. Finally, it was useful to assign all relevant variables within each of these variable classes to each piping system and non-piping component. For example, all relevant materials and degradation mechanisms associated with the surge line were identified.

There were also several deficiencies uncovered in the pilot elicitation which required refinement prior to the formal elicitation. First, the understanding of elicitation questionnaire and the quality of the responses suffered in the pilot elicitation because there was no direct interface with each panelist during the completion of the elicitation questionnaire. Also, it was obvious that not all important variable classes had been appropriately identified during issue development and more comprehension was necessary. Next, the global questions should have been considered as a function of piping system, non-piping component, material, and degradation mechanism; and not independently from these variables. The use of NUREG/CR-5750 estimates to provide baseline values for adjustment was also realized as a limitation because the underlying conditions associated with these estimates were not sufficiently decomposed. Many panelists also thought that projecting estimates over a single 35-year period was too difficult and that breaking up the time period into shorter intervals would have made the assessment easier. Finally, it was evident that active and passive system LOCAs should be considered separately due to their different failure mode characteristics. All these deficiencies were addressed during the development of the formal elicitation process.

3.2 Panel Selection

Panel selection (Block 1) is a critical step in the process. The success of the elicitation is a direct result of the broad expertise of the panel members, and the ability of each member to provide corollary information in their specific areas of expertise. This information exchange enhances the general understanding of the remaining panel members. Potential panel members were sought within industry, academia, national laboratories, contracting agencies, other government agencies, and international agencies. Initially, a pool of 55 nominally qualified people was established by querying knowledgeable sources within the industry and NRC. Twenty-five people were solicited for the panel from this pool. They were sent information about the objective, scope, and approach of the elicitation exercise as background and were asked to submit resumes and also to evaluate their relevant technical areas of expertise for the exercise. Based on this feedback, the final panel of 12 was chosen to achieve both technical and organizational variety, and ensure a diversity of opinion, expertise, and backgrounds.

The elicitation panel members are listed in Table 3.1. The organizational diversity is apparent. Two of the panel members represent the European regulatory community; three of the panel members represent commercial vendors and owner's groups; four members are primarily NRC consultants; and three members have conducted extensive relevant research for both the commercial nuclear industry owner's groups and individual plants. Panel members were also chosen to represent a range of relevant technical specialties: probabilistic fracture mechanics (PFM), piping design, piping fabrication, operating experience, materials, degradation mechanisms, operating mitigation practices, stress analysis, nondestructive evaluation, etc. All panel members have at least twenty-five years of experience in these relevant technical areas pertaining to commercial nuclear power applications. Each member's relevant qualifications are summarized in Appendix A.

Table 3.1 LOCA Frequency Expert Panel

Panel Member	Organization
Mr. Bruce Bishop	Westinghouse Electric Co LLC
Dr. Vic Chapman	OJV Consultancy Limited
Mr. Guy DeBoo	Exelon Nuclear
Mr. William Galyean	Idaho National Engineering Environmental Laboratory
Dr. Karen Gott	Swedish Nuclear Power Inspectorate
Dr. David Harris	Engineering Mechanics Technology, Inc.
Mr. Bengt Lydell	ERIN® Engineering and Research, Inc.
Dr. Peter Riccardella	Structural Integrity Associates, Inc
Mr. Helmut Schulz	Gesellschaft für Reaktorsicherheit (GRS) mbh
Dr. Sampath Ranganath	Formally GE Nuclear Energy/Now XGEN Engineering
Dr. Fredric Simonen	Pacific Northwest National Laboratory
Dr. Gery Wilkowski	Engineering Mechanics Corporation of Columbus

A facilitation team was assembled to guide the expert panel through the elicitation process. The team consisted of one normative member, six substantive members, and two recorders. All but two of the facilitation team members were NRC staff. The substantive members were chosen to provide the same broad relevant technical knowledge and background required of the panel. The facilitation team role was to formulate the elicitation objectives and scope; coordinate and provide background technical information; develop the elicitation questions; guide and record the individual elicitation sessions; analyze and summarize the panel's findings; and develop the final LOCA frequency distributions from the panel's responses.

3.3 Elicitation Training

A basic premise in using an expert elicitation process is that the panel responses as a whole have no significant systematic bias. While individual responses can be highly uncertain and differ drastically, they do not systematically over- or underestimate the quantities of interest. The extensive panel discussions (Section 3.4), questionnaire structure (Section 3.8), and individual elicitations (Section 3.9) are designed to achieve this goal. Additionally, elicitation training is an important tool to eliminate or minimize bias. It is also can increase the accuracy and consistency of the responses provided by the panelists.

The elicitation training had three specific purposes:

(i) to discuss sources of bias in the elicitation procedure;

(ii) to familiarize the panelists with the type of responses which they will be asked to make; and

(iii) to provide the panelists with practice in making elicitation responses using a training exercise.

3.3.1 Motivational and Cognitive Biases

The panelists were introduced to sources of bias with the purpose of reducing biases in their individual subjective judgments. There are two sources of bias: motivational and cognitive. Motivational biases are due to emotional and psychological factors while cognitive biases are due to limitations on how information is processed by the human brain [3.6].

Motivational biases can result when a panelist's thoughts and responses are altered by the elicitation process. There are four types of motivational biases.

1. Social pressure can lead to groupthink when panelists may suppress their doubts or differing opinions in order to attain consensus. It can also be manifested in a panelist's response to verbal and non-verbal feedback from an interviewer. In addition, a panelist's responses might be influenced by his perception of what might be acceptable to his employer or society at large.
2. Misinterpretation can occur if the elicitation question structure is inconsistent with a panelist's thought process. For example, a panelist used to thinking in deterministic terms may be unsure how to respond if asked to think in probabilistic terms. Misinterpretation can also occur if a panelist is guided by the interviewer's viewpoint rather than his own.
3. Misrepresentation can be due to incorrect assumptions about the data or the models used to analyze the issue.
4. Wishful thinking can be the result of an institutional bias, e.g., a manager who underestimates the risk of a hazardous activity. This type of bias is relatively uncommon.

Cognitive biases can result when a panelist's thinking is illogical or does not conform to normative rules such as the axioms of probability. There are four types of cognitive biases.

1. Inconsistency can occur when there are multiple issues, assumptions, definitions or algorithms involved and a panelist does not keep them all in mind simultaneously. For example, inconsistency would occur if the panelist's sum of the probabilities of a set of mutually exclusive and exhaustive events does not add to one. Inconsistency is the most common cognitive bias.
2. An anchoring bias can occur when a panelist makes a relative comparison to a base case and does not sufficiently adjust his response with respect to the base case estimates.
3. An availability bias can occur when a panelist's opinion is overly influenced by the recent occurrence of a dramatic event, e.g., the accidents at TMI or Chernobyl.
4. Underestimation of uncertainty can occur when a panelist is asked for uncertainty bounds on his estimates. It is well-established that people are often overly-confident when dealing with highly uncertain issues [3.6], see Section 3.3.2.

3.3.2 Training Exercise

The training exercise (Block 4) consisted of asking the panelists a number of quantitative questions with known answers, but in a subject area with which they are relatively unfamiliar. The purpose of asking these "almanac-type" questions is twofold:

(i) to accustom the panelists to the types of responses that they will be required to provide in their elicitations, and

(ii) to demonstrate to the panelists that, although individually they may be highly uncertain about their responses, the group response is closer to the correct answer than the individual responses.

For each question, the panelists were asked to supply three numbers: a mid-value (MV), a high value (UB) and a low value (LB). The MV is defined such that, in the panelist's judgment, there is a 50% chance that the correct answer lays above the MV and a 50% chance that it lays below the MV. The UB and LB are defined such that there is a 5% chance that the correct answer lays above or below them, respectively. In other words, the MV corresponds to the median of the panelist's subjective distribution of the correct answer, and the UB and LB correspond to the 95th and 5th percentiles, respectively. The interval (LB, UB) is a subjective 90% coverage interval for the correct answer. The elicitation exercise consisted of four questions.

According to the 2000 census, how many American men age 65 or over were there in the U.S.?

1. How many Americans men age 65 or over suffered from the following chronic conditions in 1995: arthritis, cataracts, diabetes, hearing loss, heart disease, and prostate disease? Express your answer in terms of a rate per 1,000 men.
2. What is the ratio of the rate for men 45-64 years old to the rate for men 65 and older for each of the conditions listed in Question 2?
3. What is the ratio of the rate for men under 45 years old to the rate for men 45-64 years old for each of the conditions listed in Question 2?

Note that Questions 3 and 4 ask about ratios. This corresponds to the type of questions asked in the elicitation, where almost all the questions concern the ratios of LOCA frequencies with respect to the base case conditions.

The results of the elicitation training exercise were consistent with several of the basic premises underlying the elicitation structure and methodology (see Appendix C). Recall that the (LB, UB) intervals are supposed to contain the correct answer 90% of the time. One useful evaluation is to compare the actual coverage interval to its nominally prescribed value of 90%. This provides a measure of how well the panelists are calibrated. As presented in Appendix C, between 15 and 17 responses were provided for each of the four questions. Although there were only 12 members on the panel, members of the facilitation team were also invited to participate in the exercise. The results for the four questions above are as follows.

1. Question 1: Fourteen of 17 intervals (82%) covered the correct answer.
2. Question 2: Fifteen responders provided intervals for each of the six conditions. Of the 90 intervals, 55 (61%) covered the correct answer.
3. Question 3: Sixteen responders provided intervals for each of the six conditions. Of the 96 intervals, 69 (72%) covered the correct answer.
4. Question 4: Sixteen responders provided intervals for each of the six conditions. Of the 96 intervals, 68 (71%) covered the correct answer.

The result for the first question was close to the nominal 90%, most likely because this question dealt with demographic data for which the responders had a relatively good understanding. The results for the other three questions were consistent with the well-established observation that the actual coverage probability for a subjective interval is generally less than its nominal coverage probability [3.6]. In other words, people are generally overconfident and underestimate the uncertainty in their subjective estimates. (Sensitivity analyses examining the potential magnitude of this over confidence are discussed in Section 5.6.1).

The coverage intervals for the last two questions were more accurate than for the second question. This may stem from that fact that Questions 3 and 4 asked for relative ratios rather than absolute numbers. This trend is consistent with the assumption driving the basic structure of this elicitation, namely, that relative values are easier to assess than absolute values (see Section 3.8.1). See Appendix C for the correct answers to the training exercise, as well as a full discussion of the panel and facilitation team responses.

3.4 Technical Issue Formulation

The formal elicitation was begun in February 2003 with a three-day meeting of the expert panel and facilitation team. The five principal objectives of this meeting were to define the scope and objectives of the elicitation (Section 2); provide background information about previous LOCA frequency estimates (Section 1); construct an approach for determining LOCA frequencies (Section 3); identify significant technical issues affecting LOCA frequencies (Section 3.4); and conduct elicitation training (Section 3.3). Subsequent discussion to finalize technical issues also was held in a follow-up two day meeting in June 2003.

3.4.1 Important Definitions

The first step in the decomposition was to define key technical terms and issues (Block 3), including the definition of a LOCA to ensure consistent understanding within the panel. For the purpose of this elicitation, the panel members defined a LOCA as:

> "A breach of the reactor coolant pressure boundary which results in a leak rate beyond the normal makeup capability of the plant."

The panel next defined various LOCA size categories (Table 3.2). The LOCA size categories are largely consistent with historical definitions developed for small break (SB), medium break (MB), and large break (LB) LOCAs during the WASH-1400 evaluation [3.7]. These definitions were retained in subsequent exercises to characterize plant risk [3.1] and determine initiating event frequencies [3.5]. One distinction is that, historically, break size frequencies were defined over a range of flow rates for SB (100 to 1,500 gpm [380 to 5,700 lpm]) and MB (1,500 to 5,000 gpm [5,700 to 19,000 lpm]) LOCAs. In this exercise, the panel chose to work with threshold values for each LOCA category. Additionally, three additional categories which fall within the historical LB LOCA regime were defined. These additional categories were developed to evaluate frequencies associated with the larger break sizes up to the DEGB of the largest primary system piping. These additional categories also reflect regions where different plant responses may be required to mitigate LB LOCA events of increasing size. LOCA Category 6 was chosen to correspond to the flow rate which would result from the complete rupture of the largest PWR primary piping system in the plant, i.e., a hot leg. LOCA Categories 4 and 5 were determined so that the ratios between successive LB LOCA threshold values were approximately equal between Categories 3 - 6. The flow rate increases by either a factor of 4 or 5 for each successive category between Categories 3 – 6.

Table 3.2 LOCA Category Definitions

LOCA Category	Flow Rate Threshold (gpm)	LOCA Classification
1	> 100	SB
2	> 1,500	MB
3	> 5,000	LB
4	> 25,000	LB a
5	> 100,000	LB b
6	> 500,000	LB c

In addition to considering different LOCA sizes, the panel members also considered the possibility that future LOCA frequencies may be time dependent. Three different time periods were defined: 0-25, 25-40, and 40-60 years of plant operation. The 0-25 year period is representative of the current plant experience considering that the current fleet average age is approximately 25 years. Frequency estimates at 25 years are synonymous with the current day LOCA frequency estimates. The 25-40 year period is representative of future operation up to the average end of the original design life and original plant licenses. This period allows LOCA frequency changes over the next 15 years to be evaluated. The 40-60 year period is representative of future operation between the expiration of the original plant license up through the current plant license extension period. This period evaluates LOCA frequency changes within the next 35 years. Panel members could choose to consider either average values over these time periods, maximum values at a point in time within the time period, or values representative of the end of the time period. No distinction was made because LOCA frequency differences using any of these definitions are not expected to be significant.

The panel members considered issues that affect both piping and non-piping passive system failures. The contributions from active system components (e.g. stuck open valves, gasket and seal LOCAs) were not considered. The contributions from active components should be assessed separately and added to the passive system frequencies developed in order to estimate the total LOCA initiating event frequencies from both passive and active component failures. The panel members used existing ASME rules to identify the boundaries between piping and non-piping components. However, this distinction was somewhat academic since total LOCA estimates (i.e., sum of the piping and non-piping contributions) were determined from individual panelist responses. Therefore, piping and non-piping boundary differences among the panelists do not affect the final estimates.

3.4.2 Safety Culture Issue Formulation

Issues related to the effect of safety culture on the LOCA frequencies were often raised during technical issue formulation (Block 3). While no panelists are organizational safety culture experts, they all have relevant experience and opinions on how safety culture variability can affect LOCA frequencies. The panelists believe that safety culture affects are only weakly correlated with other variables that influence LOCAs, including the effects of aging mechanisms. These considerations were also thought to be independent of specific piping systems and non-piping components. Therefore, the panelists decided to consider the effect of safety culture LOCA contributions independently from age-related contributions.

For the purposes of the elicitation, safety culture was defined to encompass a number of economic, political, social, and psychological issues. Issues discussed fell into the broad categories of deregulation, decommissioning, human error, technology transfer, lessons learned from operating experience, public perception, and management philosophy. The objective of the discussion was to provide the panelists with a clear understanding of the safety culture issues for assessment during their individual elicitations. Some specific issues discussed include the likelihood that human error can occur during maintenance and mitigation operations and impact LOCA frequencies. An example is the failure to adhere to proper procedures or misinterpretation of obvious inspection indications of degradation. The effect of an aging workforce and the ability to provide sufficient technology transfer to the less experienced replacement workers was also discussed. The effect of lessons learned from past experiences on decreasing the response time needed to mitigate the impact of new degradation mechanisms was also debated. For example, the industry experience with mitigating IGSCC in the early 1980's may provide some useful strategies for mitigating current PWSCC concerns. Additionally, with respect to management philosophy, the merits of the adoption and implementation of a risk-informed management strategy were discussed. These and other issues were addressed by each panelist within the individual elicitation.

The panel also decided at this time to consider industry and regulatory safety cultures separately. The elicitation structure was defined so that each panelist only addressed relative changes in the future safety culture resulting from issues raised in the discussion. Possible future changes are assessed relative to the current existing safety culture. While the utility and regulatory effects were considered separately, it was also clarified that the relationship between the utility and regulatory environments could be strongly correlated. This fact was considered in the assessment, as well as any relationship between safety culture and LOCA-size category. The effect of the more important safety culture issues on LOCA frequencies is discussed in qualitative rationale (Section 6.2) and quantitative results (Section 7.1) synopsis.

3.4.3 General Issue Development Structure

The remainder of the elicitation focused solely on the contribution of aging-related passive system degradation to the LOCA frequency estimates. The panel developed a structure (Block 5) for considering passive system failures which contribute to LOCAs (Figure 3.2) due to normal operating conditions and loading, and transients expected over the extended plant operating life. The total passive system frequencies were first divided by the panel into piping and non-piping contributions. The panel next agreed that the design and operating characteristics of each piping system and each major non-piping component (e.g., main coolant pumps, steam generators, pressurizer, and valves for PWRs) could impact the underlying LOCA frequencies. Non-piping components were further subdivided into relevant subcomponents (e.g., valve bonnet, valve bonnet bolts, valve casing, etc.) that possess unique operating and design characteristics. For a given LOCA-sensitive piping system or non-piping subcomponent, the panel identified five variable classes (i.e., geometry, loading history, materials, aging mechanisms, and mitigation and maintenance practices) that contain all the principal variables that affect passive system LOCA frequencies.

The elicitation structure (Figure 3.2) developed by the panel provided flexibility to the LOCA frequency estimation process. Panel members could develop LOCA frequencies based on either piping system or non-piping component level information (top-down approach in Figure 3.2), or by considering the individual effects of specific variable combinations within a given system or component and then combining important contributions (bottom-up approach in Figure 3.2). Separate elicitation questions were developed for each analysis approach (Section 3.8.4). The panel members either used one of these two approaches, a combined approach, or an alternative approach in their elicitation responses. More specific details on the issue formulation follow. Other information is available in Appendix B.

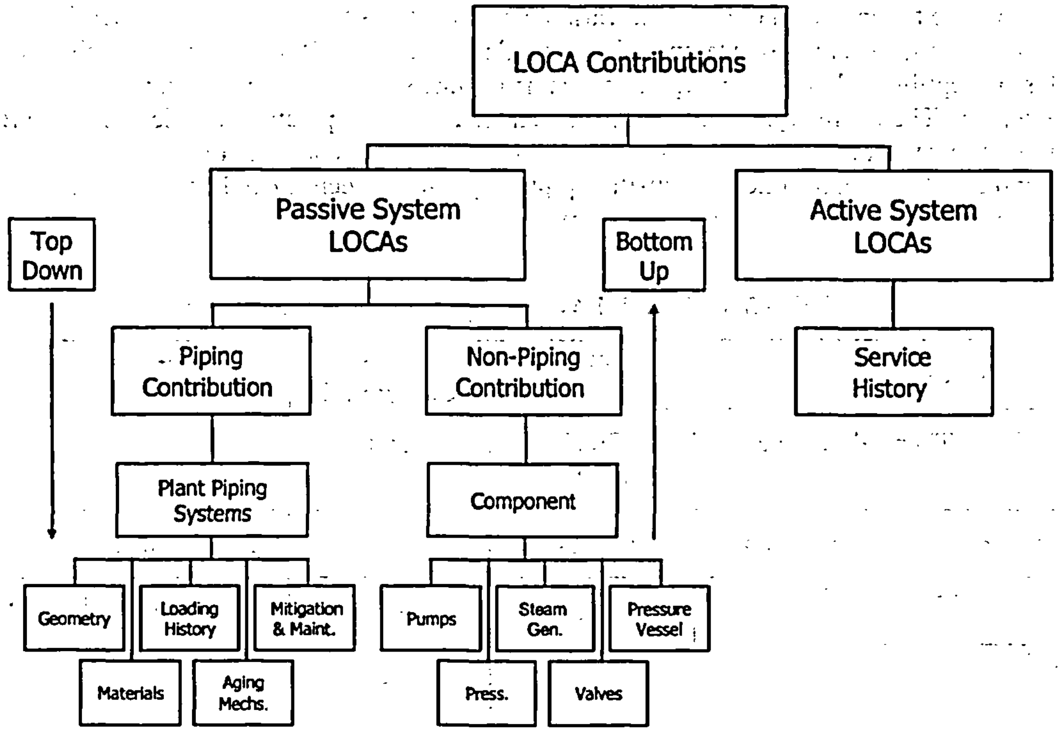

Figure 3.2 Elicitation Structure

3.4.4 Variable Assignments

The panel then developed a list of relevant variables within each of these variable classes that could affect the LOCA frequencies. For instance, under the aging mechanism variable class, mechanisms that both have been experienced or could surface in the future were identified (Table 3.3). Table 3.3 summarizes the mechanisms developed during the group brainstorming sessions. The mechanisms are segregated by the primary mechanism type. Additional sub-categories are used to identify either specific degradation mechanisms under the appropriate main category or features associated with the main category. For example, fatigue degradation was separated into low cycle fatigue which is primarily driven by thermal loading fluctuations due to plant heat-up and cool-down cycles and high cycle mechanical fatigue which could result from general loading functions on the piping, possibly due to an adjacent source of vibration. Both crack initiation and crack growth portions of life are important contributors to fatigue life, although crack initiation occupies a greater percentage of the life in the high-cycle fatigue regime. Stress corrosion cracking (SCC) is listed as a main category and it includes intergranular SCC (IGSCC) which was prevalent in BWRs in the late 1970's, transgranular SCC (TGSCC) which affects casting components, primary water SCC (PWSCC) which has more recently surfaced in PWRs, and external chloride SCC (ECSCC). Similar discussions ensued at the kick-off meeting for identifying the remaining degradation mechanisms listed in Table 3.3. A synopsis of those discussions can be found in the detailed minutes for the kick-off meeting in Appendix B. In addition, more details on the development of the variables within the other relevant piping system variable classes, i.e., geometry, loading, etc. is provided in Appendix B along with complete tables for all five of these variable classes.

3.4.5 Piping System Variable Combinations

The panel next identified those specific variable combinations that are active for each LOCA-sensitive piping system. For example, the recirculation piping in BWR plants is stainless steel and is susceptible only to certain aging mechanisms, like stress corrosion cracking, thermal fatigue, etc. The geometry, loading spectrum, and maintenance practices applicable to the BWR recirculation system were also defined. A similar exercise was conducted for all the BWR (Table 3.4) and PWR (Table 3.5) LOCA-sensitive piping systems to identify and correlate the relevant variables associated with each piping system.

Table 3.3 Material Aging Degradation Mechanisms

Main Category	Sub-Category 1	Sub-Category 2	Sub-Category 3	Sub-Category 4
Low Cycle Thermal Fatigue	Crack Initiation	Crack Growth		
High Cycle Mechanical Fatigue	Vibration	Pressure	Temperature	
Stress Corrosion Cracking	IGSCC	TGSCC	ECSCC	PWSCC
Localized Corrosion	Pitting	Crevice Corrosion		
General Corrosion	Boric Acid (ID or OD)			
Fretting Wear				
Material Aging	Thermal	Dynamic	Radiation	Creep
Fabrication Defects and Repair				
Hydrogen Embrittlement				
Flow Sensitive	Erosion/ Cavitation	FAC		
Unanticipated (New) Mechanisms				

IGSCC = Integranular stress corrosion cracking
TGSCC = Transgranular stress corrosion cracking
ECSCC = External chloride stress corrosion cracking
PWSCC = Primary water stress corrosion cracking
FAC = Flow accelerated corrosion

Table 3.4 BWR LOCA-Sensitive Piping Systems

System	Piping Matls.	Piping Size (in)	Safe End Matls.	Welds	Sig. Degrad. Mechs.	Sig. Loads.	Mitigation/ Maint.
RECIRC	304 SS, 316 SS, 347 SS	4, 10, 12, 20, 22, 28	304 SS, 316 SS, A600	SS, NB	UA, FDR, SCC, LC, MA	RS, P, S, T, DW, SUP, SRV, O	ISI w TSL, REM
Feed Water	CS	10, 12 (typ), 12 - 24	304 SS, 316 SS	CS, NB	UA, FDR, MF, TF, FS, LC, GC, MA	T, TFL, WH,P, S, SRV, RS, DW, O	ISI w TSL, REM
Steam Line	CS – SW	18, 24, 28	CS	CS	UA, FDR, FS, GC, LC, MA	WH, P, S, T, RS, DW, SRV, O	ISI w TSL, REM
HPCS, LPCI	CS (bulk), 304 SS, 316 SS	10, 12	304 SS, 316 SS, A600	CS, SS, NB	UA, FDR, SCC, TF, LC, GC, MA	RS, T, P, S, DW, TS, WH, SUP, SRV, O	ISI w TSL, REM
RHR	CS, 304 SS, 316 SS	8 - 24	CS, 304 SS, 316 SS	CS, SS, NB	UA, FDR, SCC, TF, FS, LC, GC, MA	RS, T, P, S, DW, TS, O SUP, SRV	ISI w TSL, REM
RWCU	304 SS, 316 SS, CS	8 – 24	CS, 304 SS, 316 SS	CS, SS, NB	UA, FDR, SCC, TF, FS, LC, GC, MA	RS, TS, T, P, S, DW, SUP, SRV, O	ISI w TSL, REM
CRD piping	304 SS, 316 SS (low temp)	< 4	Stub tubes – A600 and SS*	Crevice A182 to head	UA, FDR, MF, SCC	RS, T, P, S, DW, V, O, SRV	ISI w TSL, REM
SLC	304 SS, 316 SS	< 4	304 SS, 316 SS	SS, NB	UA, FDR, MF, SCC	RS, T, P, S, DW, V, O, SRV	ISI w TSL, REM
INST	304 SS, 316 SS	< 4	304 SS, 316 SS	SS, NB	UA, FDR, MF, SCC, MA	RS, T, P, S, DW, V, O, SRV	ISI w TSL, REM
Drain lines	304 SS, 316 SS, CS	< 4	304 SS, 316 SS, CS	SS, NB	UA, FDR, MF, SCC, LC, GC	RS, T, P, S, DW, V, O, SRV	ISI w TSL, REM
Head spray	304 SS, 316 SS, CS	< 4	304 SS, 316 SS, CS	SS, NB	UA, FDR, SCC, TF, LC, GC	RS, P, S, T, DW, SRV, O	ISI w TSL, REM
SRV lines	CS	6, 8, 10, 28	CS	CS	UA, FDR, MF, FS, GC, LC, MA	RS, P, S, T, DW, SRV, O	ISI w TSL, REM
RCIC	304 SS, 316 SS, CS	6, 8	304 SS, 316 SS	SS NB	UA, FDR, SCC, LC, MA	RS, P, S, T, DW, SRV, O	ISI w TSL, REM

304 SS = 304 series stainless steel
316 SS = 316 series stainless steel
347 SS = 347 series stainless steel
A600 = Alloy 600
CS = carbon steel
CS – SW = seam welded carbon steel
NB = Nickel-based weld (Alloy 82/182)
UA = unanticipated mechanisms
MA = material aging
LC = local corrosion
FDR = fabrication defect and repair
SCC = stress corrosion cracking
MF = mechanical fatigue
TF = thermal fatigue
FS = flow sensitive (inc. FAC and erosion/cavitation)
ISI w TSL = Current ISI procedures with technical specification leakage detection requirements considered.
RS = residual stress

P = pressure
S = Seismic
T = Thermal
DW = dead weight
SUP = support loading
SRV = SRV loading
WH = water (and steam) hammer
O = overload
V = vibration
TFL = thermal fatigue loading from striping
TS = thermal stratification
REM = all remaining mitigation strategies possible (eg. *not unique to piping system)*

Table 3.5 PWR LOCA-Sensitive Piping Systems

System	Piping Matls.	Piping Size (in)	Safe End Matls.	Welds	Sig. Degrad. Mechs.	Sig. Loads.	Mitigation/ Maint.
RCP: Hot Leg	304 SS, 316 SS, C-SS, SSC-CS CS – SW	30 - 44	A600, 304 SS, 316 SS, CS	A82 304 SS, 316 SS, CS	TF, SCC, MA, FDR, UA	P, S, T, RS, DW, O, SUP	ISI w TSL, REM
RCP: Cold Leg/Crossover Leg	304 SS, 316 SS, C- SS, SSC-CS, CS – SW	22 - 34	A600, 304 SS, 316 SS, CS	A82 304 SS, 316 SS, CS	TF, SCC, MA, FDR, UA	P, S, T, RS, DW, O, SUP	ISI w TSL, REM
Surge line	304 SS, 316 SS, C-SS	10 - 14	A600, 304 SS, 316 SS,	A82 304 SS, 316 SS	TF, SCC, MA, FDR, UA	P, S, T, RS, DW, O, TFL, TS	TSMIT, ISI w TSL, REM
SIS: ACCUM	304 SS, 316 SS, C-SS	10 - 12	A600, 304 SS, 316 SS,	A82 304 SS, 316 SS	TF, SCC, MA, FS, FDR, UA (FAC)	P, S, T, RS, DW, O	ISI w TSL, REM
SIS: DVI	304 SS, 316 SS	2 – 6	A600, 304 SS, 316 SS,	A82 304 SS, 316 SS	TF, SCC, MA, FS, FDR, UA (FAC)	P, S, T, RS, DW, O	ISI w TSL, REM
Drain line	304 SS, 316 SS, CS	< 2"			MF, TF, GC, LC, FDR, UA	P, S, T, RS, DW, O, V, TFL	ISI w TSL, REM
CVCS	304 SS, 316 SS	2 – 8	A600 (B&W and CE)	A82	SCC, TF, MF, FDR, UA	P, S, T, RS, DW, O, V	ISI w TSL, REM
RHR	304 SS, 316 SS	6 – 12			SCC, TF, MA, FDR, UA	P, S, T, RS, DW, O, TFL	ISI w TSL, REM
SRV lines	304 SS, 316 SS	1 – 6			TF, SCC, MF, FDR, UA	P, S, T, RS, DW, O, SRV	ISI w TSL, REM
PSL	304 SS, 316 SS	3 – 6		A82	TF, SCC, MA, FDR, UA	P, S, T, RS, DW, O, WH, TS	ISI w TSL, REM
CRDM	A600	4 – 6			SCC, TF, MF, LC, FDR, UA	P, S, T, RS, DW, O	HREPL, ISI w TSL, REM
RH	304 SS, 316 SS	< 2	A600		MF, SCC, TF, FDR, UA	P, S, T, RS, DW, O, V, TS	ISI w TSL, REM
ICI	304 SS, 316 SS	< 2	A600		MF, SCC, TF, FW, FDR, UA	P, S, T, RS, DW, O, V	ISI w TSL, REM
INST	304 SS, 316 SS	< 2			MF, SCC, TF, FDR, UA	P, S, T, RS, DW, O, V	ISI w TSL, REM

304 SS = 304 series stainless steel
316 SS = 316 series stainless steel
A600 = Alloy 600
A82 = Alloy82
SSC-SC = Stainless steel clad carbon steel
CS = carbon steel
CS – SW = seam welded carbon steel
UA = unanticipated mechanisms
MA = material aging
GC = general corrosion
LC = local corrosion
FDR = fabrication defect and repair
SCC = stress corrosion cracking
MF = mechanical fatigue
TF = thermal fatigue
FS = flow sensitive (inc. FAC and erosion/cavitation)
ISI w TSL = Current ISI procedures with technical specification leakage detection requirements considered.
B&W = Babcock and Wilcox
CE = Combustion Engineering
RS = residual stress loading

P = pressure loading
S = seismic loading
T = thermal loading
DW = dead weight loading
SUP = support loading
SRV = SRV loading
WH = water (and steam) hammer
O = overload
V = vibration
TFL = thermal fatigue loading from striping
TS = thermal stratification
REM = all remaining mitigation strategies possible (eg. not unique to piping system)
TSMIT = thermal stratification mitigation
HREPL = RPV head replacement

Table 3.6 Reactor Pressure Vessel (RPV) Failure Modes

Failure Modes	Material	Degradation Mechanisms	Mitigation/ Maintenance	Comment
Vessel Head Bolts	high strength steel	LC	Human error	Removal leading to human error during refueling
CRDM connections	SS			welded, bolted, threaded + seam weld
Nozzles	SSC-CS			
RPV wastage	SSC-CS	LC		Boric acid wastage (upper & lower head, shell)
RPV Corrosion Fatigue	SSC-CS	MF		Initiate at cladding cracks (upper & lower head, shell)

RPV= reactor pressure vessel
SS = stainless steel
SSC-CS = stainless steel clad carbon steel
LC = local corrosion
GC = general corrosion

3.4.6 Non-Piping System Failure Modes

A similar exercise was conducted for the non-piping LOCA-sensitive components and subcomponents. Table 3.6 illustrates the variables considered important for the reactor pressure vessel (RPV) in BWRs and PWRs. The panel identified failures in the RPV shell (due to wastage or corrosion fatigue), the head bolts, the CRDM connections, and the nozzles as passive system LOCA-sensitive RPV subcomponents. The group generally did not have information on component geometries, loading, and mitigation/maintenance practices at this meeting and salient information was researched later as necessary by each panelist prior to the elicitation. However, several specific failure scenarios were discussed for each of these subcomponents. An RPV shell failure could occur by boric acid wastage from the outer shell or due to corrosion fatigue within the shell emanating from fissures in the stainless steel cladding. The CRDM connections are those that connect to the reactor CRDM nozzles. Failure would imply a failure of this connection which could be bolted, welded, or threaded depending on the RPV design. Vessel head bolt failure is only thought possible due to either human error during refueling as the head is removed and reinstalled or a common cause degradation mechanism such as boric acid corrosion that is occurring simultaneously in several bolts. Nozzle failure includes both the larger main coolant loop nozzles, as well as the smaller CRDM and instrumentation penetration nozzles. Nozzle failure is possible due to SCC or thermal fatigue. Other RPV failures were discussed as well, and similar discussions were held for all the LOCA-sensitive non-piping components and subcomponents. See Appendix B for more details, and for similar tables for other non-piping components. While these initial failure scenarios were discussed for all the non-piping components during the issue development meeting, the panelists were invited to consider other possible failure scenarios that should be considered during the elicitation.

3.4.7 Reference Case Development

Reference cases were developed for each BWR and PWR piping system by assigning specific values to each variable category (e.g., material, geometry, loading, mitigation, and aging mechanism) within Tables 3.4 and 3.5. Similar cases were also developed for the PWR piping systems (Appendix B). The reference cases were defined to create a simplified set of conditions that could be used as a starting point to assess LOCA frequencies within each system. The reference cases are representative of possible conditions that could lead to a LOCA, but they do not necessarily represent the most important LOCA contributor for a given piping system. The actual BWR and PWR reference cases are described in Appendix B. The development of the reference case conditions is analogous to the development of the base case conditions (Section 3.5). However, the reference cases are not associated with a quantified LOCA frequency. See Section 3.5 for more information on the base-case development. More information on the use of the reference cases within the elicitation is provided in Section 3.8.4.

3.5 Base Case Development

The elicitation structure provided the panel members with a way to prioritize and assess important contributing variables to the generic LOCA frequency distributions. The most challenging aspect for each panelist was to quantify the frequencies associated with the important contributing variables that they individually identified. In order to improve the accuracy of the assessment, piping and non-piping base case frequencies were developed to provide the panelists with quantitative estimates for a set of well defined conditions that they could use for anchoring their responses. A clearly defined set of conditions and assumptions were developed by the elicitation panel for each base case (Block 6). These conditions were then analyzed by a member of the base case team (Block 7) using either PFM or classical statistical and/or Bayesian assessment of service experience data to predict the frequencies associated with each piping and non-piping base case.

This development and use of base cases was employed so that the panelists never had to provide absolute frequencies during the elicitation. Instead, they chose appropriate base cases and provided a relative ratio to express the difference between the base cases and the important contributing variables that they identified. This technique is based on the premise that relative ratios are easier to assess, and therefore more accurate, than absolute numbers. The rationale for this premise is also discussed in Sections 3.3.2 and 3.8.1. Note, however, that the decision to use and the application of these base cases was made by each individual panelist. Some panelists utilized them extensively, others considered the relative trends expressed by the base case estimates, and others chose not to utilize them at all.

3.5.1 Piping Base Cases

The objective of the piping base case study was to analyze various PWR and BWR piping systems and materials using both operating experience and fracture mechanics analyses with shared information and a common set of conditions and assumptions. The insights from these analyses were provided to the panelists to help them formulate their opinions on the comparative LOCA likelihood between the base case conditions and other systems and degradation mechanisms not explicitly modeled. It was impractical to model all possible LOCA sensitive components, materials, and degradation mechanisms. Therefore, the goal in the base case development was to consider several important variables in BWR and PWR plants including piping size, aging mechanisms, and piping materials. Several aging mechanisms were chosen based on operating experience and a consideration of those mechanisms that provide the greatest LOCA challenge. A range of piping sizes were evaluated to provide information on the spectrum of pipe break sizes considered in the elicitation. The most common piping materials and welds were considered. Finally, sensitivity analysis was conducted to identify the effects of certain variable and modeling assumptions.

3.5.1.1 Piping Base Definition - Five different base cases were defined for the piping systems: two BWR and three PWR cases (Block 6). The variables established for analysis are summarized in Table 3.7. The loading spectrum for each of these systems was assumed to be the normal steady-state and transient loading histories expected over 60 years of operation. Maintenance and mitigation was assumed to be typical in-service inspection (ISI) for all systems with leak detection resolution as required by the technical specification limits. The BWR systems were assumed to be operating with normal water chemistry (NWC) and the recirculation system base case assumed that a weld overlay (WO) had been applied after 20 years of operations to mitigate IGSCC. The inspection technique and periodicity for the recirculation base case are as specified in generic letter (GL) 88-01 [3.8]. Separate LOCA frequency estimates associated with each of the piping bases cases were calculated for each defined LOCA category at 25, 40, and 60 years of plant operations.

Table 3.7 Piping Base Case Definitions

Plant Type	System	Pipe Size (in)	Pipe Material	Safe End Material	Weld Material	Aging Mechanisms	Maintenance & Mitigation
BWR	Recirculation	12, 28	304 SS	NB	NB	IGSCC	ISI, NWC, WO
	Feedwater	12	CS	NA	NA	FAC, TF	ISI, NWC
PWR	Hot leg	30	304 SS	NB	NB	PWSCC	ISI
	Surge line	10	304 SS	NB	NB	PWSCC, TF	ISI
	HPI make-up	4	CS or SS	NA	NA	TF	ISI

HPI = high pressure injection
CS = carbon steel
TF = thermal fatigue
SS = stainless steel
NB = nickel-based
NA = not applicable

3.5.1.2 Piping Base Estimation - Four panelists were selected to separately calculate the frequencies associated with each base case defined in Table 3.7: Victor Chapman, William Galyean, David Harris, and Bengt Lydell (Block 7). These four panelists are referred to as the base case team. Both Chapman and Harris were selected to utilize a PFM-based approach, while Galyean and Lydell were chosen to utilize Bayesian and other statistical analysis techniques to extrapolate frequencies from accumulated service data.

There were several ground rules established by the elicitation panelist for the base case frequency estimations. These are enumerated as follows:

- Base case members should collaborate and use common information as the basis for the estimates.
- All base case calculations should model the established conditions (Table 3.7) as closely as possible so that methodologies and results can be more easily compared.
- Each base case team member could choose any methodology for estimating the base case frequencies.
- The PFM analysis should model service conditions as closely as possible. In particular, the predicted crack leaking rates should be commensurate with operating experience.
- The base case team members should provide best estimate base case frequencies. Explicit uncertainty calculations were not required.
- Frequencies should be calculated for the entire piping system, and not just on a per weld basis.
- Some additional sensitivity analyses should be conducted to evaluate the effect of variables not modeled in the base case calculations.

Separate calculations were conducted by each base case team member to establish a range of base case frequencies. As mentioned, these absolute frequencies are used to anchor panelist responses to the elicitation questions. However, the differences among the results stemming from the four different calculations provided valuable insights as well. The elicitation panelists, by understanding the details of each approach, could identify the qualitative and quantitative effects of important assumptions and modeling differences. The panelists could also assess the underlying uncertainties associated with each calculation methodology. The sensitivity results could also be used to assess the impact of other variables on the base case frequencies.

Periodic meetings were held among the base case team members. The purpose of these meetings was to identify and share necessary background information, resolve technical ambiguities, discuss and resolve problems, and review progress. It was apparent early in the process that the Table 3.7 base case definitions were not sufficient to fully perform the calculations. Therefore, additional technical details were standardized including the piping layout and configuration, weld census, applied stress information, and the model BWR and PWR plant types. Additionally, the service history data associated with the base case piping systems was available to the team members through a searchable database developed by SKI in 1998 [3.9]. The BWR base cases assumed a BWR/4 plant type. The PWR hot leg and surge line base cases were modeled based on a three-loop Westinghouse plant while the HPI make-up line is representative of a Babcock & Wilcox (B&W) plant. More details about the refined base case definitions and a summary of background information are provided in Appendices D – G.

Each base case team member then conducted initial base case calculations and sensitivity analyses. The initial sensitivity analyses included the effect of ISI, the impact of seismic events at 25, 40, and 60 years of plant operation for some of the systems, the effect of proof testing, and the effect of weld overlay on the BWR-1 base case, i.e., recirculation system. More details about the individual calculations and sensitivity analyses are contained in Appendices D – G. Once the initial base case frequencies were obtained, results were shared among the base case team members and reasons for differences were discussed.

Then, each base case team member prepared a presentation to summarize his calculation methodology and results. The goal of these presentations was to make the underlying details of each calculation apparent and provide the remaining panelists with sufficient information to independently assess the calculations in support of their elicitation responses (Block 9). Therefore, the presentations had identical formats so that the various approaches could be easily compared. The presentations explained the general assumptions, general approach, input variable determination and assumptions, detailed calculation procedure, results, and results of sensitivity analyses. Additionally, some major differences among the four methodologies were summarized. A summary of the approaches followed by the four base case team members in developing their piping base case frequencies is provided next.

3.5.1.2.1 Bengt Lydell's Base Case Estimates - This analysis was one of two base case estimates using available service history data to develop the base case frequencies. Base Case Report 2 (see Appendix D) develops BWR and PWR LOCA frequency distributions using a "bottom-up approach". Statistical analysis of relevant service experience data is used to quantify the precursor failure rate and rupture frequency (i.e., LOCA frequency) of individual welds. Here the precursor failure rate is defined as the rate at which particular degradation is found and repaired/replaced in service based on operating experience. This is an attribute-influence approach in that the statistical analysis of service experience data is restricted to consideration of the unique material combinations and degradation susceptibilities of primary pressure boundary piping systems. As there are few relevant events to estimate the rupture frequency in a purely statistical manner, engineering judgment is required in the estimation of the conditional rupture probabilities given that a precursor failure exists. The rupture frequency is then simply the precursor failure rate multiplied by the conditional rupture probability. The failure rate and rupture frequency (i.e. LOCA frequency) for an entire system is calculated by concatenating the individual weld failure rates and rupture frequencies. As a final step, Markov model theory is used to evaluate the influence of alternate strategies for in-service inspection and leak detection on the frequency of leaks and ruptures, and to calculate age-dependent LOCA frequencies. See Appendix D for a more detailed description of this analysis (Block 10).

3.5.1.2.2 Bill Galyean's Base Case Estimates - This analysis also relied primarily on available service history data to develop the base case frequencies, but in contrast to the previous method used a "top-down" approach. The approach (see Appendix E) starts with the straightforward calculation of a best estimate LOCA frequency using the number of failures (i.e., LOCAs) divided by the total number of reactor operating years. The U.S. nuclear power service history consists of zero Category 1 (i.e., greater than 100 gpm [380 lpm]) LOCAs over the entire operating experience, which totals 2,647 LWR (calendar) years through April 2003. This operating history data was used to update a non-informative prior distribution in a Bayesian calculation, to produce a posterior probability distribution on the total LOCA frequency. The overall LOCA frequency calculated in this manner is approximately 1.9E-4/LWR-year (0.5/2600 LWR-years). This total frequency is then partitioned using the relative ratios of crack and leak events from the available service data, to pipe and non-pipe passive component contributor categories. Then, within the piping and non-piping categories, the frequency is further partitioned using the available crack and leak data into the base case piping systems and degradation mechanisms. The partitioned frequencies representative of the base case conditions are used as the base case frequencies for LOCA Category 1. The frequencies for the other LOCA categories were calculated using the common assumption that the frequency of a LOCA decreases as pipe size increases. A somewhat arbitrary scaling factor of ½ orders of magnitude was used to scale successive LOCA categories by assuming a lognormal probability distribution on LOCA frequency. This assumption appears to be reasonably consistent with historical LOCA frequency estimates. See Appendix E for a more detailed description of this analysis (Block 10).

3.5.1.2.3 Dave Harris's Base Case Estimates - This analysis was one of two base case estimates using primarily probabilistic fracture mechanics (PFM) to establish base case frequencies. This approach utilized the PRAISE code for all calculations [3.10]. Separate analyses were conducted for the five different piping base cases and a number of sensitivity calculations were also performed. Individual welds were analyzed and then converted into piping system related frequencies. Benchmarking of service history data for through-wall flaws was conducted where possible. Included below is a synopsis of conditions modeled and the analyses conducted for each base case.

The PWR hot leg (PWR-1) base case was modeled to separately consider fatigue crack growth of pre-existing defects and PWSCC initiation and growth. Stresses from an earlier PRAISE study were used. The case of PWSCC growth from pre-existing defects was selected as the base case. Sensitivity studies were made of the effects of hydrostatic testing, seismic events, and degraded material properties. The PWR surge line (PWR-2) base case considered initiation and growth of fatigue cracks. Stresses provided in NUREG/CR-6674 [3.11] were used although all seismic loads were removed. To obtain results for leaks greater than 100 gpm [380 lpm], an alternative procedure was required that involved extrapolating the lengths of through-wall cracks obtained from a PRAISE run to determine the failure probability given the existence of a through-wall crack. The HPI/make-up nozzle (PWR-3) was modeled with a failed thermal sleeve. Stresses for an intact thermal sleeve from NUREG/CR-6674 [3.11] were employed, but a fatigue crack was considered to initiate immediately with a depth of 0.12 inches (3.0 mm). This depth corresponds to the initiation depth of applicable fatigue crack initiation relationships used within PRAISE.

Two piping sizes were considered for the BWR recirculation system (BWR-1). The smaller lines (12 inch) are more prone to fail due to IGSCC than the larger lines (28 inch), but the larger lines are the only contributor to the larger LOCA categories. Initiation and growth of IGSCC cracks was modeled. Default residual stresses were employed, except for the 12 inch line which was repaired with a weld overlay after 20 years of service. The weld overlay greatly improves the residual stress situation, and employs a material that is less prone to cracking. The application of the overlay was found to provide a large reduction in the failure probability. Applied stresses in the 12 inch line were varied in order to provide agreement with field observations of leaks, with a mean normal operating stress of 12 ksi (83 MPa) providing the best agreement. This stress also provided good agreement with field observations of part-through cracks. While the 12 ksi (83 MPa) mean normal operating stress results in better agreement with field observations of cracking, a 20 ksi (138 MPa) stress value was used for the BWR-1 base case estimates. The 20 ksi (138 MPa) value was chosen for the base case estimates to be consistent with the stresses utilized in the other base case analyses. This choice is somewhat arbitrary however, because the base case frequencies using either 12 ksi (83 MPa) or 20 ksi (138 MPa) normal stresses are similar. This consistency occurs because although the per joint failure frequency is higher for the 20 ksi (138 MPa) stress level, the number of joints that are subject to this higher stress level is much lower than the number of joints that are subject to the 12 ksi (83 MPa) stress level. See Appendix F for more discussion of this phenomenon.

The feedwater system (BWR-2) was modeled considering fatigue crack initiation and growth. Flow accelerated corrosion (FAC) is an important LOCA mechanism, but a probabilistic model is not available for this mechanism. System stresses were again obtained from NUREG/CR-6674 [3.11]. As for the surge line analysis, in order to obtain results for leaks greater than 100 gpm, an alternative procedure was required that involved extrapolating the lengths of through-wall cracks obtained from a PRAISE run for leaks (any through-wall crack). See Appendix F for a more detailed description of this base case analysis (Block 10).

3.5.1.2.4 Vic Chapman's Base Case Estimates - This analysis also relied primarily on a probabilistic fracture mechanics (PFM) approach for establishing the base case frequencies, but using the RR PRODIGAL code [3.12]. RR PRODIGAL is a basic fatigue failure probability model developed by Rolls Royce (RR) for the British Naval Nuclear program. The PRODIGAL code first simulates the weld construction in order to determine the defect distribution and density for both buried and surface breaking defects at the onset of service life. A failure probability using standard linear elastic fracture mechanics methods is then determined for both the buried and surface breaking defects. Failure is achieved when the defect either exceeds the R6 failure criteria [3.13], or simply grows through to the full thickness of the weld. The failure probability for all initial defects is then combined to form the total failure probability. The specific procedures used to develop the base case frequencies are as follows:

1 Evaluate the basic fatigue failure probability using the RR-PRODIGAL code using the transient data supplied.
2 Evaluate an elastic crack opening displacement (COD) as a function of defect size.
3 Use expert judgment to extend this COD beyond the elastic limit.
4 Evaluate the mean defect cross-sectional area for a given defect size using its associated COD.

5 Evaluate the mean leak rate for a given defect size using correlations developed for the elicitation..
6 Use expert judgment to assess the defect length distribution at failure.
7 Multiply Steps 5 and 6 to obtain the conditional probability of a leak rate greater than the prescribed leak rates for LOCA Categories 1 through 6.
8 Multiply the conditional probability of Step 7 with the basic fatigue failure probability in Step 1 to arrive at the required final probability of a leak greater than the prescribed leak rate for each of the leak categories.

Note, for steps 2, 3, 4 and 5 above, the probabilities for the mean leak rate are determined as a function of assumed defect size for an assumed leak. The defect length probability is multiplied by this to obtain the conditional probability of leakage as a function of LOCA category assuming that a leak exists. The RR-PRODIGAL code supplies the leak frequency. Individual welds were analyzed and then converted into piping system related frequencies using this method. See Appendix G for a more detailed description of this base case analysis (Block 10).

3.5.1.3 Panel Review of Base Case Analysis - The base case team members made separate presentations to the panel at the second elicitation meeting. The presentations consisted of the assumptions, approach, results, and sensitivity analyses utilized by each team member for each base case. The panelists provided feedback and a discussion of the differences among the four chosen methodologies and the subsequent effect on the differences in the base case frequency estimates (Block 9). Refinements to the base case calculations were suggested by the panelists to possibly minimize differences in the initial base case frequency estimates and ensure that the defined base case conditions were appropriately modeled. Refinements included conducting additional calculations to compare service history and PFM predictions of leaking crack frequencies at 25 years of operating life for each base case. Bengt Lydell was selected to estimate leak rates from service history data because of his access and knowledge of the most extensive piping precursor database. David Harris was selected to calculate leak frequencies using the PRAISE PFM code. It was suggested that additional benchmarking could be performed based on this comparison. Revised stress histories for several base case systems were also developed. Crack initiation and growth rate information for PWSCC cracking was also reevaluated for the appropriate base cases. All refinement calculations were subsequently completed after this meeting and provided to the panelists for use in the elicitation.

The elicitation panelists also identified additional sensitivity analyses that they thought would provide useful information for subsequent elicitations. These analyses included an examination of the effect of degraded material properties on base case frequencies and a comparison of crack growth rates with and without a weld overlay for the recirculation system. The results of the sensitivity analyses were also made available to the panelists. Additional details on these refinement and sensitivity calculations specified by the panelists are contained in meeting minutes (Appendix B). More details on the base case team refinement and sensitivity calculations are contained in the calculation descriptions provided by each base case team member (Appendices D – G). The base case frequency results after all refinements were completed are summarized in Section 4.

3.5.2 Non-Piping Base Cases

Non-piping base cases were similarly defined to provide frequencies associated with well-defined failure conditions and for anchoring to estimate the LOCA frequencies of other, similar components (Block 6). This definition and use is philosophically identical to the use of the piping base cases. However, the non-piping base cases were developed in a fundamentally different manner that was necessarily less comprehensive. The failure scenarios, design, and operation of LOCA-sensitive non-piping components are quite disparate compared to the piping systems. Possible non-piping failure scenarios include steam generator tube rupture and control rod drive nozzle ejection. These share many of the same attributes as postulated piping failures. However, non-piping failure scenarios also include RPV rupture, valve and pump body failures, and pump flywheel failures. Manway and vessel head failures resulting from common cause bolting failures are also possible. These failure scenarios are fundamentally different not only from piping failures, but also from each other. Therefore, it was not pragmatic to develop base case frequencies for anchoring each unique type of failure scenario. Further, generic service history databases for non-piping LOCA precursor events did not exist so there was no basis for making or benchmarking most non-piping base case failure predictions.

3.5.2.1 Non-piping Precursor Database - Therefore, the first step in the non-piping base case development was to create a separate precursor database for LOCA-sensitive non-piping components. This database was based primarily on licensee event reports (LER) contained in the sequence coding and search system (SCSS) database. The non-piping database identified all LOCA precursor events (e.g., cracks or leaks) between 1990 and 2002. These events were catalogued by the relevant non-piping component (i.e. valve, pump, RPV, pressurizer, or steam generator) and subcomponent (e.g. body, bonnet, nozzle, etc. for valves) identified by the panel during issue formulation (Section 3.4 and Appendix B). The aging mechanism and calendar year associated with each event was identified in the database as was the type of precursor event. Relevant precursor events include leaks, surface cracks, through-wall-cracks, and bolted connection failures. This database of 216 events was provided to the panel for use during the elicitation analogously to the database of precursor piping events. More information about the non-piping LOCA precursor database development and its contents is contained in Appendix H.

3.5.2.2 Steam Generator Rupture Estimates - The historical steam generator tube rupture frequency was defined as one non-piping base case. Only ruptures resulting in flow rates greater than 100 gpm (380 lpm) were considered. This distinction is consistent with the Category 1 LOCA definition in this elicitation. Ruptures of this size provide the most definitive data on passive system failures in smaller components and represent the majority of pressure-retaining boundary passive system failures that have resulted in flow rates greater than 100 gpm (380 lpm). The non-piping precursor database was used to estimate this frequency. Additionally, steam generator leakage incidents, which are much more prevalent than ruptures, were examined as a function of time to illustrate possible trends. Results of this analysis are contained in Section 4.

3.5.2.3 Non-Piping Base Cases Analyzed using PFM - A number of other specific failure scenarios were also identified as being particularly important. These were analyzed using PFM to provide additional anchoring frequencies. These included ejection of control rod drive mechanism (CRDM) nozzle tubes, evaluation of low temperature over pressurization (LTOP) events in BWRs, and examination of non-LOCA transient contributions to pressurized thermal shock (PTS) events in PWRs. Only the non-LOCA PTS transients were considered for anchoring because the LOCA transient frequencies are based on the assumed LOCA initiating event frequencies which are being reevaluated herein.

One of the panelists, Dr. Pete Riccardella, analyzed the LOCA frequencies associated with BWR vessel rupture and CRDM penetration ejection. The BWR analysis considered pressure vessel failures due to both manufacturing flaws within the welds and service-induced flaws from SCC that grow into the low alloy steel. An NRC-approved PFM code (VIPER) was utilized [3.14]. Irradiation embrittlement was modeled and the vessel leak and fracture frequencies associated with both normal operation and LTOP loading were calculated. The most important risk contribution from the calculation is an LTOP event (1,150 psi [7.9 MPa] at 88°F [31°C]) which has an assumed event frequency of 1E-3. The current analysis builds on earlier research [3.15] which was used as technical justification for reducing inspection requirements for BWR reactor vessels. More details on the analysis procedure are contained in Appendix I.

The study of CRDM ejection frequencies utilized a similar PFM program (MRPER) developed under a materials reliability program to evaluate CRDM cracking [3.16]. The analysis predicts crack initiation using a Weibull analysis based on observed leaking or cracked nozzles statistics through the spring of 2003. Probabilistic crack growth rates are based on laboratory PWSCC measurements and analysis was conducted for four characteristic plant types (i.e., B&W, CE, Westinghouse 2-loop, and Westinghouse 4-loop). The effect of inspection type, interval, and probability of detection was also considered. More details on this analysis are contained in Appendix I.

The final non-piping base case built on parallel work being undertaken to risk-inform the PTS screening criteria in 10 CFR 50.61 [3.17]. This study is evaluating PWR vessel failure probabilities as a function of the level of irradiation embrittlement to develop screening criteria for minimizing the PTS risk. Four PWR plants were investigated: Palisades, Beaver Valley, Calvert Cliffs, and Oconee. These plants span the plant vendors and design variables, and also have unique embrittlement characteristics. For each plant, probabilistic risk assessment, including a consideration of human errors, was used to determine the most likely event scenarios that result in safety system actuation. Structural integrity assessment has revealed that both high and low pressure safety system injection can induce high thermal stresses into the vessel. Therefore, the plant-specific thermal-hydraulic transient associated with each safety system actuation scenario was considered.

Vessel failure probability is predicted by first considering the probability of crack initiation from weld and baseplate flaws which are created during vessel manufacturing. The likelihood of crack arrest prior to the crack broaching the vessel thickness is then considered. Finally, the vessel failure frequency is calculated as a function of the level of material embrittlement. The final PTS screening criteria will be chosen such that the vessel failure probability is appropriately low. More details on the PTS reevaluation are available in [3.17].

The PTS challenges are appropriate to consider in the expert elicitation because they could lead to vessel failure and a resulting large LOCA. However, some care is required in this assessment. The biggest PTS risk contribution is usually from SB, MB, and LB LOCA initiating events, especially as material embrittlement increases. Further, the PTS risk is proportional to the LOCA frequencies. Because of this dependency, the total PTS risk cannot be used as a base case for determining the frequency of other pressure vessel or non-piping failures. However, there are other loading transients which contribute to PTS risk. These transients include stuck open primary and secondary valves and feed and bleed operations. The stuck-open valve and feed and bleed transient contribution to PTS risk ranges from 1% to 80% depending on the plant type and the embrittlement level. The panelists were only asked to consider the frequency associated with PTS risk from these transients for possible anchoring during the elicitation process.

3.6 Elicitation Background Information

The elicitation panel identified certain other generic information necessary for the elicitation. Additionally, individual panel members requested specific information to supplement this generic information. All information generated by these requests was catalogued and made available to all panel members to aid in developing their elicitation responses. In order to facilitate communications among the panel members, a repository of this information was maintained by Engineering Mechanics Corporation of Columbus (Emc^2) on an ftp site. Background information stored on this site includes references on relevant prior studies, a complete set of BWR and PWR piping isometric drawings, weld census information for the different piping systems, information on service stresses and transients, the piping and non-piping databases, and information pertaining to the piping and non-piping base case development. Additionally, all presentations, meeting minutes, elicitation questions, action items, and response tables were included. A complete listing of the ftp site contents is also maintained at the site. This information will be saved and catalogued after the elicitation has been completed.

3.7 Flow Rate Correlation

LOCA size categories were defined in the elicitation as a function of flow rate of the makeup water supply (Table 3.2).. This definition is historically consistent and is based on plant response and mitigation action requirements that are a function of the flow rate as well as the break location. However, the panel expertise was concentrated in the areas of structural analysis, materials, and fracture mechanics, not thermal-hydraulics. Therefore, it was more natural for the panelists to develop their responses as a function of the effective break area. In order to facilitate this evaluation, it was necessary to develop a correlation between the makeup flow rate and effective break area. The LOCA size categories were defined as cumulative frequencies at a given flow rate; these flow rates were then converted to an equivalent pipe diameter. The LOCA frequencies associated with each LOCA size category relates to the cumulative frequency of a single-ended break of the cited size, and all larger breaks (including double-ended breaks) of that size and larger.

Three separate correlations were developed for PWR liquid, BWR liquid, and BWR steam lines. The solutions utilized are a function of reactor type and the pipe length/diameter (L/D) ratio. A large L/D ratio implies saturation at the break plane. At small L/D ratios, the fluid does not reach these conditions. The leakage rate is also assumed to correspond to the flow rate required for the makeup water supply to mitigate the postulated break. Makeup water supply is assumed to exist at 70°F (21°C) under atmospheric pressure.

For the BWR steam lines, the correlations are assumed to be independent of pipe size and are modeled using equations by Todreas and Kazimi [3.18] using the upstream value for specific heat. The BWR liquid correlations were developed by assuming that the upstream pressure is approximately the steady-state operating pressure. For small L/D ratios (> 9" diameter), the Moody correlation was employed [3.19]. For large L/D ratios (< 9" diameter), the Zaloudek correlation was used [3.20]. The PWR liquid correlations were developed under the assumption that the upstream pressure is saturated at a temperature of 600°F (315°C). Once again, the Moody correlation was used for small L/D (> 8" diameter) while the Zaloudek was employed for large L/D (< 6" diameter) correlations.

These correlations are summarized in Table 3.8.

Table 3.8 Break Size to Leak Rate Correlation

LOCA Category	Flow Rate (gpm)	BWR: Steam		BWR: Liquid		PWR: Liquid	
		Flow Rate Flux (gpm/in^2)	Eff. Break Size (in)	Flow Rate Flux (gpm/in^2)	Eff. Break Size (in)	Flow Rate Flux (gpm/in^2)	Eff. Break Size (in)
1	100	355	1/2	595	1/2	687	1/2
2	1500	355	2 1/4	595	1 3/4	687	1 1/2
3	5000	355	4 1/4	595	3 1/4	687	3
4	25,000	355	9 1/2	595	7 1/4	687	6 3/4
5	100,000	355	19	375	18 1/2	641	14
6	500,000	355	42 1/4	375	41 1/4	641	31 1/2

It should be noted that these correlations are different from those used in NUREG/CR-5750 [3.5] and in other past LOCA efforts. The earlier correlations date to the NUREG-1150 [3.1] plant risk study. The BWR correlations used in NUREG-1150 were developed from a matrix of calculations conducted using existing thermal-hydraulic codes [3.21] specifically developed for each plant studied in NUREG-1150. A variety of break areas were postulated in certain systems and different combinations of mitigation equipment were assumed to be operational. Calculations were performed for each plant. The reported correlation of break area to flow rate is an amalgam of these results. It is assumed that the PWR correlations are similarly based, although their development is not as well-documented.

These prior correlations were not adopted for this exercise because there was concern about their generic applicability given the plant specific nature of the calculations performed. There is also concern about the accuracy of the thermal-hydraulic codes existing at the time of the NUREG-1150 study. Discharge leak rates are highly uncertain. They are difficult to develop and are a function of upstream conditions, break location, plant configuration, and mitigation reliability, as well as break area. Based on these considerations, the simple closed-form approximate solutions are sufficient for the generic correlations required for this elicitation. Application of the elicitation LOCA frequency results may require plant specific calculations to evaluate break flow rate history and required mitigation response for assumed break locations and sizes.

3.8 Elicitation Question Development

The elicitation questions were developed in concert with the panel members (Block 8). One objective was to develop questions that were precise enough to be unambiguous, yet general enough so that the question set could be minimized. Additionally, careful language was chosen to avoid terminology specific to only a subset of the technical specialties represented by the panelists. Another objective was to make sure that a variety of approaches could be used to develop answers to the questions. Initial questions were formulated after the kick-off meeting. These questions were modified based on panel member feedback prior to the base case review meeting. Modifications continued after the base case review meeting and up through the first several individual elicitations. Elicitation questions were posed in the following areas: piping base case evaluation, safety culture, PWR piping, BWR piping, PWR non-piping, and BWR non-piping.

3.8.1 Elicitation Question Philosophy

None of the quantitative questions required the panelists to assess absolute LOCA frequencies. The premise behind this philosophy is that an assessment of the relative likelihoods for comparable LOCA frequency contributors is a more natural reflection of knowledge and experience than is an assessment of the absolute frequencies of each contributor. A relative assessment allows a direct comparison of the effects on LOCA frequencies resulting from specific combinations of materials, loading characteristics, aging mechanisms, and mitigation procedures for all the LOCA-sensitive components. Assessments of these interactions and comparisons of these relationships best match the panel expertise. An assessment of absolute LOCA frequencies would not only require consideration of these relationships, but would also require estimation of the frequencies associated with each postulated set of conditions. However, all of the conditions rarely occur, and there is little supporting data for the members to assess the absolute frequencies. Consequently, assessing relative frequencies should lead to more accurate results. The elicitation questions have been structured accordingly, to require relative rather than absolute assessments. More detail on the elicitation questions pertaining to the base case, safety culture, piping, and non-piping evaluations follows.

3.8.2 Piping Base Case Elicitation Questions

The piping base case evaluation questions required each panel member to address the accuracy and applicability of the four base case calculations. First, each panelist was required to assess how well each calculation modeled the base case conditions defined by the panel (Table 3.7). Each panel member was then asked to comment on the differences among the four base case team members' results for each base case, and assess how modeling differences and variability in model input information led to these differences. Additionally, each panelist was asked to evaluate the reasonableness of the differences in the base case results and determine if the differences provide an accurate measure of the uncertainty inherent in LOCA estimates. Each panel member was then asked to assess the accuracy of each base case calculation and to choose, if desired, a specific set of results for anchoring his or her future elicitation responses. The actual elicitation base case evaluation questions are contained in Appendix J (Block 8).

3.8.3 Safety Culture Elicitation Questions

The first set of quantitative questions focused on the influence of safety culture. As detailed in the issue formulation section (Section 3.4), the panelists decided to evaluate the effects of safety culture separately from a consideration of other variables which affect the LOCA frequencies (e.g., piping system, material, degradation mechanism, etc.). This is justified because safety culture effects are judged by the panel to be independent of these other variables, and are strictly a function of general regulatory, general nuclear industry, and plant specific attitudes and practices.

This question asked each panelist to assess future safety culture with respect to the current safety culture. The future time period was decomposed into two intervals: the next 15 years (25 to 40 years), and the following 20 years (40 to 60 years). The current safety culture always serves as the baseline and two ratios were required, one for each time period, to predict relative future safety culture changes with respect to this baseline. The questions asked panelists to first separately consider the effects of utility and regulatory safety culture. Then, the panelists were asked to assess the degree of correlation between the utility and regulatory safety cultures. The questions also required the panelists to quantify these effects as a function of the LOCA size, if a relationship exists. The actual safety culture evaluation questions are contained in Appendix J.

3.8.4 Piping and Non-piping Elicitation Questions

Two parallel question sets were developed to determine piping and non-piping LOCA frequency contributions due to passive system failure. The question sets were structured to support either a bottom-up or a top-down philosophy (Figure 3.2). The bottom-up approach requires a fundamental consideration of explicit combinations of variables (i.e., loading, geometry, materials, degradation mechanisms, and mitigation), which lead to LOCAs in relevant piping systems and non-piping components (Figure 3.2). The bottom-up approach is consistent with historical PFM analysis of LOCA challenges as well as the Barsebäck-1 analysis [3.22].

The top-down approach allowed a more global consideration of piping system and non-piping component contributions without explicit consideration of individual degradation mechanisms, materials, etc. This approach is more consistent with a LOCA frequency development using classical service history analysis as in NUREG/CR-5750 [3.5]. These two alternative structures were developed to support approaches commensurate with the specific technical expertise and philosophy of individual panelists. Panelists were free to choose the top-down or bottom-up approach, a combination of the approaches, or they could develop their own methodology. Panelists were encouraged to attempt both the top-down and bottom-up question sets to evaluate the consistency of their responses. A few panelists did answer both question sets and were able to iterate their final responses so that both the top-down and bottom-up quantitative results matched their qualitative expectations. Also, several panel members did pursue alternative approaches that they developed. A summary of the various philosophies and approaches from each panelist is provided in Appendix K.

The piping and non-piping top-down and bottom-up questions were identically structured. This commonality allowed the panelists to utilize similar approaches for addressing passive systems using either approach. In each question set, the assessment structure is consistent with the relative assessment philosophy described previously. The panel member first is asked to identify important LOCA-contributing factors. Then, he is asked to select an appropriate base or reference case for comparison with each LOCA-contributing factor. He is then asked to make a relative assessment between the important contributing factor and the chosen base or reference case conditions for each LOCA category and operating time period (25, 40, or 60 years). The assessment continued for all important LOCA-contributing factors. These contributions of each individual factor are then analytically combined to develop the final LOCA frequency estimates for each panelist (Section 5).

Reference cases (Section 3.4.7 and Appendix B) were developed by the panel analogously to the base cases for the purpose of making relative assessments within piping systems that do not have an associated base case. The reference cases, like the base cases, represent a specific set of defined conditions for each important LOCA variable class (Section 3.4.7). Base case conditions have been previously summarized in Table 3.7. If a panelist chose to use the reference cases, he was required to first make a relative assessment between the base case and reference case conditions. This is necessary because only the base case conditions are associated with fundamental, absolute frequencies. There was no explicit calculation of the frequencies associated with the reference cases. Once this assessment was made, the reference case associated with a piping system could be used for assessing issues within that piping system.

3.8.4.1 Top-Down Questions: Specific Considerations - This structure of issue decomposition, selection of important issues, relative comparison with base or reference cases, and final combination of contributions allows the absolute LOCA frequencies to be developed for each panelist starting from only the service history and/or PFM based estimates of LOCA frequencies associated with the simplified base case conditions. No other absolute LOCA frequency quantification was required.

The only difference between the top-down and bottom-up question sets discussed previously is the decomposition and consideration of the important contributing factors. The top-down approach asked the panelist to identify the most important piping systems or non-piping components. The relative comparison is made between the chosen base case condition and the important piping system/non-piping component. This analysis is continued for all the piping systems and non-piping components.

The following example illustrates the top-down approach. A panel member first lists the important piping systems (or non-piping components) for a PWR plant. He chooses the instrument lines, drain lines, CVCS, hot leg, cold leg, and RHR system and has appropriate rationale supporting this selection. Then, for the RHR system, he chooses a base case. In this example, he picks the hot leg cracking due to PWSCC because he feels that this mechanism is the most likely to cause a LOCA in the RHR. He then compares the LOCA contributions of the RHR to PWSCC cracking in the hot leg as a function of LOCA size. He then determines the following ratios for LOCA Categories 1 – 6, respectively: 100, 50, 20, 10, 5, and 0. These estimates are based on the panelist's determination that PWSCC cracking in the RHR is 100 times more likely to result in a Category 1 LOCA than PWSCC hot leg cracking. However, Category 5 LOCAs are only expected to be five times more likely. The RHR system cannot support a Category 6 LOCA which is why this ratio is zero. The panelist determined these ratios by assessing precursor differences between the RHR and hot leg and the knowledge that most of the precursors were due to mechanical fatigue which would tend to cause smaller LOCAs instead of complete piping failure. The panelist also considered the loading environment, and number of welds in the RHR system compared with the hot leg in making this assessment. This panelist would conduct a similar analysis for all the important PWR piping systems listed above.

3.8.4.2 Bottom-up Questions: Specific Considerations - The bottom-up approach asked the panelist to identify important combinations of piping materials, degradation mechanisms, loading, geometry, and mitigation measures within a given piping system or non-piping component. The relative comparison is then made between the base or reference case condition and the selected variable combination. This assessment continues for all important variable combinations for all relevant piping systems or non-piping components.

The following example is applicable to the bottom-up approach. A panelist is assessing BWR LOCA contributions. This panelist believes that IGSCC cracking under typical hydrogenated water chemistry and inspection procedures provides the greatest LOCA contribution. He chooses the recirculation system BWR base case for comparison. This base case evaluated IGSCC LOCA frequencies in normal water chemistry with a weld overlay applied after 20 years of operation. Based on laboratory studies, the panelist believes that crack growth rate of existing IGSCC cracks is similarly retarded using either a weld overlay or hydrogenated water chemistry. However, the weld overlay also contributes an additional margin of two to the failure stress for typical IGSCC crack sizes. In the recirculation system, the lower applied failure stress transient is 10 times more likely than the transient require to fail a joint with a weld overlay. Therefore, the panelist determines that the LOCA frequencies associated with the hydrogenated water chemistries without an overlay are a factor of 10 higher than the base case LOCA frequencies in the recirculation system for all LOCA sizes. The panelist then repeats this assessment for the IGSCC

LOCA susceptibility of other piping systems and then similarly considers LOCA contributions from other important degradation mechanisms.

3.8.4.3 Additional Non-piping Considerations - As mentioned, the non-piping questions were structured analogously to the piping questions. However, an additional level of decomposition was necessary for the non-piping considerations. For each non-piping component (i.e., vessel, steam generator, pump, valves, and pressurizer), LOCA-sensitive subcomponents (e.g., tubes, manway, tube sheet, shell for the steam generator) were identified. This subcomponent decomposition is analogous to the piping decomposition by system and was necessary due to the different types of failures associated with each non-piping subcomponent. As with the piping systems, the subcomponents have specific combinations of materials, geometry, degradation mechanisms, loading, and mitigation which affect the LOCA likelihood. The subcomponent decomposition is summarized in Section 3.4.6.

The only other difference between the piping and non-piping questions is that specific failure scenarios were explicitly developed for various non-piping subcomponent variable combinations. The failure scenarios represent a combination of factors that could lead to non-piping LOCA failures. This distinction was necessary because the non-piping failure scenarios were more disparate than the piping considerations and it was not efficient to decompose those failures into single sets of appropriate variable classes as was done for the piping systems. An example is the improper torque application of the reactor head bolts due to human error that is not discovered during either bolt tension testing or initial system pressurization. Another failure scenario could be CRDM nozzle ejection due to PWSCC cracking which is not found during planned inspections. Many different failure scenarios were developed by the panelist during the issue formulation (Section 3.4.6 and Appendix B). Each panelist was also free to develop and evaluate other relevant scenarios.

Other than the additional decomposition and the distinction between the terminologies of variable combinations (piping) and failure scenarios (non-piping), the piping and non-piping questions are identically structured. As previously discussed, the top down non-piping approach focused on assessing those important failure scenarios for relevant non-piping subcomponents regardless of the non-piping component type. The bottom-up approach required a more detailed assessment of possible failure scenarios in each non-piping component and subcomponent.

3.8.5 Elicitation Response Requirements

All elicitation questions, except the base case evaluation, required both a qualitative and quantitative assessment. For each quantitative assessment, the panelists were requested to provide a mid-value, a lower bound, and an upper bound. As in the training exercise (see Section 3.3.2), the mid-value is defined such that, in the panel member's opinion, the unknown true value for that particular question has a 50% chance of falling above or below the mid-value. The lower bound is defined such that the true value has a 5% chance of falling below the bound. The upper bound is similarly defined such that the true value has a 5% chance of being above the bound. The mid-value, lower bound, and upper bound were interpreted in subsequent analysis (Section 5) as the median, 5^{th} and 95^{th} percentiles, respectively, of a subjective distribution for the true value of the answer.

There were examples created for each elicitation question to clarify the question requirements for the panelists and to illustrate possible approaches that could be used to answer the questions.

3.9 Individual Elicitations

Each panel member had between one and four months to prepare their elicitation responses after the base case review meeting was held and the elicitation questions were finalized. During this time period, individual elicitation sessions were conducted separately between each panel member and the facilitation team (Block 11). With the exception of the last elicitation session with Mr. Helmut Schulz of GRS in Germany, each of the sessions took place at the NRC headquarters in Rockville, MD. Mr. Schulz's elicitation was conducted via video teleconference (VTC). Each session lasted a full day.

The elicitation sessions addressed each elicitation question in order; starting with the base case evaluation questions, then the safety culture questions, followed up by the piping and non-piping related questions. The objectives for the individual elicitation sessions were to:

1. Obtain and discuss the quantitative and qualitative responses to the elicitation questions.
2. Identify any inconsistencies between the quantitative and qualitative responses.
3. Provide additional clarification to the elicitation questions, as necessary.
4. Identify necessary follow-on work for each panel member.
5. Solicit feedback about the process.

The primary objective was to obtain the quantitative responses to each question and understand the qualitative rationale used as the basis for these responses (Block 12). Each panel member used a different approach to obtain quantitative estimates (Appendix L) and it was important that the facilitation team fully understood each participant's approach so that their results could be subsequently analyzed. Most of the panelists provided the facilitation team with written responses to the elicitation questions in advance of the interview. These responses were typically the starting point for the review and discussion held during the interview. The facilitation team's role was to evaluate the responses to ensure that they reflected the panel members' judgments and were consistent with the elicitation question requirements. While preparing their elicitation question responses, the panel members were asked to self-assess and not answer questions in areas where they had little or no expertise. In areas where they had limited technical expertise, the panel members were urged to make estimates, but to reflect their uncertainty in the upper and lower bounds.

The discussion and review of the panelist's responses quite often revealed inconsistencies between the quantitative estimates and the supporting qualitative beliefs. These inconsistencies were explored to understand their genesis and possible resolution. Additionally, other weaknesses or incomplete areas in the initial question responses were discussed. Sometimes, response deficiencies resulted from a lack of understanding of particular elicitation questions and their requirements. These remaining ambiguities were clarified during the interview.

Each elicitation was taped to provide a record of the elicitation. Minutes from each elicitation were also recorded. The audio tapes and minutes were also used to clarify information and opinions. Approaches were also developed with each panelist for subsequent follow-on work. A list of action items was also developed for both the panel members and the facilitation team to reflect the additional work necessary to resolve inconsistencies, complete missing information, and strengthen weak or ambiguous information provided in these initial elicitation responses. This action item list was used by each panelist to help refine his analysis. Each panel member then had another one to four months to revise his initial input to address the follow-on work identified during the elicitation.

3.10 Final Elicitation Responses

After the individual elicitations were completed, and each panel member's refined elicitation responses were analyzed (Block 13), a third meeting was held with the entire panel. The purpose of this meeting was to summarize and discuss the important quantitative and qualitative results arising from the individual elicitations. The qualitative insights provided for each of the elicitation questions was a focal point. These were discussed among the panelists so that all could share in this combined knowledge. Each panelist benefited by understanding insights that he may not have explicitly considered in the development of his elicitation responses. Additionally, each panelist gained from the insights in areas outside of his expertise (Block 14).

The quantitative LOCA frequencies associated with important piping systems and all the non-piping components were presented separately. The qualitative insights were also illustrated in the context of the quantitative results. Important degradation mechanisms, loading, and other rationale associated with the LOCA frequencies for each system were illustrated as underlying rationale. Also, the rationale associated with quantitative outliers was also presented in those situations where it was unique and had possibly not been considered by others. This discussion was used to highlight issues and concerns from individual panelists so that the entire panel could assess their merits. Quantitative comparisons among piping systems and non-piping components were also provided so that the panelists could understand the implications of responses that they developed for the decomposed technical issues. Quite often, panelists did not calculate total LOCA frequencies to understand these interdependencies.

Another objective of this meeting was to thoroughly explain the methodology used to combine all the quantitative piping and non-piping LOCA frequencies for the decomposed piping systems and non-piping subcomponents to develop comprehensive LOCA estimates from each panelist's elicitation responses. The facilitation team reviewed these individual calculations with each panelist so that each could understand how his initial responses had been used in this development. Sometimes, the raw input had to be converted to fit the uniform analysis (Section 5.1) and response (Section 5.2) frameworks developed for this calculation. Also, additional assumptions were sometimes necessary to fill in missing information in the panelists' responses. The facilitation team made assumptions or added missing information, whenever possible, in a manner consistent with the philosophy, approach, assumptions, qualitative insights, and other quantitative results already provided by the panelist. The most common reason for augmenting the results was to provide complete coverage intervals for every elicitation response. While panelists often did not provide every coverage interval, they typically did assess coverage intervals in other places in the questionnaire. If necessary, the facilitation team created the missing coverage intervals to be consistent with these other responses. Any imputed data was subject to panelist review and acceptance prior to final use. Additionally, panel members were also required to ensure that their responses were correctly transferred to the response framework (Section 5.2) and verify the results calculated from their input responses.

Another purpose of this meeting was to discuss the quantitative frequencies associated with the non-piping base cases. The qualitative conditions for all the non-piping base cases had been developed by the panel in previous meetings. Also, the quantitative calculations had been either directly provided to the panelists or made available on the ftp site for consideration. However, these quantitative calculations had yet to receive the same scrutiny as the piping base case results. The analysis methodology and results used to determine the steam generator tube rupture frequencies, the CRDM ejection frequencies, BWR vessel failures due to normal operation and

LTOP loading, and the PWR vessel PTS analysis were shared. There was also a presentation provided for the non-piping database development. The panel asked questions and provided feedback to gain a thorough understanding of the non-piping base case calculations and results similar to the piping base case evaluation in the previous meeting. Some concern was raised during this evaluation that not all panelists may have properly addressed common cause or conditional failures stemming from a single steam generator tube rupture or CRDM ejection. Subsequent elicitation questions were therefore developed to explicitly address this concern. More information on this meeting is contained in Appendix B.

A number of final action items were developed from this meeting. First, each panelist was asked to review the calculations that the facilitation team performed on his elicitation responses. Each panelist was to ensure that no errors were made in the entry of his responses, and was also asked to comment on the appropriateness of assumptions or missing information added by the facilitation team during the analysis of his results. Each panelist was asked to provide alternative assumptions or missing information if desired. The panelists were also asked to separately consider conditional and common cause failures for steam generator and CRDM ejections.

Finally, all the panelists were given one more opportunity to revise their elicitation responses based on the discussion of the qualitative insights and rationale, a more thorough understanding of the non-piping base case calculations, and a more complete understanding of the analysis procedures used by the facilitation team in processing their raw input. Most panelists did not choose to change their preliminary estimates. However, a few did modify their preliminary responses based on knowledge gained at this final meeting (Block 15). Any necessary iteration was again conducted with each panelist to ensure that the final responses are consistent and reflect the qualitative rationale. These final responses were then analyzed once again to determine the LOCA frequency estimates provided by each panelist (Block 16).

3.11 Review and Reporting

A multilayer review was conducted using both the panelists and external reviewers to validate the accuracy and acceptability of the reporting and the analysis of the panelists' responses. First, a draft report (Block 17) was created and circulated to the panelists for review. The panelists provided review and feedback (Block 19) on every report section in order to increase the clarity and understanding of the elicitation process, analysis, and results presentation. However, they conducted the most substantive technical review of the background (Section 1), approach (Section 3), base case results (Section 4), qualitative results (Section 6), and the summary and conclusions (Section 9). Several panelists provided additional background information (Section 1) to ensure that the service history and PFM descriptions were comprehensive and balanced. The piping and non-piping base case team members contributed summary descriptions of their approaches (Section 3.5) and ensured that the base case results and sensitivity analyses were accurately and clearly reported (Section 4). The piping base case team members also provided some rationale to explain differences among the results.

The panelist review of and contributions to the qualitative results section (Section 6) was vital to ensure that the results were summarized in a manner that reflects the important insights, yet also presents dissenting views when appropriate. Group consensus of these insights was not explicitly sought during the elicitation process; however, many of the reported insights were shared by the majority. The panelists also reviewed the summary and conclusions (Section 9) that pertain to the base case and qualitative results to ensure that they were complete and accurate. A revised report (Block 20) was developed which addressed the feedback received from the panelists. The panelists were given the opportunity to conduct a final review of this report and additional final comments have been incorporated as warranted.

In parallel with the panelists' review, an external peer review (Block 18) was conducted of the baseline elicitation response analysis (Section 5.5), the sensitivity analyses (Section 5.6), and the corresponding results (Section 7) contained in the draft report (Block 17). These are the portions of the process that the elicitation panelists are not qualified to review. Two peer reviewers were chosen: Dr. C. Atwood of Statwood Consulting and Dr. A. Brothers of Pacific Northwest National Laboratory. Dr. Atwood is a statistician with experience in estimating rare event frequencies for PRA use and is also an author of NUREG/CR-5750 [3.5]. Dr. Brothers is a decision analyst with knowledge of elicitation processes and procedures.

The draft report was initially provided to the peer reviewers so that they could read Sections 5 and 7. A 1 ½ day kick-off meeting was then held. The purpose of the kick-off meeting was to present and clarify the analysis procedures, sensitivity analysis matrix, and results in detail so that the reviewers thoroughly understood the approach. Some initial errors found in the analysis procedure were also discussed at the meeting. Ideas for conducting additional sensitivity analyses and alternative response analysis and aggregation schemes were also evaluated during the meeting. After the meeting, the facilitators were in contact with the reviewers to answer any additional questions. The peer reviewers provided some preliminary feedback within a week of the kick-off meeting and then provided their initial review a month after the kick-off meeting.

The peer reviewers provided several recommendations and insights which are detailed in References 3.23 and 3.24. Dr. Brothers stated that while this was not a focal point of his review, the elicitation process appears adequate and sound for determining the stated objectives. Dr. Atwood found some errors in the baseline analysis procedure (Section 5.5) and provided an exact formulation for means of summed distributions that were initially estimated. These errors were corrected and summation approximations eliminated wherever possible. The accuracy of remaining approximations has been evaluated using selected Monte Carlo analysis as requested in Reference 3.23. Both peer reviewers also suggested additional sensitivity analyses to verify other analysis assumptions and approximations. All important sensitivity analyses have been conducted and the calculational methodology has been updated as warranted to reflect these findings. The updated procedure is reflected in the analysis section (Section 5) of this revised report (Block 20). The elicitation responses were reevaluated using the updated analysis procedure (Block 16) to develop the final LOCA frequency estimates (Block 22).

One focal point of the peer review recommendations pertains to the aggregation of the individual LOCA frequency estimates (Section 5.4). Aggregation schemes other than the baseline methodology (Section 5.5) could be appropriate depending on the assumptions and interpretation of the individual total LOCA frequency estimates. The most reasonable alternate aggregation scheme creates a mixture distribution from the individual responses. This aggregation was conducted as an additional sensitivity analysis (Section 5.6) for comparison with the baseline results (Section 7.6) and is included in the revised report (Block 20).

The revised report (Block 20) will be made available for public comment (Block 21). All public comments will be addressed and the report will undergo a final revision as necessary. After the public comment period, the elicitation process will be completed and the LOCA frequency estimates contained in this report will be considered final (Block 22).

3.12 References

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4. BASE CASE RESULTS

As indicated earlier (Section 3.5), base case frequencies were developed by a subset of the panel. These frequencies were then provided to the other panelists as possible anchoring frequencies for use during their elicitations. Panelists had the option of using the base cases and associated frequencies or developing an alternative approach. Base case frequencies were developed for five piping systems, two BWR systems and three PWR systems. Base case frequencies were also developed for a number of non-piping failure scenarios including BWR vessel failure, CRDM ejection, steam generator tube rupture, and non-LOCA PTS transients. There were also two databases available containing precursor failure information for LOCA-sensitive piping systems and non-piping components as discussed in Section 3 that could be used for anchoring during the elicitations. The results of the associated base case analysis and sensitivity studies are provided in this section.

4.1 Piping Base Case Through-Wall Cracking Frequencies

Bengt Lydell (BL) and David Harris (DH) developed through-wall cracking frequencies associated with each base case up to 25 years of average plant life. The purpose of this exercise was to provide the most direct comparison between service history information and probabilistic fracture mechanics (PFM) results and provide information for benchmarking the PFM results. Bengt Lydell used a comprehensive service experience data base (Section 3.5.1.2.1) to obtain direct estimates of the through wall crack frequencies. David Harris (Section 3.5.1.2.3) determined frequencies using the PRAISE code. Some of the PFM results were adjusted, or benchmarked, through manipulation of variable input parameters (Appendix F) in order to more closely match the service history through-wall cracking frequencies.

Figure 4.1 is a comparative plot of the leak frequency results after 25 years of service. In this figure, the hot leg (PWR-1), surge line (PWR-2), HPI/MU line (PWR-3), recirculation system (BWR-1), and feedwater system (BWR-2) base cases are all represented. These through-wall cracking frequencies represent LOCA precursor events due to the material degradation mechanisms specified in the base case definitions (Section 3.5.1.1). The flow rate associated with these through wall cracks range from effectively zero up to the technical specification leak rate limit of approximately 1 gpm (3.8 lpm). Leaks bigger than 1 gpm (3.8 lpm) are assumed to either be detected by the leak detection systems or result in a LOCA.

There is relatively good agreement between the service history (Bengt Lydell) and PFM (David Harris) results for the PWR-3 and BWR-1 base cases. This is to be expected because the PFM results were effectively benchmarked against the service history estimates for both of these base cases. It is worth noting that IGSCC events in the BWR recirculation system (BWR-1) and thermal fatigue cracking in the HPI make-up nozzles (PWR-3) have relatively high service frequencies and there is correspondingly a larger body of through-wall cracking data associated with these cracking mechanisms. However, there is significant disagreement between the service history and PFM results for the other three base case results. The PWR-1 and PWR-2 base cases represent cracking in the hot leg due to PWSCC and in the surge line due to PWSCC and thermal fatigue, respectively. The PFM results are quite sensitive to initial input variable distributions for these base cases. Relatively minor changes can result in through-wall cracking frequency differences of several orders of magnitude (Appendix F). The through-wall cracking PFM frequencies could have been chosen to provide better agreement between the PWR-1 and 2 service history results. However, there is the least amount of service experience available, and consequently the lowest frequencies and highest uncertainty associated with the PWR-1 and 2 service history estimates. Therefore, the PFM through wall crack results used for benchmarking (Figure 4.1) were chosen to provide more realistic and comparable Category 1 LOCA estimates to the service history results (Section 3.5.1.2).

The most significant difference occurs for the feedwater system base case (BWR-2). The PFM estimate is approximately 6 orders of magnitude less than the service history estimate. Reasons for this difference are not readily apparent. The PFM result only includes fatigue frequency contributions and not flow accelerated corrosion (FAC) as required for the PWR-2 base case because there is no FAC degradation model in the PRAISE code. However, FAC is not a mechanism that typically results in through-wall cracking prior to failure. Therefore, it is not expected that these large differences are due to high FAC through-wall cracking frequencies in service. The service history data could be analyzed more closely in an attempt to resolve this difference, but this was not a principal concern since both estimates were provided for panel consideration. It should be stressed that one principal objective of the base case evaluations was to illustrate the variability which is possible using different estimation schemes. This variability was important for the panel to consider in their individual elicitations.

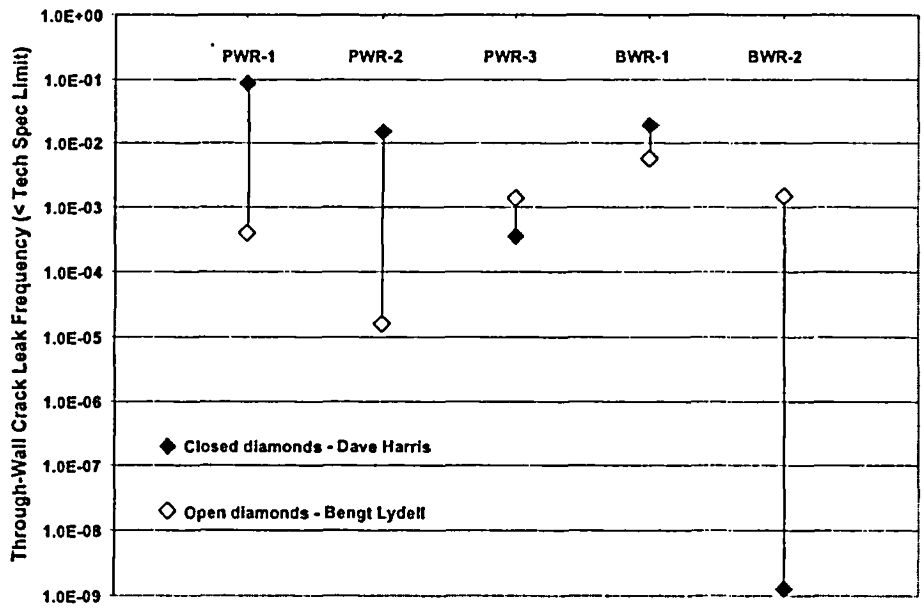

Figure 4.1 Leak Frequency Comparison Between Service History-Based and PFM-Based Analyses

4.2 Piping Base Case LOCA Frequencies

Piping base case LOCA frequencies were developed by four of the panel members: Bill Galyean (BG), Bengt Lydell (BL), Dave Harris (DH), and Vic Chapman (VC). As previously described in Section 3.5, base case frequencies were intended to be developed for three time intervals: 0-25 years (current day), 25-40 years (end of original design life), and 40-60 years (end of plant life extension). However, only BL and DH predicted base case frequencies beyond 25 years. Bill Galyean and VC provided estimates only for 25 years, although BG has indicated (Appendix E) that his base case frequencies are assumed to be largely constant with time. Frequencies were developed for all six LOCA categories for the hot leg base case (PWR-1). For the other piping systems, estimates were made up to the largest LOCA size amenable to each piping system. All four base case team members developed LOCA frequencies for the PWR piping base cases. All team members except VC developed frequencies for the BWR piping base cases. Two of the experts made their estimates by analyzing existing service history data (BG and BL) while two employed PFM analyses (DH and VC). Where possible, benchmarking of DH's results was performed between the calculated and service-estimated through-wall cracking frequencies (Section 4.1). The VC PFM results were not benchmarked. More detailed information on the approaches utilized by the four base case team members is provided in Section 3.5.1.2 and in Appendices D through G.

The piping base case frequencies (per calendar year) are summarized in Table 4.1. Note that the operating experience-based frequency estimates (BL and BG) were based on calendar years of plant operation where various reactor states from full power to shutdown occur. The PFM-based estimates (VC and DH) were based on reactor years of operation where it is assumed that the reactor is at full power over the entire year. The PFM estimates in Table 4.1, and subsequent tables, have been multiplied by a conversion factor of 0.8 to account for the approximate fleet average shut-down time per year [4.1] and to ensure that all frequencies are reported on a consistent per calendar year basis. This adjustment is a minor consideration in light of the uncertainty in these estimates, but is performed for consistency.

It should also be noted that these base case estimates (Table 4.1) are those that the base case team members believe are most accurate. These estimates evolved during the elicitation and were not finalized prior to the onset of the individual elicitations. Therefore, interim estimates were provided to the panel for anchoring during the individual elicitations. These interim estimates are not reported. The interim results of BL and DH are generally within a factor of 2 of the final estimates (Table 4.1) although some results did vary by a factor of 10 or more. The interim results for BG and VC, however, can vary by an order of magnitude from the final estimates. While most of differences are relatively insignificant in light of the overall uncertainty associated with the elicitation process, the individual elicitation responses were reanalyzed when necessary to account for the changes between the interim and final base case estimates.

The results in Table 4.1 are interesting because large differences exist among the results for the four base case team members. These differences are a reflection of modeling differences and not a reflection of actual plant operations. These uncertainties are explored further using Figures 4.2 and 4.3. These figures depict the piping base case frequencies (per calendar year) for Category 1 and 3 LOCAs, respectively, at 25 years of plant operations. The variability among the majority of the Category 1 LOCA estimates (Figure 4.2) is approximately one order of magnitude for all five piping base cases. However, the results for VC are quite distinct from the others for the PWR-1 and PWR-3 base cases while the results for DH are quite distinct from the other results for the BWR-1 base case.

The variability between the two service history-based analyses (i.e., the BG and BL results) is always relatively small, less than an order of magnitude for all five base case Category 1 estimates (Figure 4.2). This consistency is not surprising given the similarity of the approaches. BG and BL used different service history databases, but all the information in BG's database was contained within the more expansive database used by BL. Additionally, both BG and BL used similar conditional pipe rupture relationships (Appendices D and E).

Table 4.1 Piping Base Case Frequency Results by Participant

LOCA Cat.	25 Years Participant				60 Years Participant			
	BG	BL	DH	VC	BG	BL	DH	VC
	PWR-1 Hot Leg (per calendar year)							
1	1.5E-07	7.4E-07	3.2E-08	1.8E-11		9.7E-07	3.2E-08	
2	4.6E-08	7.6E-08	3.8E-11	1.8E-12		1.0E-07	3.8E-11	
3	1.5E-08	2.9E-08	1.1E-12	4.4E-13		3.8E-08	1.1E-12	
4	4.6E-09	1.1E-08	1.1E-12	7.0E-15		1.4E-08	1.1E-12	
5	1.5E-09	3.8E-09	8.8E-16	1.4E-15		4.9E-09	8.8E-16	
6		1.3E-09	8.8E-16			1.6E-09	8.8E-16	
	PWR-2 Surge Line (per calendar year)							
1	1.5E-08	1.3E-07	4.8E-07	1.9E-07		1.8E-07	1.4E-05	
2	4.5E-09	1.5E-08	2.2E-09	3.1E-08		2.1E-08	9.6E-08	
3	1.5E-09	5.4E-09	1.5E-10	1.2E-08		7.4E-09	1.4E-08	
4	4.5E-10	1.6E-09	6.3E-14	1.9E-09		2.1E-09	4.0E-11	
5		5.3E-10		3.9E-10		7.3E-10		
6								
	PWR-3 High Pressure Injection/Make-Up (HPI/MU) (per calendar year)							
1	2.3E-06	1.6E-05	6.2E-05	6.7E-12		2.0E-05	3.2E-04	
2	7.0E-07	2.3E-06	6.2E-05	7.4E-13		3.3E-06	3.2E-04	
3	2.3E-07	9.2E-07		2.8E-13		9.5E-07		
4								
5								
6								
	BWR-1 Recirculation System (per calendar year)							
1	5.3E-05	9.5E-06	9.2E-03			1.9E-05	5.7E-04	
2	1.6E-05	1.2E-06	6.8E-03			2.4E-06	5.0E-04	
3	5.3E-06	4.6E-07	4.8E-03			9.2E-07	5.0E-04	
4	1.6E-06	1.5E-07	3.1E-03			3.0E-07	5.0E-04	
5	5.3E-07	3.0E-08	2.1E-06			6.1E-08	3.0E-06	
6								
	BWR-2 Feedwater Lines (per calendar year)							
1	4.0E-06	2.5E-06	<1.0E-7			2.6E-06	1.0E-07	
2	1.2E-06	3.4E-07	<1.3E-11			3.4E-07	1.3E-11	
3	4.0E-07	1.2E-07	<6.1E-13			1.3E-07	6.1E-13	
4	1.2E-07	4.1E-08	<1.1E-17			4.1E-08	1.1E-17	
5		7.3E-09				7.4E-09		
6								

The variability between the two PFM-based (DH and VC) results was often substantial. For example, the PWR-1 and PWR-3 PFM-based results differ by 3 and 7 orders of magnitude, respectively (Figure 4.2). The DH results are much closer to the service-history estimates in both of these cases. These differences are largely attributable to modeling considerations. For the PWR-1 base case, VC considered only fatigue in his PFM analysis while DH considered both fatigue and PWSCC. For the large diameter pipe analyzed in the PWR-1 base case (i.e., the hot leg), PWSCC is a much more significant contributor to the failure frequency. This causes VC's PWR-1 Category 1 LOCA estimates to be substantially lower than DH's results. For the larger category LOCAs, i.e., Category 5, the difference is not that great. This is probably attributable to the fact that the distribution of initial surface crack lengths assumed by DH may have tended toward shorter surface crack lengths than did the distribution assumed by VC. The large, seven orders of magnitude difference between the VC and DH PFM predictions for PWR-3 (HPI/MU nozzle) can again be attributed to modeling considerations. Dave Harris considered accelerated initiation and elevated stress due to a failed nozzle thermal sleeve. This is typically the condition evident when service cracking has been apparent. The DH results are also benchmarked by the through-wall cracking frequency for PWR-3 (Figure 4.1). However, VC analyzed the main weld of the nozzle assuming that the thermal sleeve was intact. The intact sleeve significantly decreases the thermal transients, thus leading to the insignificant failure frequencies calculated by VC. David Harris conducted a sensitivity analysis by considering an intact thermal sleeve with a similar transient stress history to VC. The LOCA frequencies for these more similar conditions only vary by three orders of magnitude, instead of seven. For the PWR-2 base case, thermal fatigue (which was modeled by both DH and VC) is a much more important contributor and the two PFM estimates are much closer, differing by less than one order of magnitude for the Category 1 LOCAs. Neither DH nor VC modeled PWSCC for the PWR-2 base case.

The variability among the Category 3 LOCA base case estimates (Figure 4.3) when compared with the Category 1 LOCA estimates (Figure 4.2) would be expected to increase as the event frequency decreases. Small analysis variations and differences can result in greater relative disparity among the LOCA frequency estimates as the absolute frequencies decrease. However, for four of the five piping base cases, the LOCA Category 1 and 3 estimates exhibit similar variability. The biggest increase occurs for the BWR-2 base case. The variability between the base case frequencies for the Category 1 LOCA estimates is less than 2 orders of magnitude for this piping system while the variability among the Category 3 estimates is approximately 6 orders of magnitude. This additional variability results from the substantial LOCA Category 3 frequency decrease predicted by DH. The approach employed by DH predicts that any failures associated with the BWR-2 base case will likely lead to smaller LOCAs, while the conditional failure probability associated with a larger LOCA is much smaller than is expected by BG and BL.

The base case frequency estimates, and the corresponding variability, are not significantly affected by the time period (i.e., 25 versus 60 years) of operation. From Table 4.1, BL typically predicts LOCA frequency increases of less than a factor of 2 between his 60 and 25 year estimates for all five piping base cases and all LOCA categories. The DH 25 and 60 year results are identical for the PWR-1 base case. However, there is approximately a two order of magnitude increase in the PWR-2 estimates at 60 years when compared with 25 years and a one order of magnitude decrease in the BWR-1 Category 1 through 4 LOCA estimates at 60 years, but almost no change in the Category 5 LOCA estimates. For PWR-3, the DH 60-year estimates increase by approximately one half order of magnitude. Increases in the LOCA estimates with time in both analyses are caused by continued aging. Decreases typically result from repair of defects (or component replacement) due to the discovery of degradation through inspection or leak detection. The leak frequencies in DH's analysis are determined by averaging the cumulative LOCA probability over the time period being evaluated, either 25 or 60 years. Instantaneous frequencies may therefore differ from these simplified estimates.

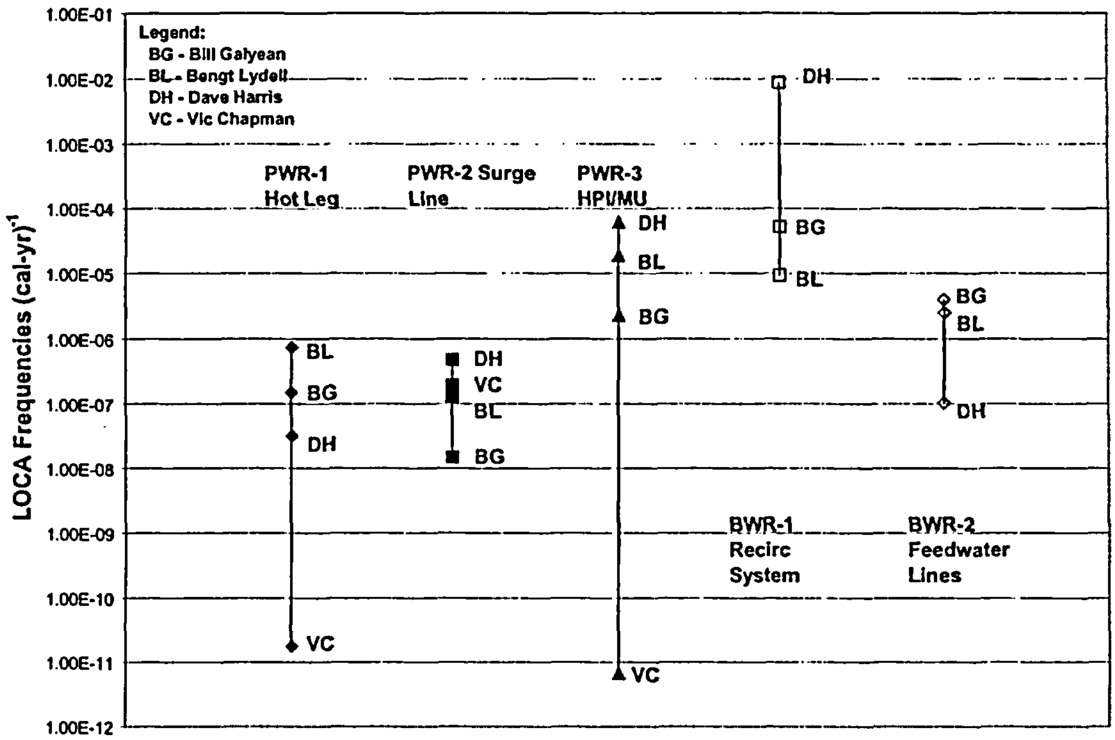

Figure 4.2 Category 1 Piping Base Case Frequencies at 25 Years

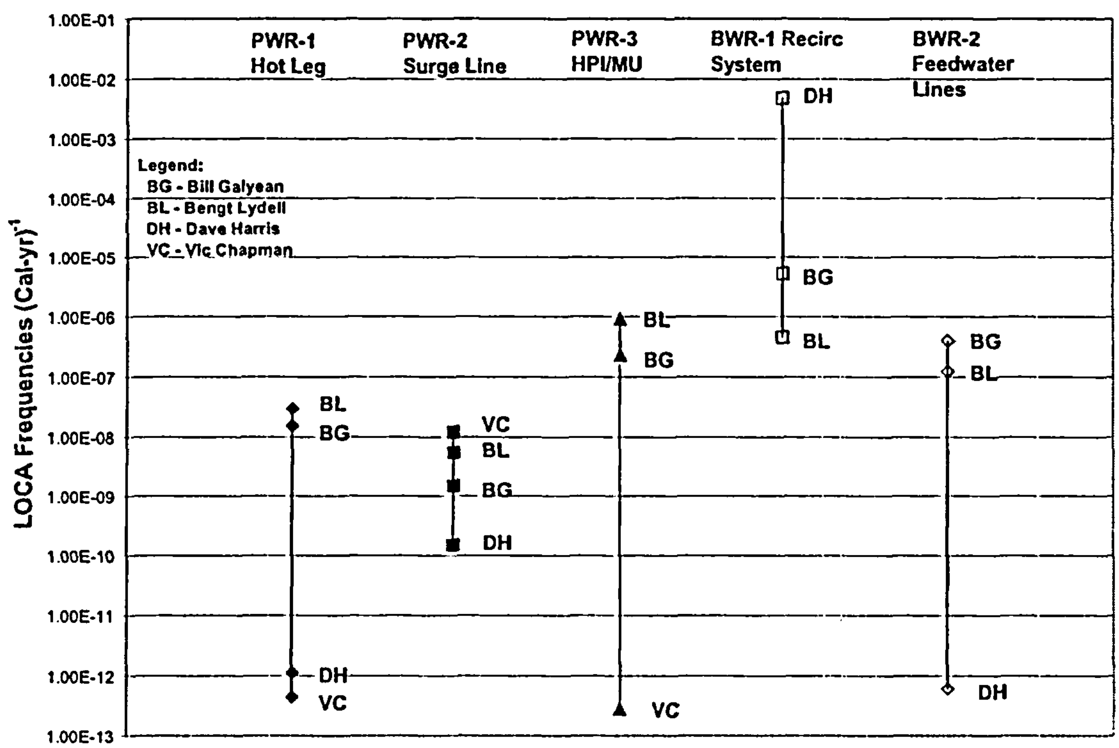

Figure 4.3 Category 3 Piping Base Case Frequencies at 25 Years

Figures 4.4 and 4.5 fully illustrate the relationship between the estimated base case frequencies and the LOCA size category. The PWR base case LOCA frequency estimates are illustrated in Figure 4.4, while the BWR base case results are shown in Figure 4.5. In both figures, the results for each base case are connected by a vertical line to indicate the variability among the results (as in Figures 4.2 and 4.3). The base case results have been offset slightly and the numbers above each vertical symbol represents the base case number. For example the 1 above the results in Figure 4.4 indicates that these results are for the PWR-1 base case while the 1 above the results in Figure 4.5 indicates that these results are for the BWR-1 base case. The results for each base case team member are presented using the same symbol type in both figures. The symbol fill is also varied to indicate the base case number.

Both BG and BL predict that the PWR-2 (surge line) frequencies are about an order of magnitude less than the PWR-1 (hot leg) frequencies for all of the LOCA categories (Figure 4.4). The DH predictions also predict a similar trend. Conversely, VC estimates that the PWR-2 frequencies are about 4 orders of magnitude higher than the PWR-1 frequencies. These VC differences are largely due to the fact that PWSCC was not modeled in his PWR-1 analysis as discussed previously. It is also apparent in Figure 4.4 (as in Figures 4.2 and 4.3) that the service history-based analyses are more consistent. For all LOCA categories, BL's PWR base case predictions are less than an order of magnitude higher than BG's predictions for all three PWR base cases. This is much closer agreement than the PFM results, but is not surprising given the similarity in the approaches discussed previously. Also, most of the differences among the LOCA Category 1 estimates remain relatively consistent as the LOCA size increases. In fact, three of the base case team members (BG, BL, and VC in Figure 4.4) predict approximately a ½ order of magnitude decrease with each successive LOCA size increase for all three PWR base cases. Only the DH results tend to predict a more dramatic decrease between the Category 1 and bigger LOCA sizes in the PWR-1 and PWR-2 base cases. Big decreases between the Category 1 and 2 predictions occur due to the expectation that the cracking in these base cases is much more likely to result in a smaller break size. Beyond Category 2, the DH LOCA frequencies decrease by approximately ½ order of magnitude per LOCA category up to LOCA Category 4. The DH results also decrease substantially between Category 4 and 5 for the PWR-1 base case.

There are several implications which result from these trends. The first is that the average conditional probabilities between successively increasing LOCA categories is approximately 0.5, and all four base case results are in relatively good agreement about this relationship. More importantly, and consistent with previous discussion (Figures 4.2 and 4.3), is the finding that much of the variability is apparent in the estimates of the smallest LOCA sizes for these base cases. If the LOCA Category 1 estimates were normalized to eliminate variability, subsequent LOCA Categories would not differ as widely.

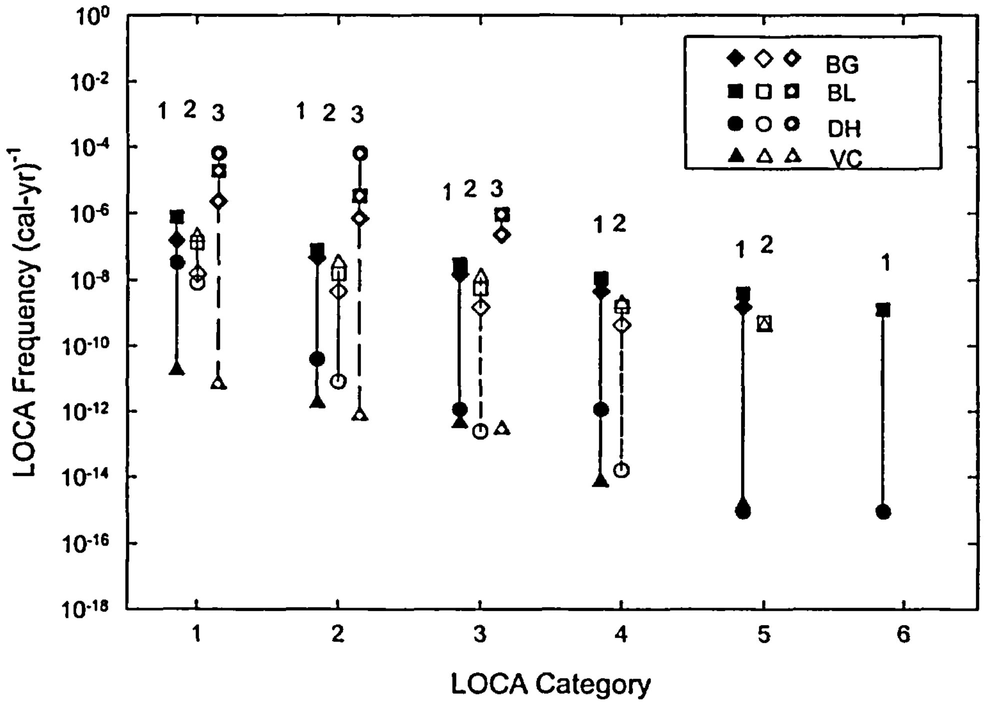

Figure 4.4 PWR Piping Base Case Frequencies at 25 Years of Plant Operation

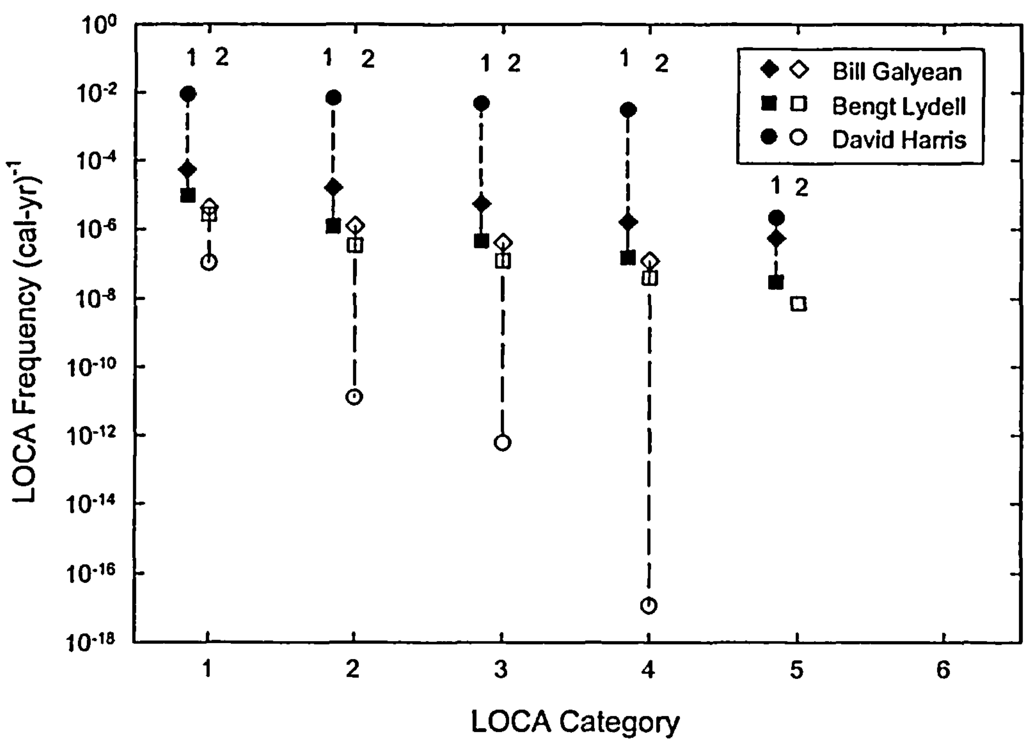

Figure 4.5 BWR Piping Base Case Frequencies at 25 Years of Plant Operation

The BWR base case results (Figure 4.5) exhibit many similar trends as the PWR results (Figure 4.4). Both BL and BG predict slightly higher frequencies for the recirculation system (BWR-1) than for the feedwater system (BWR-2). Both service history-based predictions are once again consistent but, contrary to the PWR results, the BG predictions tend to be slightly higher. The service history base case frequencies again decrease by approximately one half order of magnitude for with each successive LOCA size increase because the same conditional failure probability relationship is used as for the PWR analyses.

However, the DH results for BWR-1 LOCA Category 1 are two orders of magnitude higher than the other estimates. This is a consequence of using the service history-base through wall cracking frequency (Figure 4.1) to calibrate these PFM estimates. There is also very little expected frequency decrease between the through wall-cracking frequencies and LOCA Category 4 estimates. A precipitous, three orders of magnitude decrease in the DH estimates only occurs for LOCA Category 5. The implication is that DH predicts that through wall cracking and LOCAs are equally likely up to a break size of 7 inches (178 mm) (Category 4). Only breaks larger than this become much less likely. As in the PWR comparisons, variability among the BWR-1 estimates is largely a function of the variability evident in the Category 1 results.

The DH result for BWR-2 for the Category 1 LOCA frequency is much closer, yet lower than the service history estimates. The DH BWR-2 results for larger LOCA categories decreases precipitously compared to the other estimates. This increased variability with increasing LOCA size is contrary to the other base case results. However, as discussed previously, the DH BWR-2 analysis only modeled thermal fatigue and does not consider FAC contributions. Thermal fatigue Category 1 frequencies are therefore lower. Additionally, thermal fatigue failures are much more likely to result in small LOCAs due to leak-before-break, where any FAC failures are likely to result in larger LOCAs. This explains the increased variability with increasing LOCA size for the BWR-2 results.

4.3 Piping Base Case Sensitivity Studies

A number of sensitivity studies of the base case systems were conducted to evaluate the effects of various parameters on the piping base case frequencies. These studies were also conducted by various base case team members. They are useful for examining the effect of certain variables and analysis choices on the base case estimates. These analyses examined the effect of in-service inspections (ISI) using different leak detection strategies, the effect of different IGSCC mitigation strategies on the BWR-1 estimates, and the effect of the component load history for several base cases. A number of unique sensitivity analyses were conducted for the PWR-1 base case including evaluating the influence of hydro-testing, safe shutdown earthquake (SSE) seismic loading, and the effect of degraded material properties (notably fracture toughness) on the base case LOCA frequencies.

4.3.1 Effect of In-Service Inspection with Different Leak Detection Strategies

Bengt Lydell (BL) used a Markov model approach to investigate the sensitivity of the base case LOCA frequencies to different in-service inspection (ISI) strategies. This approach determines the ISI effectiveness factor, *I*, which is defined as the ratio of the time-dependent LOCA frequencies with inspections to the applicable frequencies if no inspections were conducted. Three different inspection strategies were investigated: (1) no ISI, (2) Section XI inspection with probability of flaw detection (POD) equal to 0.5, and (3) risk-informed ISI with the POD equal to 0.9. See Appendix D for more details.

The effect of ISI was evaluated for all the base case systems. Figure 4.6 illustrates the results for the BWR-1 evaluation for Category 1 LOCAs. Here, baseline frequencies are depicted as a function of operating time assuming that no active ISI is conducted. However, plant walk downs are performed with varying periodicity to search for primary system leaks. Four different walk down periodicities are considered: none, after the hydro test at each refueling outage, weekly, and during each shift. As can be seen in Figure 4.6, the Category 1 LOCA frequency decreases by about one order of magnitude for each successive periodicity increase between no walk down, refueling outage walk downs, and weekly walk downs. However, there is little additional benefit to increasing the walk down frequency to more than a weekly basis. Also, most of the benefits associated with increased inspection periodicity occur over the first 25 years of service. Beyond 25 years of plant operation, the slopes of the LOCA frequency results (Figure 4.6) are not a strong function of inspection periodicity. Specifically, the 60 year frequencies are about ½ an order of magnitude greater than the 25 year estimates for the range of reported walk-down periodicities. This increase is consistent with BL's 25 and 60 year base case results (Table 4.1).

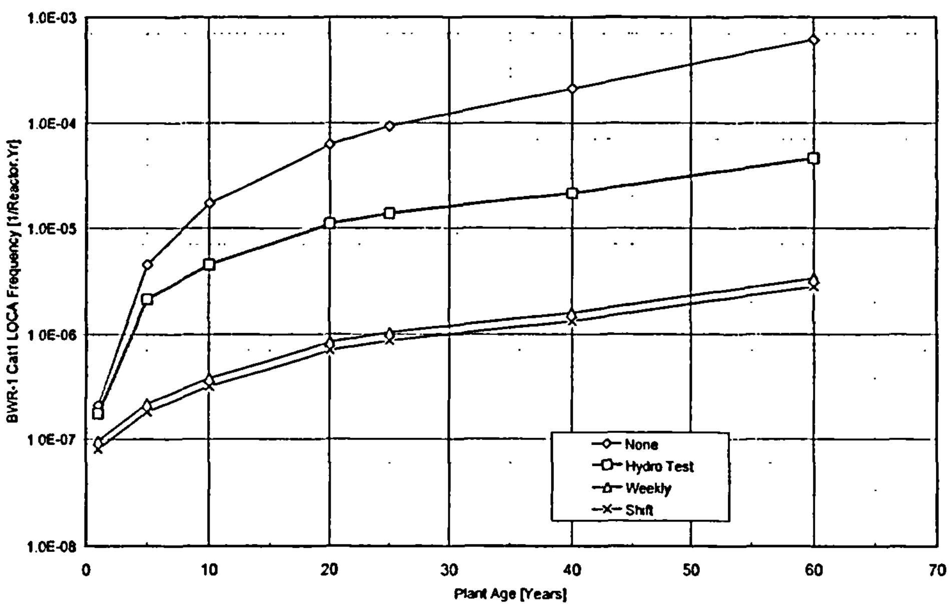

Figure 4.6 BWR-1 LOCA Frequencies as a Function of Plant Operating Time for the Case of No ISI with Different Leak Detection Strategies

Figure 4.7 illustrates the effect of different walk down periodicities on the BWR-1 base case Category 1 LOCA frequencies when ISI is routinely performed and the probability of detection (POD) is 0.9. The same four leak detection strategies as portrayed in Figure 4.6 are included. Comparing Figures 4.6 and 4.7, one can see that the ISI program does not significantly decrease the LOCA frequency estimates. Without performing walk downs, the LOCA frequencies after 25 years of plant operations are 9×10^{-5}/reactor year without ISI (Figure 4.6) and 5×10^{-5}/reactor year with the assumed ISI (Figure 4.7). This less than a factor of two difference is not terribly significant. Differences are even less when the probability of detection drops to 50%. For example, the BWR-1 Category 1 LOCA frequency after 25 years of plant operations is 6×10^{-5}/reactor year when the POD is 50% and no walk downs are performed. The effect of ISI is similar when other walk down periodicities are considered. Additionally, these trends are consistent for the other base case piping systems. This analysis therefore suggests that conducting frequent plant walk downs is a better strategy for decreasing LOCA frequency risk than is performing ISI. More details are provided in Appendix D.

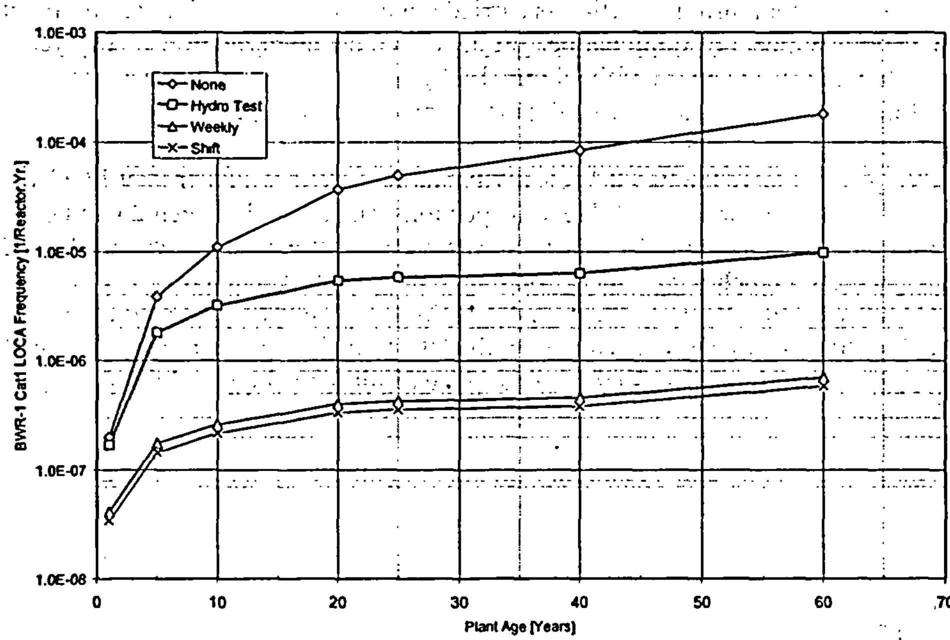

Figure 4.7 BWR-1 LOCA Frequencies as a Function of Plant Operating Time for the Case of ISI with 90% POD with Different Leak Detection Strategies

4.3.2 PFM ISI Assessment

An assessment of the effect of ISI was also carried out using PFM analysis by VC for the surge line elbow weld in the PWR-2 base case. The defect distribution and density was generated by RR-PRODIGAL. The POD relationship as a function of the crack depth to component thickness ratio (a/t) was defined by the following equation:

$$f_{\text{POD}} = \Phi\left(c_1 + c_2 \ln\left(\frac{a}{t}\right)\right) \qquad \text{where } c_1 = 1.526 \text{ and } c_2 = 0.533 \qquad (4.1)$$

This POD curve is shown in Figure 4.8 and the c_1 and c_2 parameters were chosen so that the POD for defects that are 70 percent of the component thickness is approximately 0.90. This POD is believed to conservatively represent historical detection resolution, but for future inspections that conform to more modern standards, this POD could be much better. This POD relationship is more complex than the constant POD assumed in the previous ISI sensitivity analysis (Section 4.3.1). More details are provided in Appendix G.

The results of this ISI sensitivity study on the 60 year cumulative failure probabilities are summarized in Table 4.2 and Figure 4.9 for various ISI intervals. The reduction factor in the table and figure is defined as the ratio of LOCA frequency without ISI to the LOCA frequency with the assumed ISI periodicity. Periodicity varies from no ISI to inspections conducted every 10 years. A special case assuming only a pre-service inspection (PSI) is also considered. These results suggest that even with this rather poor inspection capability, and for a weld with a high failure probability, reductions in the cumulative probability of failure of a decade or more can be achieved with two or three inspections during the life of the plant. The results also indicate that going beyond three inspections leads to little additional benefit assuming that indications are not uncovered. In fact, if a fourth inspection is carried out at forty years and no repairable indications are found, there is almost no additional benefit in conducting an inspection at fifty years. It should be noted that this analysis of the thermal fatigue risk to the surge line assumes that no new cracks are formed during service. More details are provided in Appendix G.

It is interesting to compare the two ISI sensitivity studies. Both results predict LOCA frequency reductions of about a factor of 2 to 3 due to inspections carried out between 25 and 60 years of plant operation on the applicable 60 year LOCA frequency estimates. However, the VC study in this section predicts a much bigger reduction due to inspections carried out prior to 30 years of service. Conversely, the BL study (Section 4.3.1) does not predict similar reductions due to ISI during the first 30 years of service. However, the BL study does predict that periodic plant walk down inspections during the first 30 years of service result in the biggest LOCA frequency reductions.

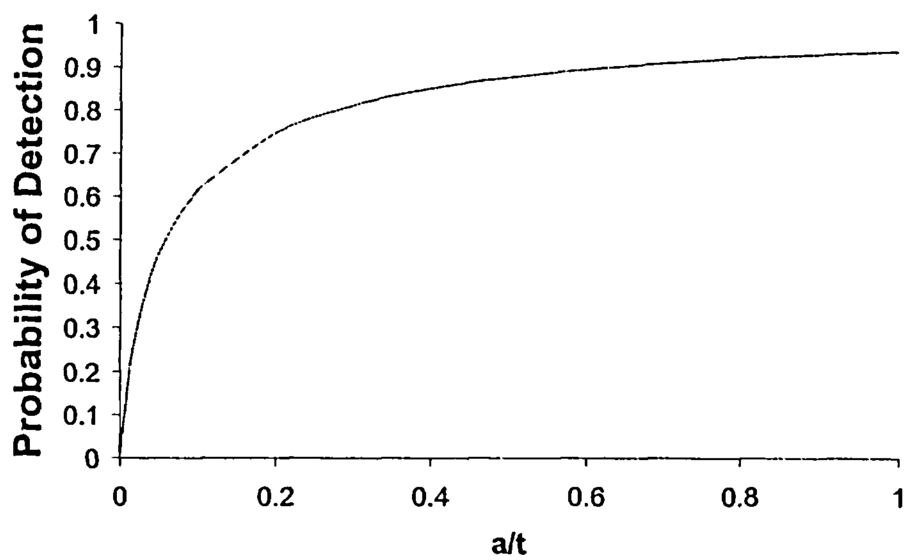

Figure 4.8 Probability of Detection (POD) Curve Used by Vic Chapman in ISI Sensitivity Analysis

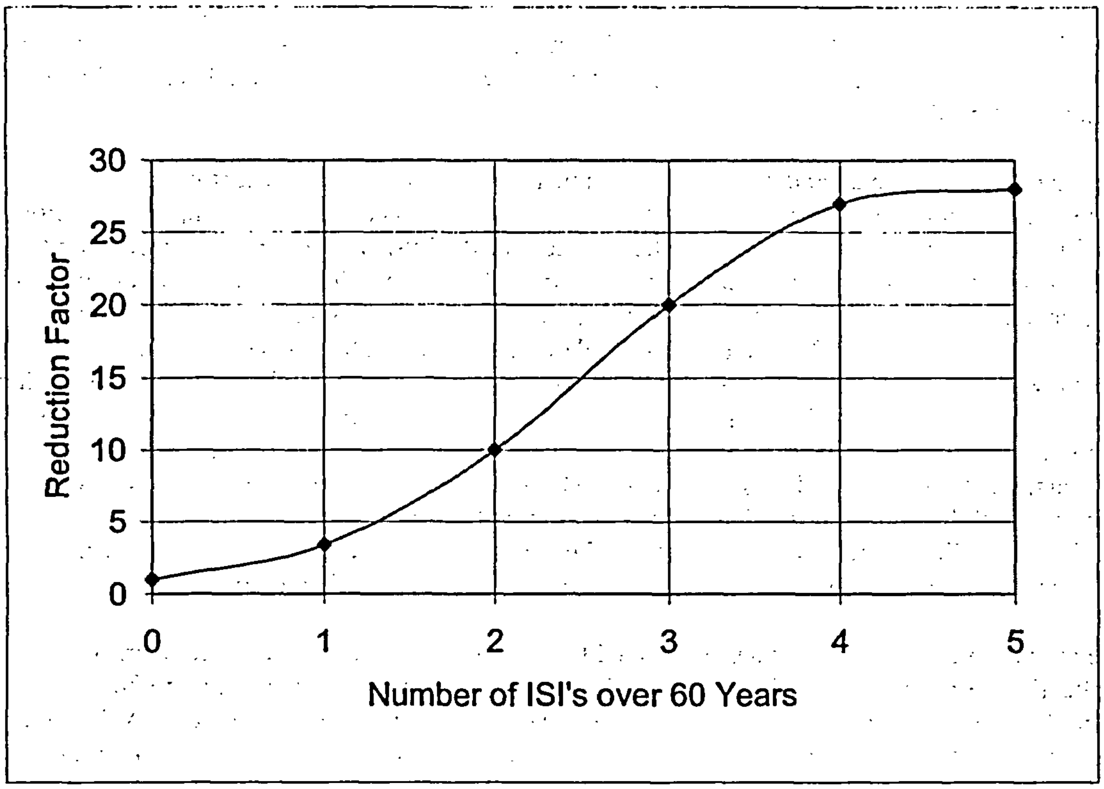

Figure 4.9 Graphical Results from Vic Chapman's ISI Sensitivity Analysis

Table 4.2 Tabular Results from Vic Chapman's ISI Sensitivity Analysis

ISI case	Cumulative Probability of Failure at 60 years	Factor for General Use
No ISI	1.3 x 10-4	1
0 years (PSI)	4.2 x 10-5	3
10 years	3.8 x 10-5	3.4
10, 20 years	1.3 x 10-5	10
10, 20, 30 years	6.5 x 10-6	20
10, 20, 30, 40 years	4.8 x 10-6	27
10, 20, 30, 40, 50 years	4.7 x 10-6	28

4.3.3 Effect of Weld Overlay on IGSCC Mitigation

The BWR-1 base case assumed that a weld overlay was installed at 20 years of service to mitigate the effects of IGSCC. In order to examine the effect of this mitigation strategy on the estimated LOCA frequencies, a sensitivity analysis was performed assuming that no weld overlay was applied. The piping system chosen for this analysis was the 12-inch diameter section of the recirculation system defined within the BWR-1 base case. There are two similar recirculation loops in this base case with a total of 121 field, shop, and safe end welds. The piping is assumed to be fabricated from Schedule 80 Type 304 stainless with a nominal wall thickness of 0.687 inches (17.4 mm).

Because IGSCC is the degradation mechanism being considered, the time at stress primarily influences the cracking rate while the number of stress cycles is of secondary importance. For the baseline condition, the default residual stress pattern in PRAISE was assumed. These default residual stresses for intermediate size lines vary around the circumference and linearly through the thickness. The residual stress at a given angular location at the ID is normally distributed with a mean of 1.86 ksi (12.8 MPa) and a standard deviation of 2.89 ksi (19.9 MPa). These values were derived from benchmarking the PRAISE results with field observations of leaks for the recirculation system. The mean residual stress at the OD is -1.86 ksi (-12.8 MPa), and the standard deviation is the same as at the ID. For the case where the weld overlay was applied at 20 years, an alternative residual stress field [4.2] was assumed (Figure 4.10). The PRAISE code cannot model the nonlinear gradients illustrated in this figure, so a linear approximation was used. The linear gradient generally underestimates the beneficial effect of the weld overlay within the inner half of the pipe wall thickness. More details are available in Appendix F.

The cumulative probability for a Category 1 LOCA, both with and without weld overlay application, are illustrated in Figure 4.11. The results, of course, are identical up to 20 years when the overlay is applied. The derivative of the curves in Figure 4.11 is the LOCA frequency as a function of time. After 40 years, the LOCA frequency (i.e., derivative of curve) with no overlay is about 7 times greater than the frequency with the overlay applied. The effect of the weld overlay also leads to consistent LOCA frequency reductions for LOCA Category 2 – 4 (not shown) at both 40 and 60 years. More detailed results are provided in Appendix F.

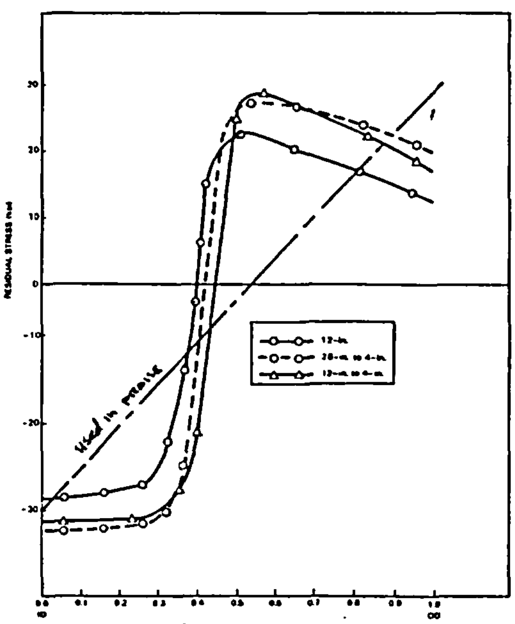

Figure 4.10 Through-Wall Residual Axial Stress Distribution Used in Weld Overlay Analyses

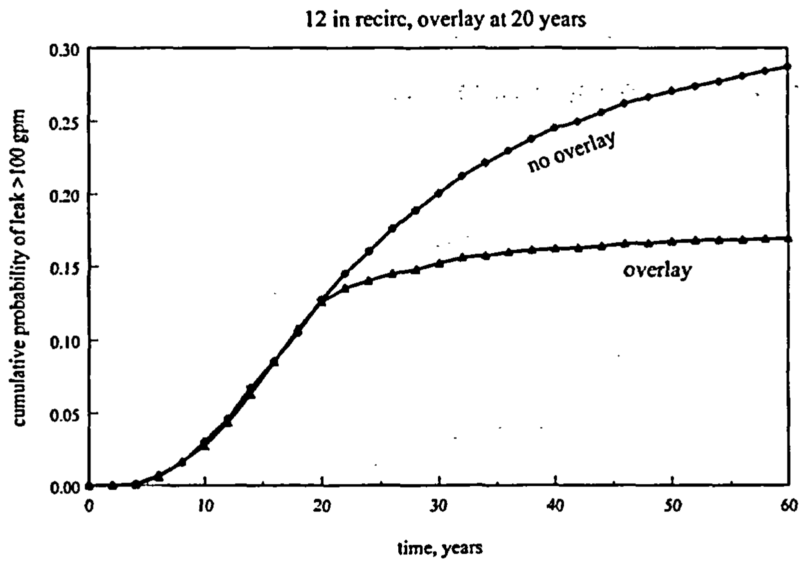

Figure 4.11 Effect of Weld Overlay Repairs on the Cumulative Category 1 LOCA Frequencies for 12-inch Diameter BWR Recirculation System Piping

4.3.4 General IGSCC Mitigation Effectiveness

Weld overlays are just one of several IGSCC mitigation strategies for BWR plants. Other effective strategies include hydrogen and noble metal water chemistry improvements, pipe replacement with more crack resistant materials, and post-weld heat treatment of susceptible welds. These strategies are coupled with higher quality inspections to identify cracking before it gets severe. All of these methods have been used by various plants for mitigating IGSCC. The service history data contains information on IGSCC cracking events both before and after these various mitigation techniques were generally applied.

While it is more difficult to assess the effectiveness of individual mitigation techniques, it is straightforward to examine the general effect of mitigation on the IGSCC cracking frequency. Bill Galyean simply segregated the service history data into events prior to 1985 and events after 1985 to represent the pre-mitigation and post-mitigation periods, respectively. He then calculated the average IGSCC cracking frequency for the pre-mitigation and post-mitigation time periods. This LOCA precursor frequency is then converted into LOCA frequencies for each category using the generic conditional LOCA probability relationship that BG used to develop his base case frequency estimates. This analysis reveals that the pre-mitigation cracking frequency is approximately 60 times higher than the post-mitigation frequency for all LOCA categories (Figure 4.12). In BG's analysis, the cracking frequency is assumed to be directly proportional to the LOCA frequency, so that this factor separates the pre-mitigation and post-mitigation LOCA frequencies as well. More details are provided in Appendix E.

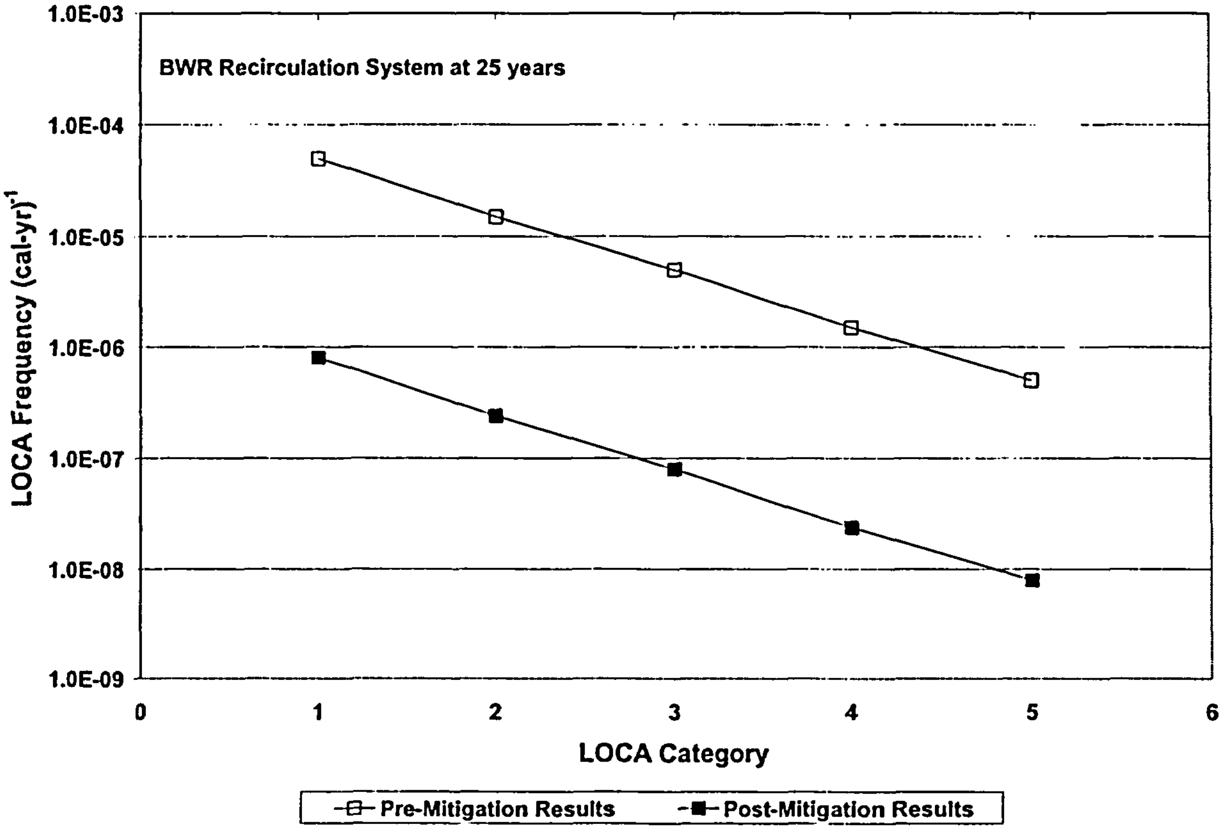

Figure 4.12 Pre and Post-Mitigation Recirculation LOCA Frequencies

4.3.5 Effect of Applied Loading Magnitude

The load history and applied stress magnitude significantly influence the LOCA frequencies calculated using PFM analysis of components. David Harris (DH) conducted a sensitivity analysis for the BWR-1 base case to examine the effect of the normal operating stresses on precursor leak and LOCA frequencies. The analysis was conducted for a 12-inch diameter recirculation system for which a weld overlay repair was installed at 20 years of plant operations (part of the BWR-1 base case). The PRAISE code was used for this analysis and frequencies for a single weld joint are summarized in Figure 4.13. Obviously, the precursor leak and LOCA frequencies for a modeled weld increase as the normal operating stresses increase.

The predicted precursor leak frequency for the 12 ksi (83 MPa) mean normal operating stress is $6x10^{-4}$ per weld reactor year. This frequency coincides quite well with the pre-mitigation cracking rates experienced in service (see Appendices D and F). Increasing the normal operating stress from 12 to 15 ksi (83 to 103 MPa) (Figure 4.13) leads to approximately an order of magnitude increase in the frequencies associated with all break sizes. A smaller frequency increase occurs when the applied stress increases from 15 to 20 ksi (103 to 138 MPa).

The selection of the applied stress for the base case analysis is not as important as Figure 4.13 would indicate because as the assumed applied stress increases, the number of weld joints in a system which will be subject to that stress level decreases. The 20 ksi (138 MPa) applied stress was chosen for the BWR-1 base case analysis because this was more consistent with the approach used for the other systems. However, it was assumed that this high applied stress only exists at 2 weld joints in the recirculation system. The lower 12 ksi (83 MPa) stress is likely to exist at all 49 welds in the simulated system. Therefore, both stresses lead to similar recirculation system leak frequencies (Table 23 of Appendix F) and both stresses correspond to the service experience-based leak frequencies (Figure 4.1) used for benchmarking the base case the analysis. In actual service, variability in the residual stress magnitude and distribution also influences the LOCA susceptibility of a particular weld joint. Possible residual stress variability has not been considered in this analysis, but higher tensile stresses at the inside pipe surface than assumed will increase the LOCA susceptibility of that weld.

The effect of the applied loading magnitude was also evaluated for the PWR-3 (HPI/MU nozzle) base case. Thermal fatigue is being modeled in this base case and the cyclic loading magnitude is therefore the principal consideration. In these results, the applied cyclic stresses is 25 ksi (172 MPa) in order to properly benchmark the service history cracking frequencies associated with this base case (Figure 4.1). However, while these stresses lead to the highest leak frequencies, the largest LOCA frequency occurs for lower applied cyclic stresses. The reason is that higher cyclic stresses promote leak-before-break in this component by fostering through-wall crack growth instead of surface crack growth that is associated with higher LOCA frequencies. See Appendix F for additional details and results of this analysis.

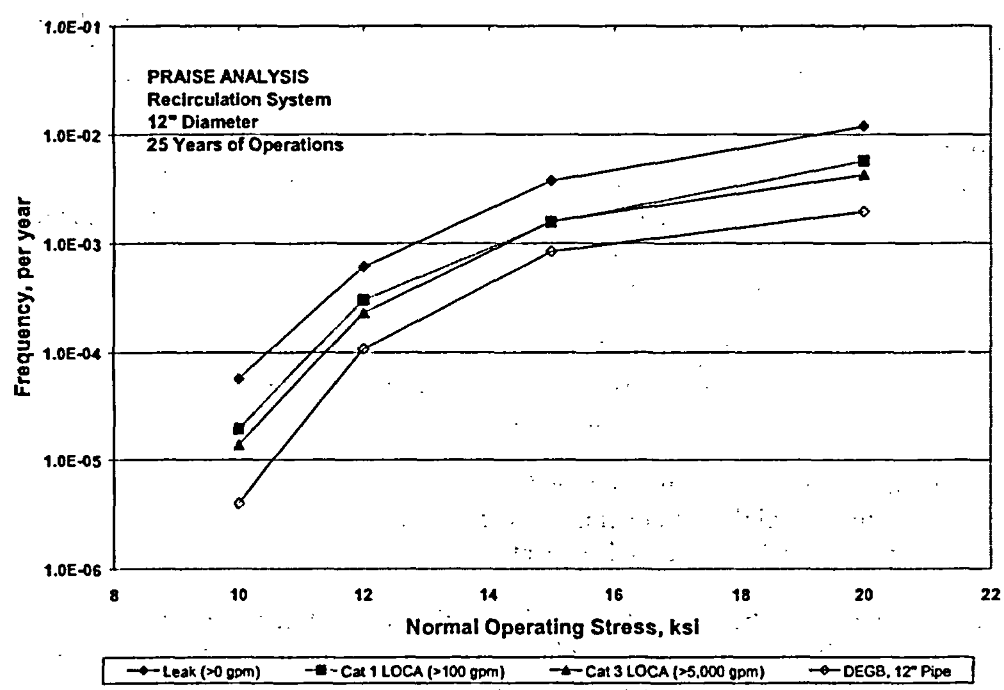

Figure 4.13 PRAISE Analysis Showing Effect of Normal Operating Stress on Various Size Precursor Leak and LOCA Frequencies for a Single Weld Joint in 12-inch Diameter Section of Recirculation System

4.3.6 Effect of Seismic Loading and Hydro Test

The impact of seismic stress was not explicitly considered within the elicitation because the principal objective of this elicitation (Section 2) is to develop LOCA frequencies under normal operating loads and expected transients. However, it is relatively straightforward to examine the effect of a prescribed seismic loading on the LOCA frequencies. The elicitation panel indicated that these sensitivity calculations would be beneficial and could serve as a surrogate for the effect of other rare emergency-faulted loads. The hydro test is conducted prior to initial plant operation. While the base case estimates include this test, it is also straightforward to evaluate the effects that this test has on LOCA frequencies.

The role of an assumed safe shutdown earthquake (SSE) seismic loading and the hydro test was studied for hot leg (PWR-1) failure frequencies. However, while hot leg PWSCC cracking is the PWR-1 base case (Table 3.7), only the impact of seismic and hydrotest loading on the LOCA frequencies associated with pre-existing defects growing by fatigue was modeled. These frequencies are only one component of the total LOCA frequencies associated with this base case. The effect of seismic loading or hydrotesting cracks initiating and growing due to PWSCC was not considered. However, the initial hydrotest does not significantly alter the failure probabilities associated with service-induced crack initiation and growth. The PRAISE code was used by DH for this analysis and seismic stresses up to 5 SSE were applied at either 25, 40, or 60 years of operation. Only one seismic load application was applied in any single analysis. This sensitivity analysis utilized similar inputs and assumptions as this base case analysis (Appendix F). The effect of the hydro test is summarized in Figure 4.14 while the effect of SSE loading is depicted in Figure 4.15.

The initial hydro test moderately affects the LOCA frequencies (Figure 4.14). The Category 1 LOCA frequencies attributed to thermal fatigue decrease by an order of magnitude while the Category 6 frequencies decrease by approximately ½ an order of magnitude with the application of the hydro test. These trends are generally consistent even if SSE loading is not included (Appendix F). The general decreases occur because hydro testing can identify flaws before the component is place in service that could eventually lead to thermal fatigue failures during service. Bigger Category 1 decreases exist because the thermal fatigue cracks modeled that grow from the pre-existing flaws, which are discovered through hydro-static testing, more likely lead to smaller LOCAs. Hence, the hydro test results in a bigger decrease of the Category 1 LOCA frequencies. Also, of note from Figure 4.14 is the fact that the effect of the hydro test is slightly more pronounced at 60 years than it is at 25 or 40 years.

Seismic SSE loading magnitudes (Figure 4.15) have a minimal effect on the Category 1 and Category 6 LOCA frequencies. There is also no substantial LOCA frequency increase for 5SSE loading. This minimal effect is attributed to the small SSE stresses that are applicable for the hot leg configuration modeled. These stresses are plant specific and other configurations, resulting in higher stresses, could have a greater impact on the LOCA frequencies. The effect of seismic loading appears to decrease slightly with operating time (Figure 4.15). This trend is likely due to identification and repair of cracks in each subsequent inspection.

However, in all these analyses, the LOCA frequencies attributed to fatigue from preexisting defects are extremely small regardless of the application of seismic loading or the hydro test. These frequencies are much less than the earlier PWR-1 base case estimates made by DH (Table 4.1 and Figure 4.4). It is not known if these trends related to seismic stress and hydro test application have similar effects when PWSCC failure is considered and the LOCA frequencies are higher. The greater crack propensity of PSWCC due to the elevated crack initiation and growth rates may cause more significant LOCA frequency increases due to the same applied seismic loading. Additionally, the hydro test may be less effective in screening for PWSCC cracks than thermal fatigue cracks that initiate from pre-existing defects because the PWSCC cracks initiate after service commences.

An additional sensitivity study was performed for seismic events for the PWR-2 base case. There are differences in the treatment of seismic events between PWR-1 and PWR-2 sensitivity analyses, because of the level of detail of the seismic stresses that was available. For the PWR-1 case, specific stresses for the seismic events were available (as discussed in Appendix F), and the seismic stresses were applied once - at 25, 40 or 60 years as previously detailed. For the PWR-2 case, a list of cyclic stress amplitudes was available, which included seismic events. The stress history without seismic loading was determined using rainflow counting with seismic stresses set to a low value. Therefore a different set of stress amplitudes was considered to evaluate the seismic loading effects (see Tables F.8 and F.9 of Appendix F). For the PWR-2 base case, when seismic events are considered, the seismic stresses do not occur at a specific time. Due to the large number of high stress cycles for PWR-2, fatigue crack initiation was considered in this sensitivity analysis. This contrasts the PWR-1 sensitivity analysis which only evaluated the seismic and hydrotest loading impact on LOCA frequencies. The baseline LOCA frequencies are much higher in the PWR-2 analysis than in the PWR-1 and the SSE loading does increase the LOCA frequencies by a factor of 3. This is a larger effect than previously predicted in the PWR-1 sensitivity analysis, but the increase is still not substantial. More details are provided in Appendix F.

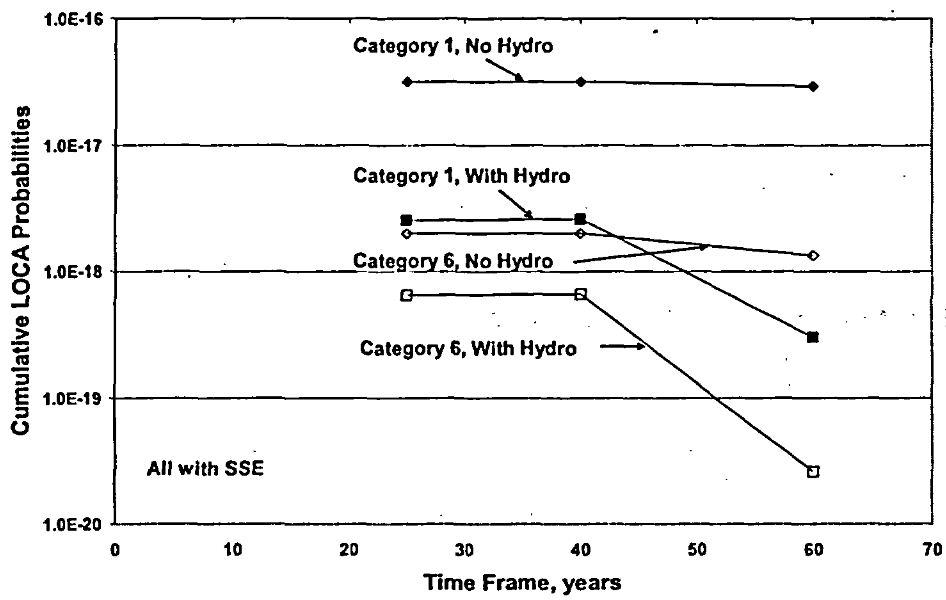

Figure 4.14 Effect of a Hydro Test on the Cumulative LOCA Probabilities for the Hot Leg with a SSE Load Included

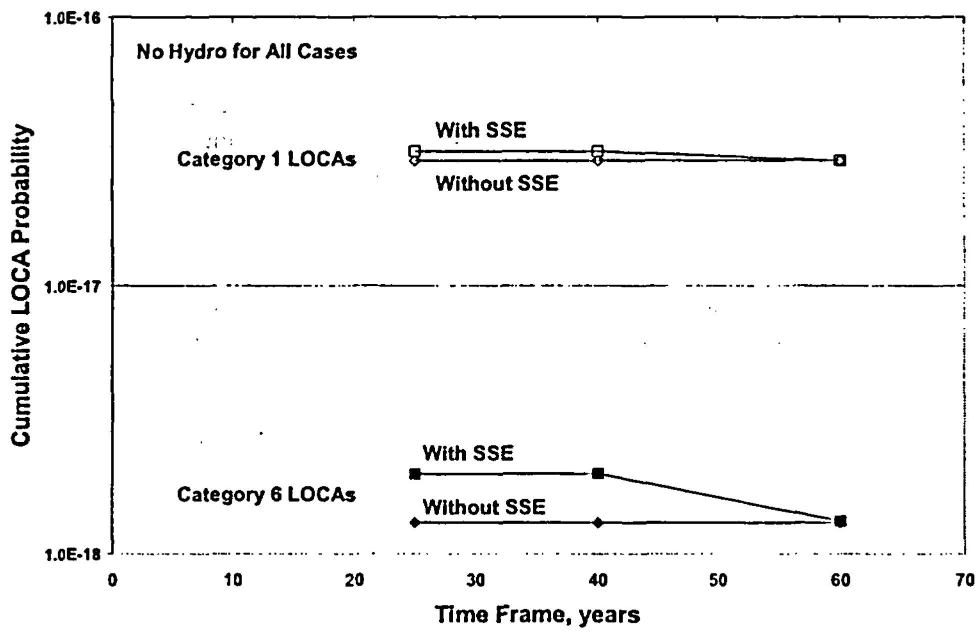

Figure 4.15 Effect of a SSE Load on the Cumulative LOCA Probabilities for the Hot Leg Without a Hydro Test

4.3.7 Effect of Degraded Material Properties

The hot leg was again analyzed by DH to examine the effect of degraded material properties on the predicted LOCA probabilities. The effect of thermal aging was simulated by increasing the material flow stress and decreasing the fracture toughness. Several different aging conditions were evaluated using the most embrittled properties consistent with a highly aged, cast austentic stainless steel alloy (CF8M). Once again, only thermal fatigue from pre-existing defects was modeled using the PRAISE code with similar inputs as the base case analysis. See Appendix F for more details.

Results for Category 1 LOCA frequency predictions are summarized in Figure 4.16. As with the hydro test and SSE hot leg sensitivity studies, the absolute failure frequencies are low because only fatigue of pre-existing defects was modeled. The baseline LOCA frequencies in Figure 4.16 are consistent with the analysis considering the application of a hydro test and no SSE loads (Appendix F). However, the LOCA frequency increases due to fracture toughness reductions are much more significant than the effects of a hydro test or SSE seismic loading (Figures 4.14 and 4.15). In this case, the limiting aging properties for CF8M causes a 5 order of magnitude increase in the LOCA frequencies compared with the unaged, baseline LOCA estimates. More modest toughness decreases still result in an order of magnitude increase in the predicted LOCA frequencies. The results of Figure 4.16 are not time dependent. Time dependency is a function of material property degradation with continued aging. However, the PRAISE code can only consider time-invariant material parameters. Continuing material property degradation with aging would be expected to result in increasing LOCA frequencies with time.

It is not known if similar order of magnitude increases would result if these aged properties were used in the complete PWR-1 base case analysis. This consideration is largely irrelevant because significant aging toughness decreases are not expected for the hot leg weld material. However, this analysis does illustrate the importance of the weld fracture toughness on the LOCA susceptibility when a cracking mechanism is present. These results are consistent with relative differences described in Section 1.2 for a previous study on the effects of material toughness [4.3].

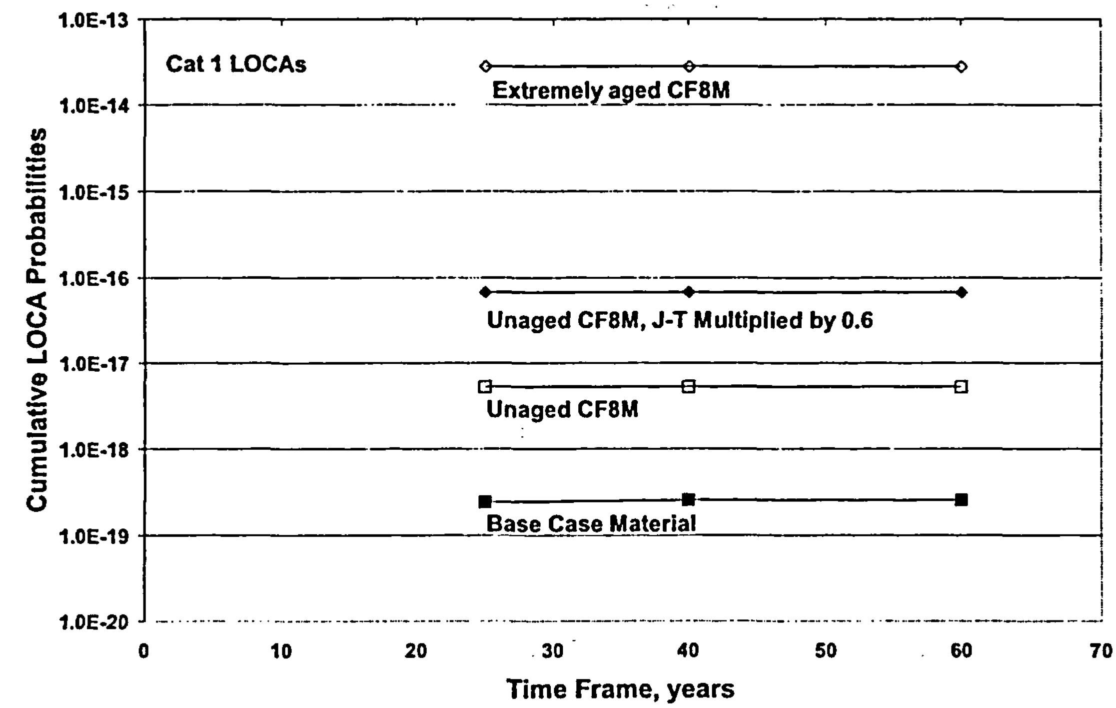

Figure 4.16 Effect of Material Properties (Fracture Toughness) on the Cumulative LOCA Probabilities for the PWR-1 Base Case

4.4 Non-Piping Base Case Results

In addition to developing base case frequencies for piping systems, base case frequencies were also developed for a number of non-piping components. A non-piping precursor database based on a review and analysis of licensing event reports (LERs) was developed. This database is described in more detail in Appendix H. In addition, a series of separate studies were conducted to develop base case frequencies for a number of non-piping components and subcomponents. These studies included: *a steam generator tube rupture frequency study, an overview of the* pressurized thermal shock (PTS) re-evaluation effort, and a BWR vessel rupture and PWR CRDM ejection analyses. More information on the details associated with each of these base cases is provided in Section 3.5.2. The results from each of these studies are provided in this section. Also, additional details and results for the BWR vessel rupture and PWR CRDM ejection analyses can be found in Appendix I.

4.4.1 Steam Generator Tube Rupture Base Case

The LER non-piping database was used to conduct this study as explained more fully in Section 3.5.2.2. From 1990 to 2002 there were 15 reports of steam generator tube leaks. There is a total of 929 reactor calendar years represented in this period, so the mean leak frequency over this period is 16×10^{-3} per calendar year. The leak events plotted as a function of calendar year are summarized in Figure 4.17. Two year increments have been selected for binning purposes.

While the number of tube leaks per year is relatively small, these events appear to be relatively constant over this time-frame. The base case steam generator tube rupture frequency was also assumed to constant over this time period under the presumption that the leaking event frequency is directly proportional to the tube rupture frequency.

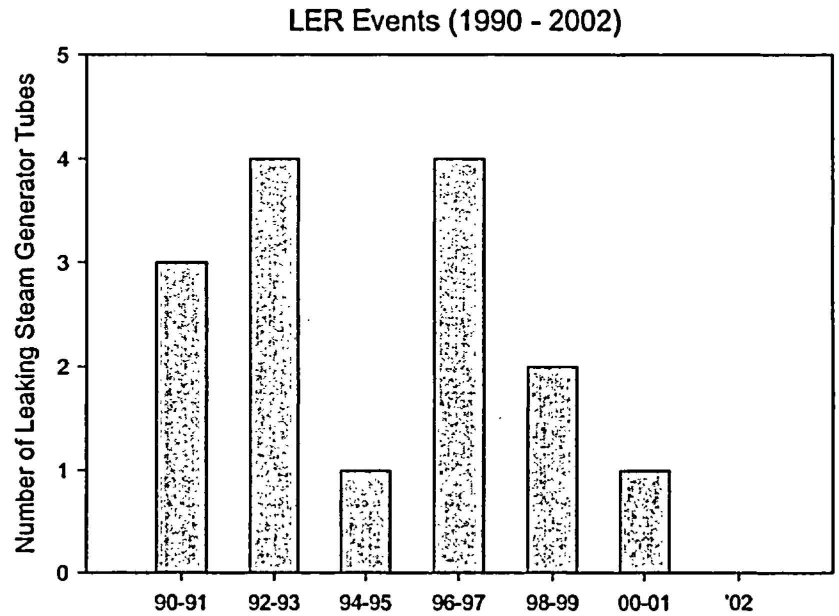

Figure 4.17 Steam Generator Tube Leaks as a Function of Event Date

In assessing the steam generator tube rupture data, the database was queried over a 15 year time period between 1987 and 2002. This search was expanded over the prior 1990 to 2002 study in order to capture a couple of known steam generator tube ruptures that occurred in 1987 and 1989. During this 15 year period, there were a total of 4 steam generator tube ruptures with resultant leak rates greater than 100 gpm (380 lpm) (LOCA Category 1). These 4 ruptures occurred at North Anna in 1987, McGuire in 1989, Palo Verde in 1993, and Indian Point in 2000. This same 15 year time period represents 1133 calendar years of reactor operation. Therefore, the frequency of steam generator tube Category 1 ruptures (with resultant leak rates greater than 100 gpm [380 lpm]) was 4/1133 calendar years, or 3.5×10^{-3} per calendar year. NUREG/CR-5750 [4.1] conducted a similar assessment of steam generator tube ruptures, and estimated a frequency of 7×10^{-3} per calendar year. These two frequencies are consistent because there has been one additional steam generator tube rupture since the completion of NUREG/CR-5750 at Indian Point. However, the number of reactor operating calendar years is more than twice in the NUREG/CR-5750 estimate

This steam generator rupture Category 1 LOCA frequency of 3.5×10^{-3}/calendar year was used as the base case value for this elicitation exercise. This base case frequency estimate was provided to each panelist. Each panelist then, as part of their elicitation, assessed the adequacy of this Category 1 LOCA steam generator tube rupture frequency estimate for both the current day and future operating time periods. This frequency estimate was also provided for anchoring the relative LOCA contributions from other Category 1 LOCA sources, multiple steam generator tube failures resulting in Category 2 and higher LOCAs, and larger LOCA (Category 2 – 6) failures in other piping systems and non-piping components per the discretion of each panelist.

4.4.2 Pressurized Thermal Shock (PTS) Base Case

The potential LOCA contribution due to pressurized thermal shock (PTS) of the RPV was analyzed using available data from the NRC's PTS Re-Evaluation Project [4.4] as discussed in Section 3.5.2.3. The total PTS through-wall cracking frequency results for 3 plants (Oconee, Beaver Valley, and Palisades) at 32 effective full-power years (EFPY) was first obtained from current results. The EFPY estimate most closely corresponds to the current fleet average of 25 years used to represent current day frequencies within the elicitation. The total PTS through-wall cracking frequencies for each plant were then adjusted to consider only the contributions due to stuck open primary side valves, stuck open secondary side valves, and feed and bleed operations. This adjustment removes the frequency contributions due to passive system LOCA failures which are subject to revision based on the revised LOCA frequencies determined by this elicitation.

The adjusted results for each plant are summarized in Table 4.3. The geometric mean of the 32 EFPY results for these three plants is also determined along with the 5th and 95th percentile of the mean estimates (Table 4.3). The 5th and 95th percentile are estimated by determining the values that lay two standard deviations from the geometric mean estimates using the results from the three plant-specific calculations. These percentiles provide some indication of the plant specific variability of the PTS through-wall cracking estimates. While these base case estimates are developed from the 32 EFPY results, the PTS through-wall cracking frequencies at 60 EFPY increase by less than a factor of two for each plant. Therefore, this calculation is not particularly sensitive to the future operating time period considered in this elicitation. The results from this PTS assessment were provided to the panelists for benchmarking other large piping and non-piping failures with expected low LOCA frequencies.

Table 4.3 Average and Average Plus and Minus Two Standard Deviations of the Adjusted Through-Wall Crack (TWC) Frequency from the PTS Re-Evaluation Project

Plant	Adjusted Category 6 LOCA frequencies from PTS study (per calendar year)	Geometric Mean (per calendar year)	95th Percentile (per calendar year)	5th Percentile (per calendar year)
Oconee	3.6E-11	5.1E-10	7.0E-08	3.7E-12
Beaver Valley	7.5E-10			
Palisades	4.8E-09			

4.4.3 BWR Vessel Rupture and PWR CRDM Base Cases

BWR vessel beltline weld failure was modeled to calculate failure frequencies due to normal operational loading and low temperature over pressure (LTOP) events. In addition, the large, predominantly feedwater, nozzles (6 to 28 inch) were analyzed to predict failure frequencies due to thermal fatigue. Both contributions are combined to provide an estimate of the total BWR RPV LOCA frequency as a function of LOCA size for use as a base case. The individual nozzle and beltline results were also available for anchoring. The PWR CRDM leak and ejection frequencies were used to determine the associated LOCA frequencies for use as a base case. A more complete description of this analysis is provided in Section 3.5.2.3 and in Appendix I.

Table 4.4 summarizes the base case LOCA frequencies for the BWR beltline weld and feedwater nozzle subcomponents as well as the combined LOCA frequencies for the BWR RPV. Table 4.5

presents the PWR base case LOCA frequencies due to CRDM ejection. LOCA frequencies are provided at 25, 40, and 60 years of plant operation in both tables. The PWR CRDM and combined BWR vessel LOCA frequencies are graphically summarized in Figures 4.18 and 4.19, respectively. The predicted 25-year PWR CRDM Category 1 LOCA frequency (Figure 4.18) is relatively high and is consistent with cracking experience within the plants. The CRDM LOCA frequencies decrease by less than an order of magnitude between the LOCA Categories 1 and 2. Category 1 LOCAs represent a partial CRDM break while the Category 2 LOCAs result from a complete ejection. The relatively small decrease between LOCA Categories 1 and 2 is a reflection of the rapid crack growth rates that are possible due to PWSCC. The Category 3 CRDM LOCAs are assumed in this analysis to result from simultaneous ejections of independent nozzles. Common cause CRDM ruptures are not considered in this base case analysis. However, the panel was asked to consider the likelihood of common cause failures in the elicitation. These base case Category 3 LOCA estimates were available for possible anchoring during this common cause consideration.

Periodic inspection over the next fifteen years is expected to decrease the LOCA frequencies by less than an order of magnitude (Figure 4.18). No further reduction in LOCA frequencies is predicted in subsequent inspections between 40 and 60 years of life. The implication is that all the cracking which previously initiated prior to the onset of periodic inspections will have been discovered by 40 years of service. After 40 years, steady state is reached and additional inspections are only likely to find new cracking indications.

Not surprisingly, the predicted BWR vessel LOCA frequencies (Table 4.4 and Figure 4.19) are much lower than the CRDM LOCA frequencies for LOCA Categories 1 and 2. This reflects the diminished concern of BWR vessel crack initiation and growth due to active degradation mechanisms, as well as the robustness of the vessel design and operation. However, the Category 3 base case BWR and PWR non-piping LOCA frequencies are actually similar. This difference is simply a function of the relative component sizes. A single PWR CRDM ejection is limited to a Category 2 LOCA while the combined BWR feedwater and vessel ruptures can contribute to all LOCA categories.

Feedwater nozzle failures provide the dominate contribution for the BWR LOCA Categories 1 – 4, but the nozzle size precludes contribution to larger category LOCAs (Table 4.4). Hence, the Category 5 and 6 LOCA frequencies are comprised solely by the beltline failure contribution. The beltline failure frequencies continue to increase with time due to the additional irradiation embrittlement while the feedwater LOCA frequencies are relatively insensitive to operating time. This time insensitivity stems from both the positive affect of mitigation procedures employed to address this degradation and the cutoff frequencies associated with the PFM analysis. More information is provided in Appendix I. These subcomponent trends explain why the total BWR vessel LOCA frequencies exhibit little time sensitivity for LOCA Categories 1 – 4 while the Category 5 and 6 LOCA frequencies continue to increase with operating time because they are governed by the beltline failures. It is interesting that after 60 years of operation, the Category 5 and 6 LOCA frequencies due to beltline failures are nearly identical to the Category 4 feedwater nozzle frequency.

It is also interesting to note (Table 4.4) that although the beltline LOCA frequencies are smaller, they decrease more gradually than the feedwater frequencies as a function of LOCA size. This trend reflects the expectation that the partial failure of the large, BWR vessel is not much more likely than a complete failure. This trend becomes even stronger after 60 years of service due to radiation embrittlement of the vessel material. The difference between a Category 1 and Category 6 LOCA after 60 years is approximately a factor of 5 (Table 4.4). Therefore, large and small LOCAs resulting for BWR vessel failures are almost equally likely after this service time. A similar trend is predicted in DH's PFM analysis for the BWR-1 piping base case frequencies for LOCA Categories 1 - 4 (Figure 4.5). LOCA decreases could be more gradual as both the component size increases and material brittleness increases. Similar trends are also discussed in the elicitation results (Section 6).

Table 4.4 BWR RPV LOCA Frequencies Broken Down by Major Subcomponent (Beltline and Feedwater Nozzles)

LOCA Category	LOCA Frequency (per calendar year)		
	25 Years	40 Years	60 Years
	RPV Beltline		
1	8.0E-9	2.4E-8	3.7E-8
2	1.9E-9	5.0E-9	2.3E-8
3	1.0E-9	2.5E-9	1.8E-8
4	4.0E-10	1.0E-9	1.4E-8
5	1.9E-10	4.5E-10	1.1E-8
6	7.9E-11	1.9E-10	8.0E-9
	Feedwater Nozzles		
1	8.0E-7	1.2E-6	1.0E-6
2	1.6E-7	2.4E-7	2.0E-7
3	3.2E-8	4.7E-8	4.00E-8
4	6.4E-9	9.4E-9	8.0E-9
	BWR Vessel – Combined		
1	8.1E-07	1.2E-06	1. 0E-06
2	1.6E-07	2.4E-07	2.2E-07
3	3.3E-08	5.0E-08	5.8E-08
4	6.8E-09	1.0E-08	2.2E-08
5	1.9E-10	4.5E-10	1.1E-08
6	7.9E-11	1.9E-10	8.0E-9

Table 4.5 PWR RPV (CRDM) LOCA Frequencies

LOCA Category	LOCA Frequencies (per calendar year)		
	25 Years	40 Years	60 Years
	RPV (CRDMs only)		
1	1.0E-03	2.2E-04	2.2E-04
2	2.0E-04	4.0E-05	4.0E-05
3	3.2E-08	1.6E-09	1.6E-09

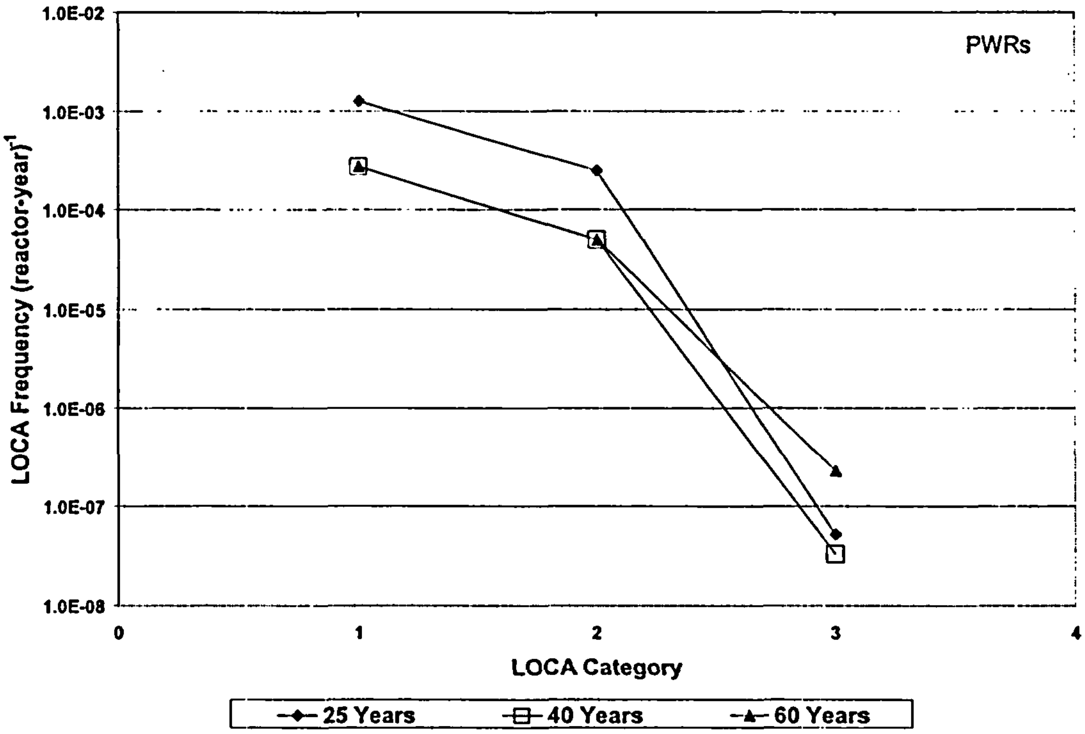

Figure 4.18 PWR CRDM LOCA Frequencies at 25, 40, and 60 Years

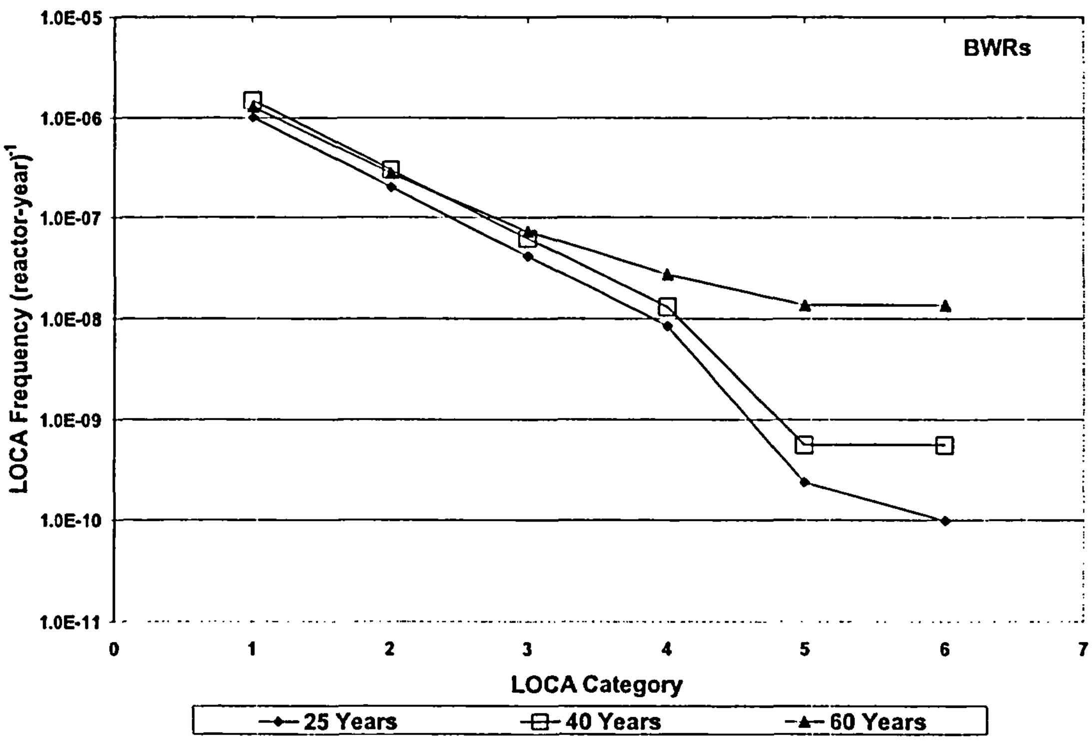

Figure 4.19 BWR Combined RPV LOCA Frequencies at 25, 40, and 60 Years

4.5 References

4.1 Poloski, J. P., et. al., "Rates of Initiating Event at U.S. Nuclear Power Plants: 1987-1995," NUREG/CR-5750, February 1999.

4.2 T. C. Chapman, et. al., "Assessment of Remedies for Degraded Piping," Electric Power Research Institute Report NP-5881-LD, 1988.

4.3 Simonen, F.A., Doctor, S.R., Schuster, G.J., and Heasler, P.G., "A Generalized Procedure for Generating Flaw-Related Inputs for FAVOR Code," NUREG/CR-6817, to be published (March 2004).

4.4 Memorandum from A.C. Thadani to S.J. Collins, Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Criteria in the PTS Rule (10 CFR 50.61), December 31, 2002.

5. ANALYSIS OF ELICITATION RESPONSES

The responses for each panel member were analyzed separately. Therefore, each panelist had unique LOCA frequencies for BWR piping, PWR piping, BWR non-piping, and PWR non-piping failures, assuming that input was provided. A unified response and analysis format was developed to ensure consistency and commonalty in processing the panelists' inputs. The panelist input values were assumed to represent the median, 95th, and 5th percentiles of their subjective uncertainty distributions for each elicitation response. The analysis structure is based on the assumption that all the responses correspond to percentiles of lognormal distributions. These distributions are then combined using a lognormal framework. The final output for each panelist is BWR and PWR-specific total passive LOCA frequency estimates of the mean, median, 5th and 95th percentiles. The LOCA frequencies for the individual panelists are then aggregated to obtain group LOCA frequency estimates. Statistical confidence bounds are calculated to provide a measure of panel diversity for the individual LOCA frequencies. A number of sensitivity analyses have been conducted to examine the robustness of this analysis procedure and to identify the assumptions and techniques that most significantly affect the final LOCA frequencies for each of the six LOCA categories. Details on the analysis of the panelists' elicitation responses follow.

5.1 Analysis Framework

As described in Sections 3.8, panel members were asked to supply three numbers for the adjustment ratios required for each elicitation question: a mid-value (MV), an upper bound (UB), and a lower bound (LB). As described in Section 3.3.2, these numbers were assumed to correspond to the median, 95th percentile, and 5th percentile, respectively, of the panelist's subjective uncertainty distribution. The goal is to estimate LOCA frequencies for four plant-type combinations (BWR piping, BWR non-piping, PWR piping, PWR non-piping) and combine these to obtain total BWR and PWR passive LOCA frequencies. These bottom-line estimates are all in the form of selected parameters of a LOCA frequency distribution. The parameters used are the mean, median, 95th percentile, and 5th percentile. These parameters are considered sufficient for regulatory applications.

There are two basic approaches to aggregating the individual panelist responses to obtain group estimates of the bottom-line parameters. The first approach is to propagate the responses separately for each panelist to obtain individual total LOCA frequency estimates. Then, the individual responses are combined to obtain group estimates of the bottom-line parameters. The second approach is to combine the responses for each question to obtain a group estimate for each elicitation question and then propagate these to estimate the total LOCA frequency parameters. In effect, the results of all the panel members are replaced by a quasi-panelist (in this case, a 13th panel member). Because the individual responses to each question varied a great deal, the second approach may lead to significant inconsistencies in the calculated responses of the quasi-panelist and consequent difficulties in interpreting and applying the bottom-line estimates. In contrast, because careful attention was paid in the individual elicitation sessions to eliminating inconsistencies, the first approach should yield self-consistent individual results. Consequently there are fewer difficulties in interpreting and applying the bottom-line estimates. Therefore, the first approach is used to aggregate the panelists' responses and estimate the bottom-line parameters.

A guiding principle in the analysis framework is to make as few assumptions as are necessary to estimate the four bottom-line LOCA frequency parameters. Although the approach adopted can be extended to estimate the entire LOCA frequency distribution, this is not done. Such an extension would necessitate fitting an entire distribution to just two parameters, e.g., a median and a 95th percentile. The approach adopted is to assume a lognormal structure and calculate the means and variances, which is all that is needed to propagate a panelist's responses to obtain estimates of the selected LOCA frequency parameters. To go beyond this would necessitate assuming more structure to the panelists' responses than is necessary or warranted.

It is important that the bottom-line LOCA frequency estimates reflect both individual uncertainty and panel diversity. Individual uncertainty stems from the uncertainties in each panel member's responses, as embodied in the upper and lower bound estimates for each elicited quantity. These uncertainties are propagated to obtain the means, as well as the 5th and 95th percentiles of the LOCA frequency estimates. Panel diversity refers to the different bottom-line estimates from the various panel members. Because of the lack of data and the variety of approaches used by individual panel members, it is to be expected that there will be large differences in their responses and the resultant bottom-line estimates. In this sense, panel diversity is simply a reflection of the existing scientific uncertainty about the LOCA frequencies being estimated. Confidence intervals for the estimated bottom-line parameters are developed to reflect panel diversity.

A common analysis framework was adopted to reflect the principles discussed above (Figure 5.1) once the structure and format of all the elicitation responses was developed. The framework implements the response analysis called for in the elicitation process (Blocks 13 and 16 in Figure 3.1) and is consistent with the elicitation questions (Section 3.8) developed to assess fundamental piping and non-piping LOCA contributing factors. A common framework is important so that a single calculation methodology can be used to process the elicitation results. Also, the common framework allows for results to be compared across the experts and allows meaningful measures of panel diversity to be developed.

The analysis steps illustrated in Figure 5.1 are summarized as follows. The piping contribution is outlined in the left-hand branch of Figure 5.1. The individual responses (Section 5.2) for the anchoring frequencies (Block 1.2) and adjustment ratios (Block 1.3) are assumed to be independent split lognormal distributions and are multiplied as described in Section 5.3.3 to determine the piping system frequencies (Block 1.4). These system frequencies are then summed for all contributing piping systems to obtain the un-normalized piping contribution (Block 1.5) as described in Section 5.3.4. The un-normalized piping contribution is then multiplied (Section 5.3.3) by the piping contribution factor (Block 1.6) to determine the piping contribution (Block 1.7).

The non-piping calculations (right-hand side of Figure 5.1) are analogous except there is an additional analysis step required to determine the non-piping component from the subcomponent frequencies. The individual responses for the anchoring frequencies (Block 1.9) and adjustment ratios (Block 1.10) are assumed to be independent split lognormal distributions and multiplied (Section 5.3.3) to determine the non-piping subcomponent frequencies (Block 1.11). These subcomponent frequencies are then summed (Section 5.3.4) for all contributing non-piping subcomponents to obtain the un-normalized non-piping component frequencies (Block 1.12). These frequencies are then multiplied (Section 5.3.3) by the component contribution factor (Block 1.13) to determine the non-piping component frequencies (Block 1.14). The non-piping component frequencies are then summed (Section 5.3.4) to calculate the un-normalized non-piping contribution (Block 1.15) which is then multiplied (Section 5.3.3) by the non-piping contribution factor (Block 1.16). The end result is the non-piping contribution (Block 1.17).

The piping and non-piping contributions are summed (Section 5.3.4) to develop individual total LOCA frequency estimates (Section 5.3.5). These individual estimates are then combined (Section 5.4) to develop the group estimates and associated confidence intervals. More details on each specific aspect of this analysis structure (Figure 5.1) and the supporting calculation methodology are subsequently discussed.

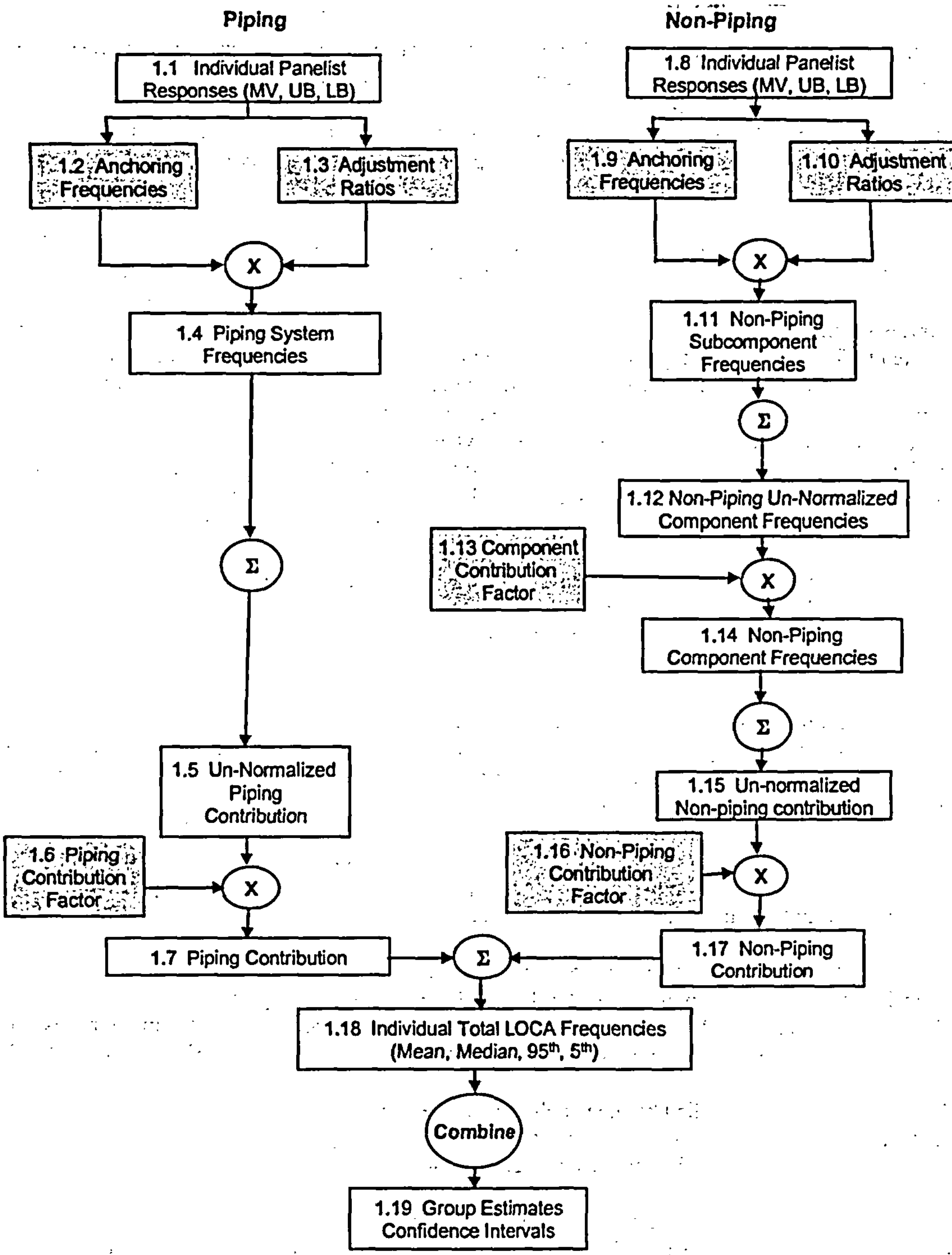

Figure 5.1 Flowchart of the Analysis Framework

5.2 Input Responses

A common response format was provided to each panelist for recording their elicitation responses prior to the individual elicitations. However, panelists were urged to provide responses in a format consistent with their approach. While several panelists utilized the common format; many panelists did not. In general, the elicitation responses and associated formats varied significantly among the panelists. The input responses required from the panelists are indicated by the shaded blocks in Figure 5.1.

All anchoring frequencies and adjustment ratios had associated mid-values, or medians, as well as bounds to represent 90% coverage intervals. The mid-values and coverage intervals for the adjustment ratios were obtained directly from the elicitation responses. The sources for the coverage intervals for the anchoring frequencies varied. A factor of three was generally used as a measure of uncertainty for all LOCA categories when a panelist chose the base case results for anchoring frequencies. This uncertainty factor is the ratio between the coverage bounds (upper and lower) and the mid-values for the piping and non-piping base case frequency estimates (Sections 3.5 and 4). The factor of three was based on a consideration of service history uncertainties, but panelists could adjust this factor if desired when reviewing their elicitation responses (Section 3.10). It was generally not necessary to develop coverage intervals associated with the anchoring frequencies for panelists who chose their own system-based frequencies or who chose alternative anchoring frequencies. The uncertainties for these estimates were already reflected in the coverage intervals for the adjustment ratio. In practice, the distinction between uncertainties in the anchoring frequencies and adjustment ratios is somewhat artificial since these uncertainties are combined as discussed in Section 5.3. However, it is still necessary to understand the source of the coverage interval input for the anchoring frequencies.

Anchoring frequencies were initially selected by each panelist (Blocks 1.2 and 1.9 in Figure 5.1) for anchoring subsequent responses. Often, these anchoring frequencies and conditions were adopted from the piping and non-piping base cases (Sections 3.5 and 4). The base case anchoring frequencies utilized in the common analysis framework pertain to each LOCA size category and are associated with 25 years of plant operation for the defined variable conditions as discussed in Section 3.5. However, several panelists chose not to utilize the base cases for anchoring. Some panelists developed absolute LOCA frequencies for each piping system/non-piping subcomponent, and some developed alternative anchoring conditions and frequencies.

Next, after choosing appropriate anchoring frequencies, each panelist chose adjustment ratios to relate the anchoring frequencies to the associated LOCA frequencies for that piping system/non-piping subcomponent after 25, 40, and 60 years of plant operation. A unique set of adjustment ratios was chosen for each time period and possibly also for each LOCA size category (Blocks 1.3 and 1.10).

The only other inputs supplied by the panelists were the contribution factors (Blocks 1.6, 1.13, and 1.16). The panelists only provided information on the largest piping system or non-piping subcomponent LOCA contributing factors (Section 3.4), and not all possible contributing factors. The elicitation questions (Section 3.8) specifically required each panelist to only quantify the contributors that make up at least 80% of the piping and non-piping LOCA frequencies. The contribution factors are used to normalize each panelist's input to ensure that the results represent the total LOCA frequency contributions. The piping contribution factor (Block 1.6) is the reciprocal of each panelist's mid-value estimate of the total contribution of the piping systems quantified. The panelist also was asked for a coverage interval for the total piping system contribution. If the panelist assessed all piping systems, the mid-value contribution factor is 1 (or 100%), as are the upper and lower bounds. As an additional example, assume that a panelist believes he has assessed piping systems that provide 85% of the total piping LOCA contribution with a LB and UB of 75% to 95% for this estimate. The piping contribution factor is then 1.18 (1/0.85) with a LB of 1.05 (1/0.95) and a UB of 1.33 (1/0.75). Obviously, the LB contribution factor cannot be less than 1 to reflect the opinion that 100% of the contributing factors have been assessed.

Panelists similarly provided one of the two non-piping adjustment factors (Blocks 1.13 and 1.16) during their elicitation. The analysis method chosen by each panelist determined which non-piping adjustment factor was required. The two possible analysis methods are mutually exclusive. Therefore, the adjustment factor associated with the approach not selected is unity by definition. If the panelist assessed each non-piping component separately, then the component contribution adjustment (Block 1.13) was required for each non-piping component (i.e., pumps, valves, vessel, etc.). The component contribution adjustment represents the total contribution (> 80%) of all quantified non-piping subcomponents for that component. For example, if the panelist was assessing RPV LOCA contributions and believed that vessel rupture contributes to 90% of the frequency associated with RPV LOCAs, then the contribution factor is 1.11 (1/0.9). Similar contribution factors would be determined for the remaining non-piping components (i.e., steam generator, pressurizer, pumps, valves) based on the subcomponents assessed. The panelist also would supply the coverage interval associated with each component contribution adjustment.

The non-piping contribution factor (Block 1.16) was required for panelists that considered specific failure scenarios without regard to the applicable non-piping component. Here the factor represents the contribution of the unique failure scenarios that were quantified to the total non-piping LOCA frequency. For example, for Category 2 PWR LOCAs, a panelist may believe that steam generator tube ruptures, CRDM ejections and pressurizer heater sleeve failures make up 95% of the total non-piping LOCA frequency. The non-piping contribution factor is then 1.05 (1/0.95). The panelist also would supply the coverage interval associated with each non-piping contribution factor.

As discussed in Section 3.10, it was occasionally necessary for the facilitation team to impute missing input response data so that complete LOCA frequency results could be developed for each panelist. Usually, any missing information pertained to coverage intervals associated with some mid-value responses. Missing information was imputed in a manner consistent with the rest of the panelist's responses. Panelists were required to either verify imputed data or directly provide this data.

A spreadsheet was developed for each panelist containing the input responses for each piping system and non-piping subcomponent LOCA frequency contribution. The spreadsheet structure uses the common response format described earlier and allows processing using the analysis framework (Section 5.1) with the calculation formulae (Section 5.3) to determine individual total LOCA frequency estimates. The spreadsheets are all identical to ensure consistent processing and foster comparisons of the panelists' responses.

There is one situation where this framework does not necessarily support the panelist's opinion. Panelist K provided global LOCA piping estimates and uncertainties for PWR plants based on plant specific studies associated with the development of risk-informed ISI methodology procedures. More information is provided in Appendix K. However, to utilize the response and analysis frameworks, this panelist's piping responses were apportioned among the LOCA sensitive systems based on the percentage of welds in each system. Panelist K provided no indication about the validity of this assumption. While the final LOCA frequency estimates and uncertainties matched panelist K's elicitation input, this analysis should not be construed to reflect panelist K's beliefs about the individual piping system LOCA frequencies. However, panelist K's non-piping responses and beliefs are consistent with the existing framework. No other deviation from the analysis framework was required for any other panelist's responses.

5.3 Analyzing Individual Elicitation Responses

For each panel member who provided sufficient information, the responses (MVs, UBs, and LBs) are propagated to obtain the bottom-line estimates, i.e., the means, medians, 95^{th} and 5^{th} percentiles of the LOCA frequency distributions of the four plant-type combinations (BWR piping, BWR non-piping, PWR piping, PWR non-piping). The BWR piping and non-piping estimates and the PWR piping and non-piping estimates are then summed to obtain individual estimates of the total BWR passive LOCA frequencies and the total PWR passive LOCA frequencies. This section describes the details of the analysis framework (Figure 5.1) that has been previously discussed. Figure 5.2 is a flow chart outlining the assumptions and calculations associated with important aspects of this framework. The philosophy, assumptions, and equations associated with each specific aspect of the analysis are subsequently provided. As flowchart block numbers (Figures 5.1 and 5.2) are referenced in this discussion, the first number also indicates the associated figure containing that block; i.e., Block 2.3 refers to Figure 5.2.

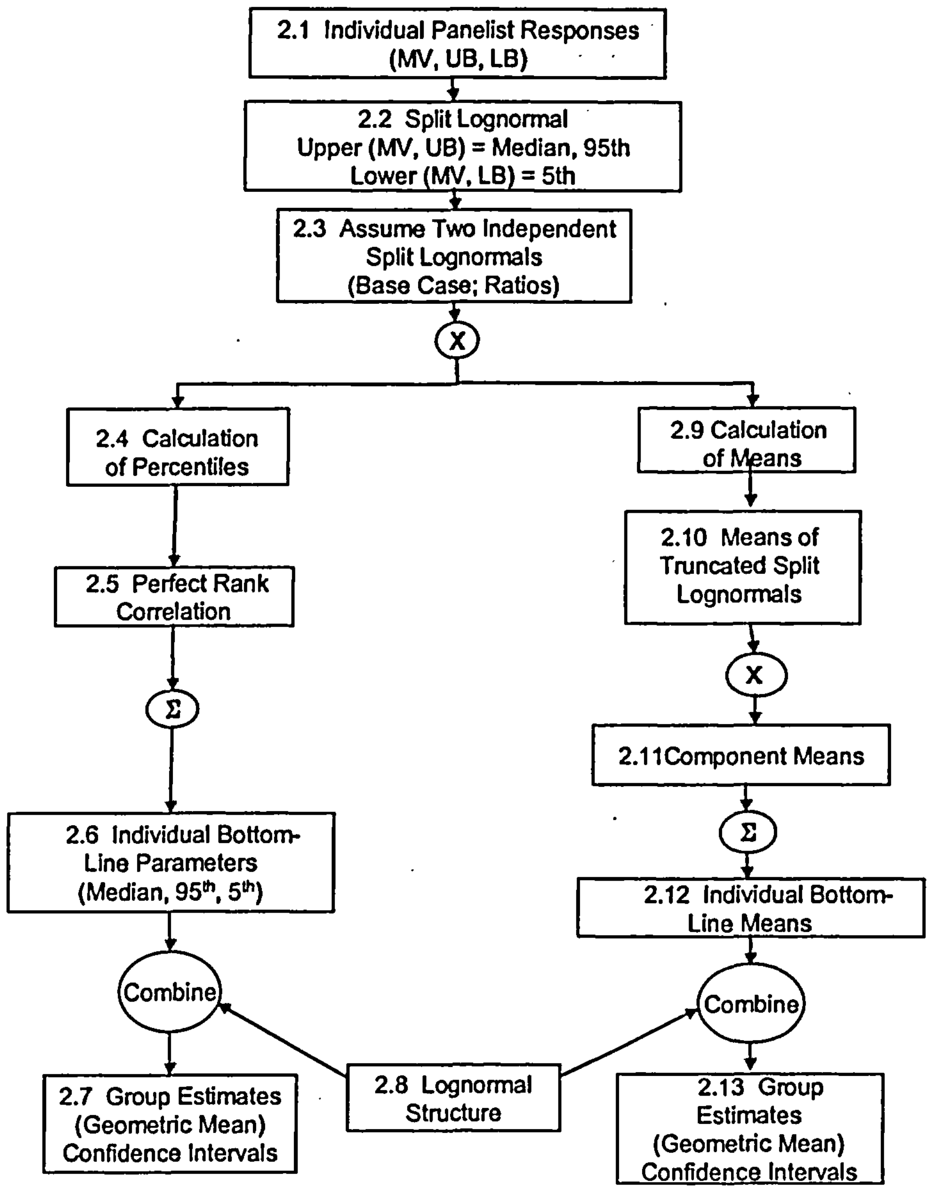

Figure 5.2 Flow Chart of Required Calculations

5.3.1 Lognormal Structure

In order to propagate the responses, it is necessary to specify a probabilistic structure for them. The assumed probabilistic structure is based on the lognormal distribution. This structure is chosen to reflect the ratio structure of the elicitation questions and the panelist responses. Accordingly, for each question, the piping (Blocks 1.1 and 2.1) and non-piping (Blocks 1.8 and 2.1) responses are assumed to be the corresponding percentiles of a lognormal distribution. That is, the MV is assumed to be the median, the UB is assumed to be the 95th percentile, and the LB is assumed to be the 5th percentile of a lognormal distribution (Block 2.2).

A complication arises if the UB and LB are not multiplicatively symmetric about the MV, as often happens. In such cases, there is no unique lognormal distribution determined by the given MV, UB and LB. Instead, the assumed probabilistic structure is a split lognormal distribution (Block 2.2). Because a lognormal distribution is determined by a median and one other percentile, two different lognormal distributions can be determined: one by the MV and UB and another by the MV and LB. For values less than the median, the split lognormal is the lower half, i.e., the part less than its median, of the lognormal distribution determined by the MV and LB. For values greater than the median, the split lognormal is the upper half, i.e., the part greater than its median, of the lognormal distribution determined by the MV and UB. Clearly, if the UB and LB are symmetric about the MV, then the split lognormal is identical to the unique lognormal distribution determined by the MV, UB, and LB. Accordingly, the split lognormal structure is assumed for both the symmetric and non-symmetric cases.

To calculate the means, the split lognormals in Block 2.2 in Figure 5.2 are replaced by split lognormals truncated at the 99.9th percentile. The remaining 0.1 percent of each distribution's area is concentrated at the 99.9th percentile. This truncation point was chosen to be reasonably conservative, yet ensure that the mean is not dominated by the extreme upper tail of the distribution. Furthermore, because the panel members provided no information beyond the 95th percentile, any truncation point beyond the 95th percentile is consistent with their responses (see Section 5.3.4.1).

For propagating the distributions to calculate the bottom-line percentiles (median, 5th and 95th), each split lognormal distribution is replaced by two full lognormal distributions. One lognormal corresponds to the upper half of the split lognormal, above the median, and the other corresponds to the lower half. The lower distributions are used only to calculate the 5th percentiles and the upper distributions are used to calculate the medians and 95th percentiles. Although this algorithm for propagating the split lognormals is not exactly equivalent to propagating the split lognormals as two-part distributions, the results are a good approximation to those obtained through exact propagation. The algorithm is used for computational convenience, and also because the percentile of greatest interest, namely, the 95th, is determined largely by the upper parts of the split lognormals (see Section 5.3.4.2).

The LOCA frequencies for the four plant-type combinations are estimated by summing the LOCA frequencies for the relevant non-piping subcomponents and components, and piping systems. Each individual component contribution to the LOCA frequency is a product of two distributions: (i) the anchoring frequencies (Blocks 1.2 and 1.9) and (ii) the adjustment ratios (Blocks 1.3 and 1.10) which account for differences between the base case conditions and the assessed component as a function of LOCA size and operating time. Each panel member's bottom-line LOCA frequency estimate for the four plant-type combinations is a normalized sum of these individual products over all piping systems (Block 1.7) and non-piping components (Block 1.17). Thus, the bottom-line LOCA frequencies are sums of products of assumed lognormal distributions which represent the panelist's responses.

5.3.2 Basic Lognormal Formulas

The distribution of Y is defined to be lognormal when the distribution of $\ln(Y)$ is normal. The distribution is denoted by $LN(\mu, \sigma)$, where $\ln(Y)$ is normal with mean μ and variance σ^2. Denote the median of Y by $m(Y)$, its expected value by $E(Y)$, its variance by $V(Y)$, its standard deviation by $SD(Y)$ and its p^{th} percentile by $b_p(Y)$. These lognormal parameters can be written as follows [5.1].

$$m(Y) = \exp(\mu) \tag{5.1}$$

$$E(Y) = \exp(\mu + \sigma^2/2) \tag{5.2}$$

$$V(Y) = \exp(2\mu + \sigma^2) \cdot [\exp(\sigma^2) - 1] \tag{5.3}$$

$$SD(Y) = \exp(\mu + \sigma^2/2) \cdot [\exp(\sigma^2) - 1]^{1/2} \tag{5.4}$$

$$b_p(Y) = \exp(\mu + k_p\sigma) \tag{5.5}$$

where k_p is the p^{th} percentile of the standard normal distribution ($\mu = 0$ and $\sigma^2 = 1$).

Some other useful relations follow.

$$E(Y) = m(Y)\exp(\sigma^2/2) \tag{5.6}$$

$$V(Y) = [E(Y)]^4/[m(Y)]^2 - [E(Y)]^2 \tag{5.7}$$

$$m(Y) = E(Y)\,[1 + V(Y)/[E(Y)]^2\,]^{-1/2} \tag{5.8}$$

Solving Equation 5.6 for σ yields

$$\sigma = [2\ln\{E(Y)/m(Y)\}]^{1/2} \tag{5.9}$$

A commonly used parameter is the error factor (EF), defined as the ratio of the 95[th] percentile to the median. Because the 5[th] and 95[th] percentiles of a lognormal are multiplicatively symmetric about the median, the error factor is also equal to the ratio of the median to the 5[th] percentile. Setting k_{95} = 1.645, it follows from Equations 5.1 and 5.5 that

$$EF(Y) = b_{95}(Y)/m(Y) = m(Y)/b_5(Y) = \exp(1.645\sigma) \tag{5.10}$$

5.3.3 Product of Independent Lognormals

The LOCA frequency for each of the four plant-type combinations is a sum of contributions from individual piping systems or non-piping subcomponents. Each of these frequencies is a product of an anchoring frequency (Blocks 1.2 and 1.9) and an adjustment ratio (Blocks 1.3 and 1.10). As indicated in Block 2.2 and as indicated in Block 2.2 and described in Section 5.3.1, the anchoring frequencies and adjustment ratios are each modeled by a split lognormal distribution. The required percentiles of the products are calculated by the following algorithm. The 5[th] percentile of each individual piping or non-piping component is calculated by multiplying the lognormals associated with the lower halves of the anchoring frequency and adjustment ratio distributions. Similarly, the 95[th] percentile is calculated by multiplying the lognormals associated with the upper halves of these distributions. The median of each individual piping or non-piping component is determined by multiplying the medians of the two distributions.

The statistical characteristics of the anchoring frequency and adjustment ratio distributions depend on the characteristics of the uncertainty bounds provided by the panelists. Because they are generally provided by different panel members, and because they have different structures (one is an absolute number and the other is a ratio), the corresponding uncertainty bounds are assumed to be statistically independent. Accordingly, the piping system or non-piping component frequency distributions are assumed to be the product of two independent split lognormal distributions (Block 2.3). According to the algorithm discussed above, the required percentiles of the piping or non-piping distributions are therefore the percentiles of the product of two independent lognormal distributions.

Denote the two independent lognormals corresponding to the upper or lower halves of the corresponding anchoring frequency and adjustment ratio distributions by Z_1 and Z_2, respectively, where $Z_i = LN(\mu_i, \sigma_i)$ for i = 1, 2. Recalling that the mid-value, upper bound, and lower bound supplied by a panelist as a response to each question are assumed to correspond to the median, 95th, and 5th percentiles, respectively, of a lognormal distribution. Therefore, and denoting the panelist responses corresponding to Z_i by MV_i, UB_i and LB_i and the median of Z_i by m_i, then $m_i = MV_i$ and, from Equation 5.1,

$$\mu_i = \ln(m_i) = \ln(MV_i),\ i = 1, 2. \tag{5.11}$$

It is convenient to convert the uncertainty bounds supplied by the panelists into error factors of the corresponding lognormal distributions. Denote the error factor of Z_i by EF_i. Then, for Z_i corresponding to the upper part of a split lognormal, $EF_i = UB_i / MV_i$ and, from Equation 5.10,

$$\sigma_i = \ln(EF_i)/1.645 = \ln(UB_i / MV_i)/1.645,\ i = 1,2 \tag{5.12}$$

Let $Z = Z_1 Z_2$ be the product of the two independent lognormals where the variable Z represents either the piping system (Block 1.4) or non-piping subcomponent (Block 1.11) frequencies. Because the sum of any number of normal distributions is always normal, the product of any number of lognormals is always a lognormal (Blocks 2.4 and 2.9). Hence, $Z = LN(\mu, \sigma)$ is lognormal. Because multiplying two lognormals is equivalent to adding their underlying normal distributions, it follows from the independence of Z_1 and Z_2 that $\mu = \mu_1 + \mu_2$ and $\sigma = [(\sigma_1)^2 + (\sigma_2)^2]^{1/2}$. In terms of the panelist input response parameters, it follows from Equations 5.11 and 5.12 that

$$\mu = \ln(MV_1) + \ln(MV_2) = \ln[(MV_1)(MV_2)] \tag{5.13}$$

$$\sigma = [(\ln(UB_1 / MV_1))^2 + (\ln(UB_2 / MV_2))^2]^{1/2} / 1.645 \tag{5.14}$$

From Equations 5.2 and 5.3, the mean and variance of Z are given by

$$E(Z) = (MV_1)(MV_2)\exp[\{(\ln(UB_1 / MV_1))^2 + (\ln(UB_2 / MV_2))^2\}/5.412] \tag{5.15}$$

$$V(Z) = [E(Z)]^2\{\exp[\{(\ln(UB_1 / MV_1))^2 + (\ln(UB_2 / MV_2))^2\}/2.706] - 1\} \tag{5.16}$$

The formulas in Equations 5.12–5.16 apply when Z corresponds to the upper parts of the split lognormals. When Z corresponds to the lower parts, the error factor of Z_i is defined by $EF_i = MV_i / LB_i$ and the formulas in Equations 5.12-5.16 apply with UB_i / MV_i replaced by MV_i / LB_i for i = 1, 2.

This analysis to calculate the product of independent lognormals was also applied at other similar junctures of the analysis process (Figure 5.1). Let Z represent either the piping (Block 1.7) or non-piping (Block 1.17) contribution, respectively. Then Z_1 represents the non-normalized piping (Block 1.5) or non-piping (Block 1.15) contribution factor, respectively, while Z_2 represents the piping (Block 1.6) or non-piping (Block 1.16) contribution factor, respectively. The non-piping analysis requires an additional product calculation to determine the non-piping component frequencies (Block 1.14). If Z represents the non-piping component frequency, then Z_1 and Z_2 represent non-normalized non-piping frequencies (Block 1.12) and component contribution factors (Block 1.13), respectively. The assumption that the various contribution factor distributions are lognormal is at best an approximation because the LB factor cannot be less than 1. However, the probability of the assumed lognormal being less than 1 is always less than 5%, so the approximation is reasonable.

5.3.4 Sum of Distributions

This section describes the methodology used to calculate the bottom-line parameters for the four plant-type LOCA frequency combinations (i.e. BWR piping, BWR non-piping, PWR piping and PWR non-piping). The contributions from any single piping system (Block 1.4) or non-piping subcomponent (Block 1.11) are summed to obtain the non-normalized LOCA frequency associated with piping (Block 1.5) and non-piping subcomponent (Block 1.12) failures.

Let any of these summed distributional quantities be denoted by X. Let $\{Z_j\}$ denote the j^{th} contribution to X. Then

$$X = \sum Z_j \tag{5.17}$$

With the exception of the mean, all other parameters of X depend on the correlation structure of the Z_j. The calculation of the mean will be discussed in Section 5.3.4.1 and the calculation of the variance and percentiles will be discussed in Section 5.3.4.2.

5.3.4.1 Calculation of the Mean - In general, the mean of X is given by

$$E(X) = \sum E(Z_j) \tag{5.18}$$

Equation 5.18 holds under all circumstances, whether the Z_j are independent or not.

The Z_j correspond to Blocks 1.4 and 1.11 in Figure 5.1. From the flowchart, each Z_j is a product of an anchoring frequency (Blocks 1.2 and 1.9) and an adjustment ratio (Blocks 1.3 and 1.10). To calculate the means, the split lognormals assumed for each of these blocks are replaced by split lognormals that are truncated at the 99.9th percentile. The remaining 0.1 percent of each distribution's area is concentrated at the 99.9th percentile. This truncation point was chosen to be reasonably conservative to ensure that the mean is not dominated by the extreme upper tail of the distribution. Furthermore, because the panel members provided no information beyond the 95th percentile, any truncation point beyond the 95th percentile is consistent with their responses.

In general, for truncation at the p^{th} percentile, the truncation point is given by Equation 5.5. Using the notation of Section 5.3.2, the mean of a truncated split lognormal is given by

$$E(w) = m(w)\{\exp\left(\frac{\sigma_L^2}{2}\right)\Phi(-\sigma_L) + \exp\left(\frac{\sigma_U^2}{2}\right)\left(\Phi(\sigma_U) - \Phi(\sigma_U - t_p)\right) + \exp(t_p\sigma_U)(1 - \Phi(t_p))\} \tag{5.19}$$

where

m(w) = median of w
σ_L^2 = variance of lower half of ln(w)
σ_U^2 = variance of upper half of ln(w)
Φ = cumulative distribution function of the standard normal distribution
t_p = truncation point of the standard normal at baseline p

The means of the truncated split lognormals corresponding to the anchoring frequencies and adjustment ratios are calculated using Equation 5.19 for $p = 99.9$ and $k_p = 3.09$. Because of the structure of the elicitation process, the panelist responses for the anchoring frequencies and adjustment ratios may be assumed to be independent. Accordingly, the mean of each Z_j is the product of the means of its two factors. Write

$$Z_j = (U_j)(V_j) \tag{5.20}$$

where U_j is a truncated split lognormal corresponding to an anchoring frequency and V_j is a truncated split lognormal corresponding to an adjustment ratio. Based on the assumed independence of U_j and V_j,

$$E(Z_j) = E(U_j)\,E(V_j) \tag{5.21}$$

The means of the LOCA frequencies corresponding to Blocks 1.5 and 1.12 are calculated by substituting Equation 5.21 into Equation 5.18 to obtain

$$E(X) = \sum E(U_j)\,E(V_j) \tag{5.22}$$

5.3.4.2 Calculation of the Variance and Percentiles - To calculate any other parameters of X (e.g., variance or 95th percentile), it is necessary to specify the correlation structure of its individual Z_j components. These components express the uncertainties of the panelists about their responses. The correlation structure is determined by the correlations between all pairs of the individual Z_j distributions. The exact correlation structure is unknown. In this study, it is assumed that the Z_j are positively correlated. This means that if one Z_j is large, then the other Z_j's will tend to be large. This is an appropriate assumption because the uncertainties in the LOCA frequency contribution factors for each panelist were positively correlated. In other words, if the uncertainty in the response for any component was high, then the uncertainties in the responses for the other components also tended to be high. For example, most panelists noted that degradation mechanisms that significantly contribute to the LOCA frequencies affect many systems similarly. If conditions lead to higher failure probabilities in one piping system due to a specific degradation mechanism, then the failure probabilities of other susceptible systems will also increase and this positive correlation will extend to the uncertainties.

For positively correlated Z_j, the correlation structure is bounded by the independent case (all pair-wise correlations are zero) and the perfect correlation case (all pair-wise correlations are one). The lower bound on the variance is attained when the components are independent. For this case, the variance of the sum in Equation 5.17 is the sum of the variances, and the lower bound on the variance of X is

$$V_L(X) = \sum V(Z_j), \tag{5.23}$$

where $V(Z_j)$ is given by Equation 5.16.

In general, let ρ_{jk} be the correlation coefficient between Z_j and Z_k and let $SD_j = [V(Z_j)]^{1/2}$ be the standard deviation of Z_j. The variance of X is given by

$$V(X) = \sum V(Z_j) + \sum_{j \neq k} \rho_{jk}\,(SD_j)\,(SD_k) \tag{5.24}$$

Hence the upper bound on the variance of X is given by Equation 5.24 with $\rho_{jk} = 1$ for all $j \neq k$ and is equal to

$$V_U(X) = \sum V(Z_j) + \sum_{j \neq k} (SD_j)(SD_k) = [\sum (SD_j)]^2, \tag{5.25}$$

where SD_j is given by $[V(Z_j)]^{1/2}$ from Equation 5.16.

Note that the upper bound on the variance may not be achievable, because it may not be possible for the correlation coefficients for all pairs (Z_j, Z_k) to simultaneously be equal to 1. For this to happen, each Z_j and Z_k would have to be linearly related. Consequently, each Z_j would have to be a linear function of Z_1. Because the Z_j are assumed to be lognormal, this implies that the variances of the underlying normals of the Z_j would all be equal. Because the elicitation responses did not even come close to satisfying this requirement, the assumption of perfect correlation is replaced by an assumption of perfect *rank* correlation. Instead of being equal to one, the correlation between all pairs (Z_j, Z_k) is assumed to be maximal. This can in fact be achieved by having Z_j and Z_k functionally related, i.e., $Z_j = F(Z_k)$ for some monotonically increasing function F.

To see why this is so, let Z_i be a lognormal random variable (RV) with parameters μ_i and σ_i, i.e., $\ln(Z_i)$ is a normal RV with mean μ_i and standard deviation σ_i. Thus, $Z_i = \exp(\mu_i + \sigma_i W_i)$ where W_i is a standard normal RV, with mean $\mu_i = 0$ and standard deviation $\sigma_i = 1$. Then $Z_1 = \exp(\mu_1 + \sigma_1 W_1)$ and $Z_2 = \exp(\mu_2 + \sigma_2 W_2)$ will have maximal correlation if $W_1 = W_2 = W$, i.e., if the same standard normal RV is used in the expressions for Z_1 and Z_2. Therefore, the sum $X = Z_1 + Z_2$ is a monotonic function of W and the percentiles of X correspond directly to the percentiles of W. It follows that $b_p(X) = \exp[\mu_1 + \square_1 b_p(W)] + \exp[\mu_2 + \square_2 b_p(W)] = b_p(Z_1) + b_p(Z_2)$, where $b_p(X)$ is the pth percentile of the RV (X). In particular, median(X) = median(Z_1) + median(Z_2), and the 95th percentile of the sum equals the sum of the 95th percentiles. The assumption of perfect rank correlation generalizes to the sum of any number of components by setting $Z_j = \exp(\mu_j + \sigma_j W)$.

The baseline estimates of the percentile parameters of the LOCA frequencies (Block 2.6) are calculated assuming perfect rank correlation. The actual correlation structure is expected to be much closer to perfect rank correlation than independence. Degradation mechanisms and other LOCA contributing factors can affect many piping systems and non-piping subcomponents similarly. Hence, there is an expectation that a strong correlation exists and this is evident in many of the individual panelist's responses.

5.3.5 Total LOCA Frequencies

The analysis framework described above is used with each panel member's responses to calculate the estimated bottom-line parameters of the LOCA frequencies for BWR and PWR piping and non-piping systems. The total BWR and PWR LOCA frequencies (Block 1.18) are calculated by summing the lognormals corresponding to the respective piping and non-piping contributions. The estimated bottom-line parameters (Block 2.6) of the total LOCA frequencies are then calculated as described in Section 5.3.4. The means are calculated using Equation 5.18 and the percentiles are calculated assuming perfect rank correlation. Clearly, the total LOCA frequencies are calculated only for those individual panelists who have supplied sufficient quantitative input response information.

5.4 Group Estimates and Confidence Intervals

Section 5.3 describes how to estimate the bottom-line parameters (means, medians, 95^{th}, and 5^{th} percentiles) of total passive LOCA frequencies based on the responses of any one panelist. The final step in the analysis framework is to combine the individual estimates to obtain group estimates for the total passive BWR and PWR LOCA frequencies (Blocks 1.18 and 2.7). Once the individual bottom-line parameters have been calculated, each parameter type is combined independently of any other parameter type. In other words, the group mean is a function only of the individual means, the group median is a function only of the individual medians, and so on. Even though the individual parameter estimates are constrained by the lognormal relations, the group parameter estimates are not. This is consistent with the guiding principle discussed in Section 5.1 of making as few assumptions as necessary. Because estimated values of the four bottom-line parameters are sufficient for regulatory applications, any additional constraints on them are superfluous. In particular, no assumption is made about the form of the total LOCA frequency distribution, e.g., that it is lognormal.

Let $\{u_k\}$, $k = 1, 2, \ldots, n$, where "n" is the number of panelists, denote a set of individual estimates of any one bottom-line parameter. For example, $\{u_k\}$ could be the set of current day estimates of the mean total frequency for LOCA Category 1 for BWRs, as calculated from the panelists' responses. A basic premise of any expert elicitation process is that the panel responses as a whole have no significant systematic bias. While individual responses can be highly uncertain and differ drastically, they do not systematically over- or underestimate the quantities of interest. Therefore, based on the u_k values only, we wish to calculate an unbiased group estimate of the parameter in question and also obtain a measure of panel diversity about the group estimate. The baseline approach (Section 5.5) used here simultaneously provides a group estimate and a measure of panel diversity. The baseline approach is based on the observation that $\{u_k\}$ typically spans several orders of magnitude (see Section 7.5) and is best described by a highly skewed distribution. Accordingly, it is assumed that the $\{u_k\}$ values are observations from a lognormal distribution, U. The assumption of this lognormal structure (Block 2.8) is the only one required to calculate group LOCA frequency estimates and confidence intervals (Blocks 1.19 and 2.7) by combining the individual estimates (Blocks 1.18 and 2.6) as follows.

It is convenient to write U in the form

$$U = g \exp(W) \tag{5.26}$$

where W has a normal distribution with mean 0 and variance σ^2. The interpretation of U is that the true value of the bottom-line parameter estimated by the u_k is g and that $\exp(W)$ is a bias factor which describes the differences between the panelists' estimates and the true value of the parameter. This structure for W assumes no systematic bias in the panelists' estimates of the true value U. Therefore g is the median of U and $\ln(U)$ has a normal distribution with mean $\ln(g)$ and variance σ^2.

Define,

$$w_k = \ln(u_k) \quad \text{for } k = 1, 2, \ldots, n. \tag{5.27}$$

From Equation 5.26, the standard estimate for $\ln(g)$ is the sample mean of the w_k individual estimates for the bottom-line parameter of interest. Hence, the group estimate of g is the geometric mean of the u_k. Denoting the group estimate of g by g^*, it follows that

$$g^* = [\Pi\, u_k]^{1/n} \tag{5.28}$$

A measure of panel diversity is a confidence interval for g^*. A two-sided $100(1\text{-}\alpha)\%$ confidence interval (where $1\text{-}\alpha$ is the desired confidence level) for $\ln(g)$ is given by

$$\ln(g^*) - t_{1-\alpha/2}(n\text{-}1)[S/n^{1/2}] \leq \ln(g) \leq \ln(g^*) + t_{1-\alpha/2}(n\text{-}1)[S/n^{1/2}] \tag{5.29}$$

where $t_{1-\alpha/2}(n\text{-}1)$ is the $100(1\text{-}\alpha/2)$ percentile of a Student's t distribution with n-1 degrees of freedom, and

$$S^2 = \sum [w_k - \ln(g^*)]^2 / (n\text{-}1) \tag{5.30}$$

is the sample variance of the w_k. The corresponding $100(1\text{-}\alpha)\%$ confidence interval for g is obtained from Equation 5.29 as.

$$g^* \exp\{- t_{1-\alpha/2}(n\text{-}1)[S/n^{1/2}]\} \leq g \leq g^* \exp\{ t_{1-\alpha/2}(n\text{-}1)[S/n^{1/2}]\} \tag{5.31}$$

For the baseline analysis, α is chosen as 0.05 to calculate a 95% confidence interval for g.

5.5 Baseline LOCA Frequency Calculations

A baseline analysis was conducted to process the panelists' elicitation responses (Section 5.2). The purposes of this analysis were to develop LOCA frequency estimates for constructing the final estimates used for regulatory analysis and to provide information for comparison with subsequent sensitivity analysis results (Section 5.6). The baseline analysis procedure and assumptions for each analysis step have been discussed in Sections 5.3 and 5.4, but they are summarized here for convenience.

There are six key elements of the baseline analysis framework:

(i) The mid-value, upper bound, and lower bound supplied by the panelists for each elicitation question are assumed to correspond to the median, 95th percentile, and 5th percentile, respectively, of a split lognormal distribution with the mean calculated assuming the upper tail is truncated at the 99.9^{th} percentile.

(ii) Panelist responses are not adjusted to account for possible overconfidence in the uncertainty estimates for each elicitation response.

(iii) Split lognormal distributions are summed by assuming perfect rank correlation among the individual terms.

(iv) The group estimates of the total LOCA frequency parameters (i.e., median, mean, 5^{th} percentile, and 95^{th} percentile) are determined by aggregating the total LOCA frequency estimates pertaining to each panelist.

(v) The group estimates of the total LOCA frequency parameters are defined as the geometric means of the individual estimates.

(vi) Panel diversity is characterized by using two-sided 95% confidence intervals based on an assumed lognormal model for the individual estimates.

These key baseline analysis elements were selected to be consistent with the analysis framework (Section 5.1). If several consistent choices were possible, then conservative analysis choices were made. Element (i) naturally results from the lognormal analysis framework defined in Sections 5.1 and 5.3.1. Element (ii) uses the raw responses provided by the panelists. However, panelist overconfidence is a well-known phenomenon [5.2], and possible adjustment schemes are evaluated as sensitivity analyses (Section 5.6.2). Element (iii) assumes maximal correlation between the contributions to the piping and non-piping frequencies for each panelist.

A number of natural aggregation points exist for combining the individual panelist responses: (1) after the total LOCA frequencies have been determined, (2) after the separate piping and non-piping component frequencies have been calculated, and (3) at each individual elicitation response. Aggregation point (1) was chosen for element (iv). This selection stems from the careful attention paid during the individual elicitation sessions on eliminating inconsistencies between the responses of each panelist. Also, the correlation between a panelist's piping and non-piping responses is retained by this selection. Therefore, aggregation of the individual total LOCA frequency estimates is the preferred choice.

The use of the geometric mean to aggregate the individual results in element (v) and confidence intervals to capture panel diversity in element (vi) results from the interpretation and structure of the individual responses. As discussed in Section 5.4, both of these elements are a consequence of the assumed lognormal structure of the individual bottom-line LOCA frequency estimates. In addition to being consistent with the actual elicitation results, this approach is motivated by the fact that the estimated LOCA frequencies being combined are based on the results of an expert elicitation. A fundamental assumption underlying the use of expert elicitation is that the elicitation responses are not systematically biased. The individual responses can be highly uncertain and they can differ drastically, but there is no significant systematic bias. The questionnaire structure and the elicitation training were designed to achieve this goal. The results of the training exercise in Appendix C are consistent with this fundamental assumption.

A consequence of the assumption of no systematic bias is that the group opinion should be somewhere in the middle of the group, especially if there are wide differences in the results. One obvious choice would be to calculate group opinion using the median of the individual opinions. Other possible choices include the geometric mean and the trimmed geometric mean because they are also estimates of the median. The arithmetic mean is another possible choice. However, the arithmetic mean of individual opinions is often not a good measure of the median group opinion when the opinions are widely varying. In this case, the arithmetic mean is dominated by the one or two largest results and cannot be fairly described as a group opinion.

There is support for the use of the median or geometric mean in the literature. For example, in Reference 5.2, Meyer and Booker state: "To overcome the influence of extreme values when forming an aggregation estimate, use the median or geometric mean" (p. 310). In Reference 5.3, von

Winterfeldt and Edwards recommend averaging probabilities (p. 136). However, they conclude: "The only context in which we have any reservations about this conclusion is that of very low probabilities - - - - - for such extreme numbers, we would prefer averaging log odds to averaging probabilities." For very low probabilities, averaging log odds is equivalent to using the geometric mean. The geometric mean combination was therefore chosen to include all of the elicitation results and ensure that the group opinion is not overly influenced by the largest opinions.

5.6 Sensitivity Analyses

A number of sensitivity analyses were conducted to examine the effects of modifying the analysis framework used to calculate the baseline LOCA frequency estimates outlined in Section 5.5. There are many possible alternatives to the baseline elements which could have been chosen to estimate the group LOCA frequency bottom-line parameters. The analyses described below were selected either to evaluate plausible alternatives or to quantify the maximum possible LOCA frequency ranges using alternative baseline elements. The sensitivity analyses apply to only one element at a time; combinations of alternatives are not evaluated.

5.6.1 Mean Determination

The baseline estimates of the mean are based on a split lognormal truncated at the 99.9th percentile. Sensitivity analyses analyzed the effects that other distributional shapes have on the calculated means for the individual elicitation responses. Several other plausible choices were evaluated: a split lognormal, a lognormal corresponding to the upper half of a split lognormal (upper tail lognormal), a split log-triangular or a normalized split lognormal truncated at the 99.9th percentile (the mass beyond the truncation point is proportionately distributed to the upper half). The effects of varying the truncation point for the split lognormal were also examined. See Section 7.6.1 for the results of the sensitivity analyses for the mean.

5.6.2 Overconfidence Adjustment

The baseline estimates assume that the upper bound (UB) and lower (LB) supplied by the panelists for each elicitation question correspond to the 95th percentile and 5th percentile, respectively, of a lognormal distribution. In other words, the baseline estimates are based on the assumption that all the (LB, UB) uncertainty intervals have 90 percent coverage, i.e., they all have a 90 percent chance of containing the true value. However, extensive experience with such subjectively determined intervals has demonstrated that people tend to underestimate the uncertainty in their answers, i.e., they tend to be overconfident. Instead of having about a 90% chance of containing the true value, nominal 90% coverage intervals only have actual coverage of about 30% to 70% [5.2].

Accordingly, sensitivity analyses were performed to evaluate the effects of assuming that the panelists' uncertainty intervals did not always correspond to the stated 90% coverage interval. The mid-values supplied by the panelists were not adjusted, i.e., they always correspond to the median of the assumed lognormal distributions. Two types of overconfidence adjustments were investigated. The first type adjusted the coverage intervals directly for each individual elicitation response. For example, one sensitivity analysis assumed that the 90% intervals supplied by the panelists correspond to only 50% coverage, i.e., the associated UBs and LBs were assumed to correspond to the 75th percentile and 25th percentiles, respectively, of their associated lognormal distributions. The second type of overconfidence adjustment was applied to the error factors associated with the bottom-line LOCA frequency estimates and not the individual responses as before. Because the error factor of any estimated LOCA frequency parameter depends on the uncertainty intervals on which the estimate is based, increasing the error factor is equivalent to an indirect overconfidence adjustment of the underlying uncertainty intervals. Both blanket and target overconfidence adjustments were applied. In a blanket adjustment, all panelists were adjusted by the same amount. In a targeted adjustment, panelists with relatively small uncertainties were adjusted more than panelists with larger uncertainties. The rationale for a targeted adjustment is that panelists with relatively small uncertainties are more likely to be overconfident. See Section 7.6.2 for the results of the overconfidence adjustment analyses.

5.6.2.1 Coverage Interval Adjustment - For any UB and LB supplied by a panelist, formulas were developed to calculate the results of any level of overconfidence adjustment. The overconfidence adjustment is accounted for by defining an adjusted UB and LB corresponding to the 95^{th} and 5^{th} percentiles, respectively, of the panelist's distribution. This allows the results of the overconfidence adjustment to be calculated by simply substituting the adjusted UB and LB into the formulas developed for the unadjusted UB and LB. As stated previously, the mid-value (MV) supplied by the panelist are not altered by the overconfidence adjustment.

The formula for adjusting a UB is presented first. Let b be the value of a UB supplied by a panelist and let p be the value of the percentile of the panelist's assumed lognormal distribution corresponding to b. Define $UB_p(b)$ as the adjusted value of the UB, i.e., $UB_p(b)$ corresponds to the 95^{th} percentile of the panelist's adjusted distribution. (For the case of no adjustment, $p = 95$ and $UB_{95}(b) = b$.) Let m be the value of the associated MV supplied by the panelist. Denote the error factor of the panelist's adjusted lognormal distribution by r_p. From Equation 5.10,

$$r_p = UB_p(b)/m \tag{5.32}$$

Before adjustment, the error factor, r, is equal to b/m. Hence, the overconfidence adjustment is equivalent to changing the error factor from r to r_p for each elicitation response.

Referring to Section 5.3.2, let σ^2 be the variance of the underlying normal of the panelist's lognormal distribution. From Equation 5.10,

$$r_p = \exp(1.645\sigma) \tag{5.33}$$

From Equation 5.1, Equation 5.5 and the relation $r = b/m$,

$$r = \exp(k_p\sigma) \tag{5.34}$$

It follows from Equations 5.33 and 5.34 that

$$r_p = r^{c(p)}, \tag{5.35}$$

where $c(p) = 1.645/k_p$.

From Equations 5.32 and 5.35, and the relation $r = b/m$,

$$UB_p(b) = b^{c(p)} \cdot m^{1-c(p)}, \tag{5.36}$$

where b is the UB and m is the MV supplied by the panelist. This is the desired formula for the overconfidence adjustment to the UB.

Because the UB and LB are not necessarily symmetric about the MV, the LB must be adjusted independently of the UB. As for the UB adjustment, let b' be the value of a LB supplied by a panelist and let p' be the value of the percentile of the panelist's assumed lognormal distribution corresponding to b'. Define $LB_{p'}(b')$ as the adjusted value of the LB, i.e., $LB_{p'}(b')$ corresponds to the 5^{th} percentile of the panelist's distribution. (For the case of no adjustment, $p' = 5$ and $LB_5(b') = b'$.)

The derivation of the formula for $LB_{p'}(b')$ is analogous to the derivation of the formula for $UB_p(b)$. The only difference is that the error factor is equal to the MV divided by the LB, instead of the UB divided by the MV. The adjusted error factor ($r_{p'}$) is defined as m/b' and Equation 5.32 can be rewritten as

$$r_{p'} = m/\mathrm{LB}_{p'}(b'), \tag{5.37}$$

where m is the MV supplied by the panelist. Equation 5.33 remains unchanged for $r_{p'}$ but Equation 5.34 for the unadjusted error factor (r') becomes

$$r' = \exp(-k_{p'}) \tag{5.38}$$

with $k_{p'}$ the pth percentile of the standard normal distribution associated with the LB and MV response.

Note that, because p' is the percentile corresponding to a LB, $p' < 50$ and $k_{p'}$ is negative. From the symmetry of the normal distribution, $k_{p'} = -k_{100-p'}$. Hence Equation 5.35 for the LB adjustment is

$$r'_{p'} = [r']^{c'(p')}, \tag{5.39}$$

where $c'(p') = 1.645/k_{100-p'}$. Then Equation 5.36 becomes

$$\mathrm{LB}_{p'}(b') = [b']^{c'(p')} \cdot m^{1-c'(p')}, \tag{5.40}$$

where b' is the LB and m is the MV supplied by the panelist. This is the desired formula for the overconfidence adjustment to the LB. Equations 5.36 and 5.40 express similar forms for the adjusted UB and LB responses.

Because an overconfidence adjustment implies that the adjusted percentile is always closer to 50% than the nominal percentile, $p < 95$ and $p' > 5$. Hence, $c(p)$ and $c'(p')$ are always > 1, and the adjusted bounds are always further away from the median than the unadjusted bounds. Therefore, the panelist's adjusted distribution is always broader than the unadjusted distribution. In other words, an overconfidence adjustment always increases the uncertainty in the panelist's response.

There were five separate overconfidence adjustments evaluated to examine the effect on the LOCA frequency estimates: two blanket adjustments and three targeted adjustments. As previously stated, the blanket adjustments adjusted the responses for all panelists by the same amount. The targeted adjustment used a two level adjustment. Those panelists expressing larger uncertainty were adjusted a lesser amount, while those panelists with a smaller uncertainty were adjusted a greater amount. The various overconfidence adjustment schemes are summarized in Table 5.1. The blanket 1 (B1) adjustment was the most severe adjustment because the responses which ideally represent 90% coverage intervals were assumed to represent a 50% coverage interval. Each successive adjustment was meant to be less severe. The B2 adjustment converted the supplied 90% coverage interval responses to 60%.

The targeted adjustment uncertainty ranges divided panelist responses into three error factor bins: less than 10 (less uncertainty), between 10 and 50, and greater than 50 (greater uncertainty). The targeted 1 (T1) adjustment did no adjustment of panelists having error factors greater than 50 (3 – 4 panelists) and adjusted the coverage interval to 50% for all remaining panelists. The targeted 2 (T2) adjustment used the same criteria, but adjusted the more uncertain responses to 80% while remaining panelists were adjusted to 60%. The targeted 3 (T3) only adjusted the panelists having an error factor < 10 from a 90% to a 60% coverage interval. Results of this analysis are presented in Section 7.6.2.1.

Table 5.1 Elicitation Response Adjustment Schemes

Overconfidence Adjustment	Coverage Interval 1	Population	Coverage Interval 2	Population
Blanket 1	50%	All	NA	NA
Blanket 2	60%	All	NA	NA
Targeted 1	90%	4	50%	remaining
Targeted 2	80%	4	60%	remaining
Targeted 3	90%	4	60%	remaining

5.6.2.2 Error Factor Adjustment - The error factor adjustment is applied to the error factor associated with a panelist's total LOCA frequency estimates. The error factor is the ratio of the 95th percentile to the median, but the median is unadjusted. Hence, the error factor adjustment is equivalent to adjusting the estimated 95th percentile of a panelist's LOCA frequency distribution. The adjustment can be performed by either changing the value of the percentile or by changing the value of the error factor. Changing the percentile value is similar to the confidence level adjustment in the previous section except that it applies to the bottom-line 95th percentile estimate rather than the UB elicitation responses.

This sensitivity analysis adjusted the error factor directly. Each panelist's error factor is compared with the geometric mean of all the error factors. Error factors falling below the geometric mean are adjusted up to the geometric mean. Error factors above the geometric mean are not adjusted. For any error factor which is adjusted, let EF_0 be its original value and let EF_a be its adjusted value. The median, m, of the adjusted LOCA frequency distribution is the same as the original median. All other bottom-line parameters of the adjusted distribution are calculated using the formulas in Section 5.3.2 from the mid value and adjusted error factors. This results in a variable adjustment as a function of plant type and LOCA category and operating time. See Section 7.6.2.2 for additional information and the results of this sensitivity analysis.

Given EF_a and EF_0, a measure of the overconfidence adjustment is the value of the percentile, p_a, in the adjusted distribution corresponding to the original error factor. From Equation 5.10, the underlying standard deviation of the adjusted lognormal distribution (σ_a) is given by

$$\sigma_a = \ln(EF_a)/1.645 \qquad (5.41)$$

From Equation 5.5,

$$EF_0 = \exp(k_a\ \sigma_a), \qquad (5.42)$$

where = k_a is the p_ath percentile of the standard normal distribution. Solving Equation 5.42 for k_a and using Equation 5.41 yields

$$k_a = 1.645\ [\ln(EF_0/\ln(EF_a)] \qquad (5.43)$$

Then, k_a can be calculated from Equation 5.43 and p_a can be determined from tables of the normal distribution. Because $EF_a > EF_0$, $k_a < 1.645$ and therefore $p_a < 95$. This result is to be expected, because increasing the error factor broadens the distribution.

5.6.3 Correlation Structure

The baseline estimates of the percentiles are calculated assuming perfect rank correlation between the summed component distributions (Block 2.6). This baseline assumption maximizes the pair-wise correlations between the distributions. Because the correlations are assumed to be positive, they are minimized by assuming the distributions are independent, making all correlations equal to zero. Accordingly, as a sensitivity analysis, the effect of assuming that the summed distributions are independent is evaluated.

An evaluation of the effects of an assumed independent correlation structure is only possible through Monte Carlo simulation. Because of the very large number of individual distributions that comprise the elicitation responses, the bottom line estimates were calculated only for selected panelist responses. Accordingly, ten simulation trials were selected to span several important variables as summarized in Table 5.2. First, both the panelists and the range of LOCA plant type combinations (i.e., PWR piping, PWR non-piping, BWR piping, and BWR non-piping) and time periods were sampled. Second, a number of distributions representing contributing piping components (or non-piping subcomponents) which must be summed to develop the bottom-line estimates were used. Third, and most important, the Monte Carlo trials were selected to span the range of distinguishing characteristics representative of the elicitation responses. These distinguishing characteristics (last column in Table 5.2) indicate whether the elicitation responses are generally symmetric (S in Table 5.2) or asymmetric (U in Table 5.2), and indicate the relative magnitude (small, moderate, large) of the upper or lower error factors (EF in Table 5.2). The characteristics are representative of the variability among the entire population of elicitation responses. Additional details concerning the Monte Carlo simulation and the results are contained in Section 7.6.3.

Table 5.2 Summary of Monte Carlo Trials

Trial Number	Number of Distributions	Panelist	LOCA Plant Type/Time Period	Distinguishing Characteristics
1	12	A	BWR-1 Piping @ 25 yrs	S, small EF
2	12	A	BWR-2 Piping @ 25 yrs	U, small upper EF
3	2	C	PWR-6 Piping @ 25 yrs	U, large upper EF
4	4	C	BWR-3 Piping @ 25 yrs	S, moderate EF
5	14	G	PWR-5 Non-Piping @ 60 yrs	U, large lower EF
6	5	C	BWR-3 Non-Piping @ 25 yrs	U, large lower EF
7	8	J	PWR-5 Non-Piping @ 25 yrs	S, large EF
8	7	I	PWR-4 Piping @ 25 yrs	S, moderate EF
9	4	E	BWR-4 Non-Piping @ 25 yrs	U, large lower EF
10	9	B	PWR-3 Non-Piping @ 25 yrs	S, small EF

5.6.4 Aggregation

As discussed in Section 5.4, the individual total LOCA frequencies are combined to obtain group estimates of the total passive LOCA frequencies for BWRs and PWRs. The baseline estimates are calculated using the geometric mean of the individual estimates (see Equation 5.28). However, there are a number of other ways to construct group estimates. A number of sensitivity analyses are performed to examine the effects of different approaches to aggregating the panelists' responses. These include different measures of group opinion, combining panelist responses at different stages of the analysis, and different methods of calculating the bottom-line lognormal parameters.

5.6.4.1 Group Estimate – As discussed in Section 5.5, the key requirement for a group estimate based on expert opinion is that it be somewhere in the middle of the group. Consequently, a group estimate should not be overly dependent on outliers or extreme values. The key requirement implies that a group estimate should be a measure of central tendency, and the geometric mean (GM) is a plausible candidate. However, because the individual estimates based on the panelists' responses were usually highly variable, the GM could conceivably be driven by one or more outliers. An alternative group estimate is the trimmed geometric mean (TGM), which is the geometric mean of the values remaining after the largest and smallest values have been omitted from the set. Still another alternative is to use the median of the distribution of individual estimates. While the median is much less dependent on outliers than the GM or TGM, it literally depends on only the center of the distribution and does not reflect any other aspect of the distribution. As a sensitivity analysis, both the TGM and median of the distribution of total LOCA frequencies are compared with the GM (see Section 7.6.4.1).

Another measure of central tendency is the arithmetic mean (AM). The AM is larger than the GM, except when all values in the distribution are equal, and therefore yields more conservative LOCA frequency estimates, except for the special case of equality. When the individual estimates span one or more orders of magnitude, as they do in this study, the AM is much more sensitive to large values than is the GM and consequently is much larger than the GM. See Section 7.6.4.1 for a comparison of the AMs to the GMs as group estimates.

5.6.4.2 Aggregation Point - The aggregation point is the point in the analysis framework where the individual responses are combined to construct a group response. The baseline analysis uses the last possible aggregation point and combines the individual total LOCA frequencies. One alternative is to combine the individual piping and non-piping LOCA frequencies to construct group piping and non-piping LOCA frequencies, respectively. The two group estimates are then added as described in Section 5.3.4 to construct group estimates of total LOCA frequencies. A second alternative is to combine at the earliest possible aggregation point, i.e., to combine the individual responses to each of the elicitation questions. Both mid-value and bounding responses are developed for the responses, and the responses are analyzed using the calculation procedure in Section 5.3 to determine total LOCA frequency estimates for BWR and PWR plant-types. This is equivalent to synthesizing responses for a 13th panelist from the responses of the other 12 panelists (Section 5.2). See Section 7.6.4.2 for a sensitivity analysis based on these two alternative aggregation points.

5.6.4.3 Aggregation Parameters - In the baseline methodology, estimates for the mean and other bottom-line parameters (i.e., median, 5th and 95th percentiles) are aggregated directly from the individual panelist responses (Section 5.5). An alternative approach is to calculate the bottom-line mean estimates from the aggregated percentile estimates by assuming a distributional relationship for the percentile estimates. The method used for this sensitivity analysis (called the MEF method) calculates the group 5th and 95th percentile estimates from the aggregated median and error factor values instead of directly aggregating these parameters as in the baseline methodology. Then, the bottom-line mean is calculated from the median, 5th and 95th percentile values by assuming that the underlying distribution is a split lognormal which is truncated at the 99.9th percentile. This is the same distributional form assumed for calculating the mean estimates from the individual elicitation responses in the baseline methodology (Section 5.5).

The 5th and 95th percentile estimates determined by the MEF approach are identical to the estimates determined by direct aggregation since all the individual panelist estimates can be equivalently represented by the 5th, 50th, or 95th percentiles (baseline methodology) or by the median and upper and lower error factors (MEF approach). However, the means can be determined at several points in the analysis and differences do result from this choice. The baseline approach calculates the means at the earliest possible point in the analysis from the individual distributions while the MEF approach calculates the means as the last step in the analysis after aggregating all individual responses. The means could also be calculated after the bottom-line percentile estimate for each individual panelist is determined. It is postulated, however, that the means from the MEF approach will differ most from the baseline approach because the characteristics of the aggregated distributions are likely to be the most different from the individual elicitation response distributions due to the intermediate processing steps (Section 5). Additional details and results of this sensitivity analysis are provided in Section 7.6.4.3

5.6.4.4 Mixture Distribution Aggregation –This is a method of expert opinion aggregation for probability distributions used in the NUREG-1150 studies [5.4] which uses the arithmetic average of the panelists' probability distributions as the aggregate probability distribution. The method is described briefly on pages 3-6 of Reference 5.5. It is stated there that one of the principles used in the expert judgment process was that the aggregation of judgments from various experts should preserve the uncertainty that exists among alternative points of view.

The mixture distribution approach assumes that the expert panel is a random sample from the population of all experts and that the goal is to obtain an unbiased estimate of the aggregate distribution function of LOCA frequency averaged over the population of all experts. For a panel of N experts, denote the cumulative distribution function for expert i by $F_i(x)$ and the group cumulative distribution function for the LOCA frequency by G(x). Then

$$G(x) = \frac{1}{N}\sum_{i=1}^{N} F_i(x) \tag{5.44}$$

$F_i(x)$ is the probability that expert i would assign to the statement that the LOCA frequency X is less than x. G(x) is the group estimate of the probability that the LOCA frequency X is less than x. The corresponding density functions are obtained by differentiating Equation 5.44:

$$g(x) = \frac{1}{N}\sum_{i=1}^{N} f_i(x) \tag{5.45}$$

Integrating Equation 5.45, the mean LOCA frequency is given by

$$\bar{x} = \int x g(x)dx = \frac{1}{N}\sum_{i=1}^{N}\int x f_i(x)dx = \frac{1}{N}\sum \bar{x}_i \tag{5.46}$$

Equation 5.44 has the form of a mixture distribution. It is sometimes said that this means that Equation 5.44 is a valid means of aggregating expert opinion only if one of the experts is correct. This is not a valid objection. One should consider the panel of experts as a random sample of experts from a population of experts. Denoting the number of experts in the population by N_{all}, the population aggregate distribution is given by

$$G_{pop}(x) = \frac{1}{N_{all}}\sum_{j=1}^{N_{all}} F_j(x) \tag{5.47}$$

If the panel of N experts is a random sample from the population of N_{all} experts, G(x) from Equation 5.44 is an unbiased estimate of $G_{pop}(x)$.

Note that no matter how large a sample is taken, there will still be large variability of expert opinion. The NUREG-1150 method captures this uncertainty (see page 9.7 of Reference 5.6).

The mixture distribution aggregation scheme will always result in higher mean and 95th percentile estimates and lower 5th percentile estimates than the other aggregation schemes (see Section 7.6.4.4). Consequently, the mixture distribution scheme exhibits the greatest difference between the 5th and 95th percentiles. However, the mean and 95th percentile estimates are often dominated by the maximum panelist estimate while the 5th percentile is often dominated by the minimum panelist estimate. These characteristics imply that the extreme individual estimates will often dominate the mean, 5th, and 95th percentile estimates. Therefore, the mixture distribution will not represent central estimates of group opinion when the differences among the individual estimates are large. Obtaining central estimates of group opinion was a principal objective of this elicitation (Section 2). Even if conservative central estimates of LOCA frequencies are required, the mixture distribution scheme may not be appropriate. In such a case, either high confidence bounds (Section 5.4) or the quartile method (Section 5.6.5) would yield conservative estimates.

5.6.5 Panel Diversity

As discussed in Section 5.5, the baseline approach to panel diversity is to construct two-sided confidence intervals for the group estimates of the bottom-line parameters of the LOCA frequencies, assuming a lognormal structure for the individual estimates (Section 5.4). The baseline measures of panel diversity are 90% confidence intervals centered on the geometric mean of the individual estimates.

An alternative approach is a non-parametric one, where no assumptions are made about the probabilistic structure of the individual estimates. For each set of individual panelist estimates, a quartile interval is constructed from the upper and lower quartiles of the individual estimates. The lower and upper quartiles represent the 25th and 75th percentiles, respectively, of the individual panelist estimates. In this quartile method, the median (or 50th percentile estimate) is a natural measure of central group opinion. A sensitivity analysis comparing the confidence interval and quartile method is presented in Section 7.6.5. In Section 7.6.4.1, it is shown that the geometric mean and median provide similar measures of central group opinion. Thus, a quartile interval is a measure of group diversity about the median analogous to a confidence interval about the geometric mean.

5.7 References

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6. QUALITATIVE RESULTS AND DISCUSSION

The detailed results from the elicitation exercise are presented in the next two sections of this report. In Section 6, the qualitative results (i.e., rationale and insights) are presented. In Section 7 the quantitative (i.e., numerical LOCA frequency estimates) are presented. The qualitative results presented herein focus on recurring insights for important technical issues which contribute to passive system LOCAs. These insights usually summarize individual responses, and do not necessarily represent a panel consensus or majority opinion. However, most insights were generally raised by more than two panel members, unless explicitly noted. Individual insights expressed by less than two of the participants are typically not presented. However, some of the more interesting individual opinions are presented with their associated quantitative results in Appendix L, where the detailed results are discussed. The bulk of the qualitative rationale contained in this section was obtained during the individual elicitations, although insights obtained from the three group panel meetings are also included. The insights from the individual elicitation sessions have been mined from the associated minutes, handouts, and written responses. Minutes were recorded during each interview session. In addition, the participants often provided handouts with their preliminary responses prior to their elicitations. Then, after each session, most participants provided more complete written rationale to accompany their final elicitation responses.

6.1 Base Case Evaluations

There were two basic approaches conducted to evaluate the base case conditions: review and assessment of service history data and probabilistic fracture mechanics (PFM) analyses. Some broad opinions surfaced regarding each approach. The advantages of using a service history-based approach are that it naturally captures the effects of all observed degradation mechanisms as well as including the actual mitigation, loading, and plant operating conditions experienced in the plants. As such, service history analysis provides an accurate characterization of past precursor events (e.g., leaks). The perceived difficulties include the fact that it is difficult to extrapolate precursor event data to predictions of LOCA frequencies, especially for future operating periods. A number of the panelists commented that they believe that the surface history predictions are accurate for assessing LOCA frequencies over the first 25 years of plant operation, but are somewhat skeptical of future operating period estimates. Future predictions can be difficult using service history data because emerging and future degradation mechanisms are not contained within the databases. Also, the use of service history data can bias the predictions if data related to past problems which have already been successfully mitigated (e.g., IGSCC in BWRs) and infant mortality rates have not been adequately screened. The data must be carefully evaluated to avoid these types of analysis pitfalls. Finally, a number of the panelists commented that only the reporting of actual leaking cracks is actually comprehensive within the database. The reporting of the non-leaking precursor events is likely under reported.

There are several perceived advantages of the PFM approaches. First, they are capable of predicting the future damage for both currently existing and newly occurring degradation mechanisms that have been adequately modeled. This is valuable in assessing the required action necessary to mitigate emerging mechanisms before their impact is fully evident in the service history data. Also, PFM is capable of predicting the possible evolution of leaking and non-leaking cracks to a larger LOCA size.

There are several disadvantages that panelists expressed when using PFM to predict LOCA frequencies. At least one panelist indicated that only a few degradation mechanisms are realistically modeled. Specifically, the accuracy of the models for predicting crack initiation and crack linking from multiple initiated cracks was questioned for SCC-type mechanisms. This potential limitation can lead to underestimates of service cracking frequency and predicted crack lengths and result in unconservative LOCA frequency estimates. However, several panelists countered that conservative design stresses and transients are usually applied in the PFM analyses that are not representative of actual service history stress values. These stresses result in conservative LOCA frequency estimates. Finally, it was felt by some panelists that the existing PFM models are too simplistic. The input variables, assumptions, and models are usually idealizations whereas the variables which are most important to LOCA susceptibility are often very complex. While the intent is to make conservative idealizations in the analysis, this contention is sometimes difficult to prove.

As a result of these beliefs, most panelists chose to anchor their elicitation responses to one or both of the service history-based base case analyses (Sections 3.5 and 4) for making current day (i.e., 25 year) LOCA frequency estimates. For future year projections (i.e., 40 or 60 year estimates), a number of the participants augmented the service history-based assessments with results from the PFM studies and associated sensitivity analysis to develop their responses. Several panelists also relied primarily on the results from PFM analysis for all their predictions, but benchmarked them in some way by the service-history data. A more complete description of anchoring philosophy chosen by each panelist is found in Appendix K.

6.2 Safety Culture Effects

There are several competing issues that can either positively or negatively affect the future utility and regulatory safety cultures (Section 3.4.2). Panelists assessed these issues to identify the strongest factors. For the most part, the panelists believe that the positive influences will slightly outweigh the negative factors and that the utility and regulatory safety cultures are highly correlated. Therefore, a slight future improvement in the overall safety culture compared with the current culture is often expected with no real distinctions between utility and regulatory aspects. Some of the qualitative rationale supporting this weak trend follows.

There are several positive safety culture influences that are expected. Several panelists believe that deregulation will have a positive effect on safety culture by requiring the industry to adopt an asset management strategy to optimize continued plant operation. These panelists believe that favorable economic incentives are in place today that bode well for the future of nuclear power. Plants are therefore being viewed as assets that need to be maintained to maximize their investment potential. As a result, management realizes that proactive and periodic maintenance is preferable to the risk of an extended outage. Another positive factor for improving future safety is the large experience and knowledge base developed during the previous 25 years of operation. This information allows plant operators to focus on actual problems, and not just perceived or postulated problems. Utilities and regulators also now have risk-informed methods to use as a basis for identifying important areas of concern. However, one panelist was disturbed by his perception that some utilities do not want to invoke risk-informed practices unless there is an immediate economic payback. However, this was not a majority opinion.

Some panelists also expect the safety culture to improve over the next 15 years due to the increased awareness of the consequences of passive system degradation. Recent high-profile examples of vessel head degradation at Davis-Besse and control-rod drive mechanism (CRDM) cracking at a number of PWR plants have illustrated the potential economic ramifications associated with extended plant outage. Also, there is a realization that problems at one plant can affect the public's perception of the entire nuclear industry. One panelist commented that another high visibility event, similar to the Three Mile Island (TMI) accident, could result in a loss of public trust that would be difficult to restore.

There are also several negative safety culture influences that may temper these predicted improvements. Several panelists cautioned that focusing on economic performance can also be detrimental. Economic pressures to keep plants operating, especially at high capacity factors, may create a poor safety environment. Utilities could be tempted to forego maintenance and inspections unless there are near-term economic incentives. Another potential negative safety culture factor is the inevitable decommissioning of plants. As plants approach their decommissioning date, safety may suffer as the cost benefit for plant maintenance decreases. There are also a number of personnel issues associated with decommissioning that could negatively affect safety culture. The morale of the plant operating personnel may suffer due to job uncertainty. Furthermore, the most competent plant personnel may migrate to other industries first and new workers may be difficult to entice to the field. The resultant workforce may then be less competent, overworked, and unmotivated during the latter years of plant operations.

A number of panel members raised concerns about the currently aging nuclear workforce. Obviously, there is a need to attract young, energized new staff into the field. However, a potential stumbling block to satisfying this goal is the fact that the number of the universities with nuclear engineering programs has decreased. Another negative influence mentioned by several panelists is complacency. With a few notable exceptions, the nuclear industry has an excellent safety record over the past 25 years. Unfortunately, this good record may breed a false sense of security. These panelists are concerned that complacency could lead to an overall erosion of safety unless proper vigilance is maintained.

Some safety culture concerns mentioned during the elicitations are due to the influence of politics and economics on the regulatory environment. Some panelists commented that the safety culture could be adversely affected by the country's energy demands. If demand continues to increase, and supplies become stretched, there may be pressure to continue plant operations at the detriment of the safety culture. A few panelists also expressed concerns that political pressure may be applied to keep plants operating and people employed at the risk of safety. Research and plant oversight can also be affected by future budget cuts at the NRC.

In addition to these positive and negative safety culture influences, the panelists believe that some factors exhibit compensating features. One example is the role of industry in ASME code activities. A few panelists believe that strong industry participation benefits safety culture outright. However, others believe that this participation can be detrimental if regulatory opinions are underrepresented on important issues. This possible weakness is compensated by the NRC's ability to not endorse aspects of the code where important disagreements exist. The design and construction of new plants also has compensating factors. As new plants, with improved designs and materials are developed, the knowledge gained could benefit maintenance, repair, and replacement activities in the existing fleet. Conversely, resources could shift to the newer, more efficient plants, which could harm safety at the older, less efficient plants. Multiple versus single plant sites are also associated with compensating factors. The detriment of a multiple plant utility is that there is typically less oversight per plant. However, some potential advantages that the multiple plant utility have is that there is often more people reviewing important decisions, and there is an increased opportunity for reassignment of personnel as plants are decommissioned. This somewhat offsets the previously discussed concerns about deteriorating morale as a result of decommissioning.

Many panelists discussed issues related to current safety culture differences among plants as a function of design-type, U.S. attitudes, and plant management philosophy. For instance, a number of panelists expect that the plant-type will influences safety culture. Specifically, the BWR plant inspectors and operators are believed to be more experienced in identifying degradation and developing successful mitigation strategies than their PWR counterparts. This opinion results from the BWR experience in mitigating IGSCC in the 1980's. The implication is that the current BWR safety culture may be better than the PWR safety culture. However, many of the same panelists also believe that the PWR experience with emerging PSWCC issues may equalize the future safety culture.

Additionally, panelists with the most extensive international experience tend to view the current U.S. safety culture less favorably than the other panelists. Specifically, a number of these panelists believe that the current U.S. safety culture is weaker than in other countries. A concern is that common U.S. practice strives for the most expedient solution without addressing the root cause of the problem. A cited example of this practice was the U.S. industry's perceived delay in adopting the lessons-learned from earlier French CRDM cracking experience. These, and other possible, differences among the current plant population are irrelevant since the elicitation questions ask the future safety culture to be assessed relative to the current safety culture. Hence, the elicitation results normalize any initial panelist differences. However, any quantitative effects stemming from current day safety culture differences are individually assessed during the current day LOCA frequency estimates.

Many panelists also expressed the opinion that safety culture is a highly plant-specific factor. Most panelists believe that the industry is generally acting in a consistent, safety conscious manner. However, individual plants can deviate markedly from general industry philosophies and practices. The vessel head degradation at Davis-Besse was a commonly cited example. Panelists argued that the LOCA frequencies at these less safety conscious plants could be greatly elevated compared with the remaining population. However, this trend would not affect either the mean LOCA frequency or safety culture estimates, but would more likely affect the upper bounds of both distributions.

The panelists were also asked to comment on the relationship between safety culture and LOCA size category. Generally, the panel members expressed the opinion that safety culture is only very weakly correlated with LOCA size, if at all. Small leaks tend to impact plant availability, so the plants have an economic incentive to guard against such events. However, a few panel members believe that plant safety is focused on preventing failure in the largest pipes due to their increased risk significance.

6.3 Passive System Failure Insights

In the sections that follow, some broad passive system failure insights are discussed. The insights are first segregated into issues associated uniquely with piping and non-piping LOCA frequency contributions. Then, insights related to general issues that contribute to the underlying LOCA frequencies are presented including aging mechanisms; the effect of component size; mitigation and maintenance; the effect of operating time; and estimation uncertainties.

6.3.1 Piping and Non-Piping LOCA Frequency Contributions

Table 6.1 indicates the major piping systems and non-piping subcomponents that provide the greatest LOCA frequency contribution for each LOCA size category. A LOCA of a given size can occur by either a complete break of the smallest component supporting that LOCA size, or a partial break of a larger component. Most panelists believe that complete failure of a smaller component is more likely than partial failure of a bigger component. Therefore most panelists expect that the smallest diameter piping system or subcomponent that could support a particular LOCA size or category is the dominant LOCA frequency contributor. This belief is usually apparent in Table 6.1.

The major BWR and PWR piping Category 1 and 2 LOCA contributors are the smaller diameter instrument and drain lines (Table 6.1). These lines are susceptible to a number of failure scenarios that the larger diameter lines are not susceptible to, e.g., mechanical fatigue of socket weld fittings. In addition, a flaw of a given size in the smaller diameter pipe is a larger percentage of the pipe circumference than an equivalent size flaw in a larger diameter pipe. Therefore, this flaw is closer to the failure flaw size in the smaller diameter pipe. Finally, these smaller diameter pipes are not subject to the same level of in-service inspection as their larger counterparts, if they are inspected at all.

Table 6.1 Major Piping and Non-Piping Contributors to the Various Size LOCA Categories

LOCA Category	BWRs		PWRs	
	Major Piping Contributors	Major Non-Piping Contributors	Major Piping Contributors	Major Non-Piping Contributors
1	Small diameter instrument and drain lines	CRDM (i.e. stub tube) penetrations	Small diameter instrument and drain lines	Steam generator tubes, pressurizer heater sleeves, CRDM penetrations
2	Small diameter instrument and drain lines	CRDM (i.e. stub tube) penetrations	Small diameter instrument and drain lines	CRDM penetrations
3	Primary: Recirculation, Secondary: feedwater, SRV, RWCU, RHR, Core Spray	RPV nozzles, pump and valve bodies	CVCS, RHR, SIS (DVI), PSL, surge line	Nozzles and component bodies
4	Primary: Recirculation, Secondary: feedwater, SRV, RWCU, RHR, Core Spray	RPV nozzles, pump and valve bodies	Surge Line, SIS (Accumulator), RHR, Hot leg, SRV	Nozzles and component bodies
5	Recirculation, RHR	RPV, pump, and valve bodies	Hot Leg, Surge line, RHR	Manways and component bodies
6	N/A	RPV body	Hot Leg	Component bodies

CRDM = Control rod drive mechanism
RPV = Reactor pressure vessel
SRV = Safety relief valve line
RWCU = Reactor water cleanup system
CVCS = Chemical volume control system
RHR = Residual heat removal
SIS (DVI) = Safety injection system (direct volume injection)
SIS (Accumulator) = Safety injection system (accumulator lines)
PSL = Pressurizer spray lines

The important non-piping contributors for Category 1 and 2 LOCAs are CRDM penetrations in both PWR and BWR plants. Additionally, in PWR plants, steam generator tubes and pressurizer heater sleeves are important contributors. The pressurizer and CRDM concerns stem from relatively recent emergence of PWSCC in Alloy 600 base and associated weld materials. This concern is currently at a peak because the mechanism is prevalent, yet mitigation strategies have yet to be fully implemented. The impact of PWSCC for these PWR small-diameter non-piping components is expected to decrease over the next 10 to 15 years as effective mitigation strategies are developed and implemented. Steam generator tube failure is also an expected dominant contributor based on the historically high failure rates and the decreased degradation tolerance associated with these components since the design safety factors for these tubes are less than for small bore piping. However, many panelists believe that future failure rates for steam generator tubes will decrease due to improved inspection programs, steam generator replacement initiatives, and improved secondary side water chemistry.

There is relatively good agreement that the recirculation lines for Category 3 and 4 LOCAs in BWR piping are the dominant LOCA frequency contributor. There are lingering concerns about IGSCC in these larger diameter recirculation system lines, especially for those lines which have not been replaced. This selection is counter to the general trend that the smallest lines supporting a LOCA category are the biggest contributors. The safety relief valve (SRV) lines, reactor water cleanup (RWCU) systems, residual heat removal (RHR) lines, and core spray lines all are smaller than the recirculation lines, but they were considered to be of secondary importance for this BWR LOCA category. The feedwater line is relatively large as well, and its inclusion was driven by FAC concerns.

In contrast to the BWR piping assessment, there was no clear consensus expressed about the major contributing PWR piping systems and non-piping subcomponents for Category 3 and 4 LOCAs. This uncertainty stems from the large number of piping systems and non-piping subcomponents that contribute to these LOCAs, and the lack of clear operational differences and data suggesting that any one system or subcomponent could dominate. The nozzle and component body references in Table 6.1 refer to all nozzles (reactor pressure vessel, steam generator, and pressurizer nozzles) and/or all component bodies (reactor pressure vessel, steam generator, pressurizer, pumps, and valve bodies). Hence, the dominant non-piping contributors cannot be additionally refined.

Nozzle failures are a concern because system and transient stresses can be highest at these locations. Additionally, past degradation has been experienced in these locations. Valve and pump component bodies are a concern because they typically are fabricated from cast stainless steel materials which are notoriously difficult to inspect. These materials are also subject to thermal aging which reduces the fracture toughness of the material. Fortunately, no known cracking mechanism exists for these materials as of this date. Reactor pressure vessel concerns were due to either pressurized thermal shock (PTS) for PWR plants or low temperature over pressurization (LTOP) in BWR plants.

There are only a few BWR and PWR piping systems that can support Category 5 and 6 LOCAs, and there is subsequently greater consensus about the dominant systems. The surge line and RHR line are thought to be important based on past degradation and the relatively large transients that are possible during normal operation. The hot leg was also typically a greater concern than the cold leg for Category 6 LOCAs because of its higher operating temperatures. For Category 5 and 6 PWR non-piping contributors, there remains little agreement on the dominant subcomponent failure mechanisms. There is also no consensus about Category 5 BWR subcomponent failures and only BWR vessel failures lead to Category 6 LOCAs. Hence, by default, BWR vessel failures are the dominant Category 6 LOCA contributor.

It was almost universally expressed that the contribution to the overall LOCA frequencies is greater for the non-piping components than for piping for the smaller category LOCAs in PWR plants. Specifically, steam generator tube, CRDM, and pressurizer heater sleeve failures are expected to be the most important Category 1 and 2 total LOCA frequency contributors. These expected higher failure rates also result in the expectation that PWR LOCA frequencies are higher than BWR LOCA frequencies for Category 1 and 2 LOCAs. However, the non-piping component contributions to the larger (Category 3 - 5) LOCA frequencies decreases dramatically due to the robustness (e.g., increased design margin) of these non-piping components compared to piping. Several panelists expect that non-piping contributions become significant again for the Category 6 LOCAs because the operating margins are similar to the largest piping systems.

Most panelists also believe that the quantitative non-piping LOCA frequency assessments are more challenging than the piping assessments. For non-piping, there are multiple components to consider, each with different operating requirements and characteristics. The associated design margins, materials, and inspection considerations also vary widely. Furthermore, there is little precursor data available on non-piping component degradation due to the historical focus on piping reliability.

6.3.2 Important Aging Mechanisms

Most panelists believe that precursor events (e.g., cracks and leaks) are a good barometer of LOCA susceptibility. Service history data, as discussed, provides an indication of the historical precursor frequency associated with degradation mechanisms in the operating fleet. Therefore, almost all panelists anchored their responses against available service history data. In addition, a number of the participants used the weld census data to determine the relative LOCA contributions for piping systems with similar operating characteristics. In this approach, the LOCA frequency ratio of two systems is identical to their ratio of LOCA-susceptible welds. For example, assume that a panelist believes that PWSCC is an important degradation mechanism and PWSCC susceptibility is equivalent for both the hot leg and surge line. Also assume that the surge line has a single PWSCC-susceptible weld while the hot leg has 7 PWSCC-susceptible welds. The hot leg PWSCC contribution is therefore a factor 7 higher than the surge line PWSCC contribution. This application of weld census information is a natural approach because many mechanisms, including fatigue and stress corrosion cracking, preferentially attack welds instead of the base metal. Reasons for this preferential attack are related to the weld metallurgy and chemistry, the relatively high residual stresses in the weld vicinity, and the higher weld defect density.

Most participants identified thermal fatigue, flow accelerated corrosion (FAC), intergranular stress corrosion cracking (IGSCC), and mechanical fatigue as the important degradation mechanisms to consider in BWR piping. The BWR plants are expected to be more prone to thermal fatigue problems compared with the primary side of PWR plants because they experience greater temperature fluctuations during the normal operating cycle. In BWR plants, thermal fatigue remains a concern for the feedwater lines and the residual heat removal (RHR) system. There was a rash of feedwater nozzle cracks reported in the 1970 to early 1980 time period in BWRs. Plant and system modifications were implemented after a detailed study of the problem and augmented inspections are being conducted based on NUREG-0619 [6.1] requirements. These mitigation measures have proven effective as no new thermal fatigue cracks have been discovered in these BWR feedwater nozzles over the last 20 years. Additionally, the U.S. BWR plants have been less susceptible to thermal fatigue damage than some of the foreign designs. However, thermal fatigue is an aging mechanism that could lead to a large LOCA because it does not manifest itself as a single crack, but as a family of cracks over a wide area. Thermal fatigue cracks also tend to propagate rapidly, and since it is not material sensitive (i.e., it can attack a number of materials), it is difficult to prioritize critical areas for inspections. These reasons explain why thermal fatigue is still regarded as an important LOCA contributor by many panelists.

The carbon-steel feedwater piping system in BWRs is the most susceptible to FAC of all the primary side (i.e., LOCA-sensitive) components (Section 3.4.5). The main steam line is the other major carbon-steel primary side system which experiences constant fluid flow. However, it is not as susceptible to FAC as the feedwater system because the erosion rates associated with two-phase flow are much less. While FAC caused a serious accident in the secondary side piping at Surry some 15 years ago, most panel members believe that the industry has inspection programs in place today to prevent the reoccurrence of such an event, especially in the primary side piping systems. However, one panel member expressed the concern that the water chemistry improvements (hydrogen water chemistry – HWC) that mitigate IGSCC could lead to FAC in unanticipated locations that are not monitored as part of these inspection programs.

Most BWRs that have implemented HWC routinely inject oxygen into the feedwater line to mitigate this possibility. In fact there has been virtually no FAC in BWR feedwater piping since the advent of the feedwater chemistry specifications with elevated oxygen levels. FAC and elevated oxygen do not coexist – the piping maintains a protective oxide coating, thus protecting the steel below. Therefore, FAC degradation can only result from interruptions in oxygen injection or excessive hydrogen injection could potentially result in the onset of FAC. Current U.S. instances of FAC in BWR piping have been confined to Class 3 service water systems and do not directly affect the LOCA frequencies calculated herein. Thus, many panelists believe that a properly designed and monitored HWC program should not result in higher FAC susceptibility, and their LOCA concerns due to FAC are minimal. Only one panelist remains concerned about the LOCA susceptibility due to FAC degradation.

The panel consensus is that the susceptibility of BWR piping systems to IGSCC is greatly reduced compared to what it was in the past. Measures such as improved hydrogen water chemistry, weld overlay repairs, stress relief, and pipe replacement with more crack resistant materials have led to this reduction. Inspection quality has also improved such that the probability of crack detection is much better than in the past. However, as indicated earlier, there remains concern about the failure likelihood of the large recirculation piping and the residual heat removal (RHR) lines that have not been replaced. The original piping materials are much more susceptible to IGSCC and many lines retain preexisting cracks that initiated and grew before HWC was adopted. Furthermore, at least one panelist is also concerned that the more IGSCC-resistant replacement piping materials may still crack under service conditions. This panelist cited the German plant experience with cracking in Type 347 stainless steel. Also, it is possible that cold work (e.g. due to grinding) could increase the IGSCC susceptibility of the low carbon (L grade) stainless steel that has been used as a replacement material in many U.S. plants. However, the U.S. BWR experience with L grade stainless steel piping has been very good thus far. For these reasons, many panelists believe that continued vigilance is required through the augmented inspection requirements in Generic Letter 88-01 [6.2] and NUREG-0313 [6.3].

Another aging mechanism of concern in BWR plants is mechanical fatigue. This is primarily a problem in smaller diameter piping, especially those with socket welds, and is caused by an adjacent vibration source. It was noted that locations susceptible to mechanical fatigue damage are not always obvious and that it is impossible to eliminate all plant vibrations. Another concern is that plant configuration changes can result in newly susceptible areas. This mechanism is also a prevalent root failure cause of small diameter piping in the service history databases.

Lastly, it was noted by some panelists that certain BWR piping systems (e.g., relief valve lines) may have an increased likelihood of operating transients (e.g., water hammer) compared with similar PWR systems. This is primarily due to increased valve openings during plant operation. These transients, combined with existing degradation, could make the BWR plants more susceptible to LOCAs. However, this is not a universal sentiment. At least one panelist strongly disagrees with this contention and believes that water hammer is a design-specific issue. This panelist believes that while some BWR RPV head cooling layout designs could be susceptible to water hammer, it is primarily a concern with secondary side systems and some secondary side standby safety systems. Furthermore, this panelist argues that even though relief valve lines in the main steam systems do get exposed to transient loads, they are designed for these loads.

Most participants identified thermal fatigue, mechanical fatigue, and primary water stress corrosion cracking, as the important degradation mechanisms to consider in PWR piping. The concerns associated with thermal and mechanical fatigue in PWR plants are similar to those previously discussed for BWR plants. The locations susceptible to thermal fatigue in PWR plants include the surge line which is subject to cyclic thermal stratification stresses. Also, as identified in Table 6.1, concerns were expressed about the direct volume injection (DVI) and chemical volume and control system (CVCS) lines due to periodic testing of these lines which imposes additional thermal cycles. The DVI and accumulator safety injection system (SIS) lines have also experienced some thermal fatigue cracking due to cold water leakage past the check valves.

Primary water stress corrosion cracking (PWSCC) is a relatively new mechanism that has manifested itself in U.S. plants over the last 5 years. It has many similar characteristics to previous IGSCC cracking in BWR reactors. It is a temperature dependent mechanism that attacks Alloy 600 type base materials and Inconel 82/182 welds. Many panel members believe that the PWSCC problems in PWRs will be resolved (i.e., mitigated) over the next 15 years. Therefore, its contribution to the overall LOCA frequencies may peak sometime over the next several years, but then decrease after that. To date, instances of PWSCC in piping systems have been observed in surge lines at the surge line-to-pressurizer weld in the United States at Three Mile Island, as well as in plants in Belgium and Japan. The hot legs have also experienced PWSCC at the hot leg-to-reactor pressure vessel weld in the United States at the V.C. Summer plant and in Sweden at the Ringhals plant. Other piping systems where PWSCC could surface in the future are the cold leg and the pressurizer spray lines. However, since the cold leg and many of the BWR systems operate at slightly lower temperatures than the hot legs and surge lines in PWRs, problems in these systems may not materialize until later in the operating life of the plant.

Many of the same degradation mechanisms existing for piping are important for non-piping components as well. For example, PWSCC, as described above for piping systems, is prevalent in smaller Alloy 600 components, such as steam generator tubes, CRDMs and other penetrations. Similarly, thermal fatigue is an important consideration at nozzle inlets and other locations which experience thermal stratification. However, there are also several degradation mechanisms that are only a concern in certain non-piping components. These include radiation embrittlement which reduces the fracture toughness of the RPV, especially in PWRs where there is less shielding. In addition, steam generator tubes, which were generally cited as a major small break LOCA contributor in PWRs, are susceptible to a variety of unique degradation mechanisms, including fretting and wear and denting from secondary side contamination.

There was also more concern expressed about the possibility of common cause failures in non-piping components. Multiple steam generator tube or CRDM failures could result from the failure of a single component due to the proximity of other components. Multiple steam generator tubes can also fail due to a sudden secondary side pressure drop if multiple tubes are sufficiently degraded. Also, bolting failures are only expected to lead to a LOCA if multiple bolts fail due to common causes, such as improper installation and inspection, or the emergence of degradation mechanisms such as steam cutting or boric acid corrosion which affect multiple bolts.

6.3.3 Effect of Component Size on LOCA Frequencies

The panelists generally believe that smaller LOCAs are more likely than larger LOCAs in piping because small piping has a greater failure propensity. As previously mentioned (Section 6.3.1), the panelists generally believe that complete rupture of a smaller pipe is more likely than an equivalent size opening in a larger pipe. A given size crack is a larger percentage of the pipe circumference in the smaller diameter line, but cracking length and likelihood is not expected to vary with piping size. Smaller piping is also often subject to fabrication flaws which exacerbates this decreased failure margin. Additionally, smaller diameter lines are often fabricated from socket welded pipe which has a history of mechanical fatigue damage from plant vibrations and is also susceptible to external failure mechanisms arising from human error (e.g., damage from equipment). Finally, small piping is typically more difficult to inspect and in-service inspection (ISI) is not routinely performed on these lines. In contrast, the larger diameter lines are inspected more rigorously and routinely and quality control/quality assurance programs are more stringent as piping size increases. One outcome of the increased failure propensity of smaller diameter lines is that the industry is experienced in dealing with failures, repair, and replacement of the smallest diameter lines such that the mitigation of small pipe damage is relatively routine.

A number of the panelists expressed the similar opinions that the smallest applicable non-piping component will contribute the greatest risk to each LOCA category and that smaller non-piping LOCAs are much more likely. As discussed previously, there are a number of LOCA-sensitive small diameter non-piping PWR components, i.e., steam generator tubes, pressurizer heater sleeves, and CRDMs, that have all experienced either damage or failure in service. Therefore, these components should provide the largest PWR Category 1 and 2 LOCA frequency contributions. In contrast, some larger non-piping components (e.g., vessels) are fabricated with more stringent controls while other larger non-piping components (e.g., valve and pump bodies) have bigger design margins than pipes. Hence non-piping failures are generally expected to be less likely than piping failures for the larger LOCA Categories. The main residual concern with large non-piping components is that inspections are difficult and not usually performed on these components. Therefore, there is more uncertainty about the level of degradation in these components.

A number of panelists believe that aging degradation may have the greatest effect on intermediate diameter (6 to 14-inch diameter) piping systems because of the small and large component trends previously discussed. As discussed, the industry has a greater tendency to replace the smallest diameter piping as problems arise and the failure rates associated with these components are likely to be unchanged in the future. Conversely, the largest piping has increased leak-before-break margin, and is subject to the highest quality inspection programs due to the high failure consequences associated with this piping. Furthermore, there is a large number of piping systems containing intermediate size piping with an associated large number of degradation-sensitive welds. Therefore, the greater future LOCA concerns are often associated with intermediate diameter piping.

6.3.4 Influence of Mitigation and Maintenance Actions

Timely and proper maintenance and mitigation practices were almost always considered by the panelists to result in decreased LOCA susceptibility. Practices such as improved ISI techniques (e.g., eddy current inspection programs for steam generator tubes), greater use of risk-informed ISI, new materials, and pipe and component replacement programs were all mentioned as having a positive influence on the estimated LOCA frequencies. Both steam generator and RPV head replacement programs were cited as effective strategies that the industry is undertaking for reducing LOCA susceptibility. Also, secondary side water chemistry improvement programs were mentioned as positive measures for decreasing steam generator tube degradation.

However, there are several potential detrimental aspects associated with any mitigation or maintenance practices. For one, the frequent opening and closing of systems for inspections can cause problems. There is also the increased likelihood for human error if components are reassembled improperly or artifacts (e.g., tools and loose debris) are left within the component after servicing. Also improper maintenance activities associated with active system components (such as a valve) could cause increased loading in a passive system component, and thus greater failure likelihood.

Although ISI programs are clearly beneficial, a number of participants expressed reservations about the viability of ISI programs for small diameter piping systems, for piping systems and components fabricated from cast stainless steels, and for welds with limited accessibility for inspections. Inspection of smaller piping may not be as viable because degradation can initiate and lead to failure in some instances even within inspection cycles. The concern with cast stainless steel components (i.e., pump and valve bodies) is that there still is no proven technique for inspecting these materials. In addition, these materials are subject to thermal aging which can reduce the fracture toughness. Fortunately, no service-induced crack initiation mechanism has been identified for these materials or else this issue would become much more serious. Limited accessibility locations are a concern because inspection quality may suffer or an inspection may not even be possible. An example of a limited accessibility location is the surge line-to-pressurizer bimetallic weld in Westinghouse PWRs. This particular weld also has experienced service cracking and is particularly susceptible to PWSCC due to the high operating temperature of this location.

As the industry moves toward greater risk-informed inspection (RI-ISI) criteria, many panelists expressed caution with this approach. The philosophy of inspecting risk-important locations is generally regarded to be sound. However, inspection locations are largely based on experience. As a result, it is imperative that early signs of degradation in new locations are aggressively addressed and incorporated into RI-ISI programs. Some panelists are concerned about relying too heavily on risk-informed regulation and eschewing good engineering practices (i.e., proactive maintenance, inspection, and material replacement programs) if the associated risk are expected to be small.

There were a few maintenance and mitigation issues discussed that are unique to non-piping components. There was some concern expressed that maintenance and inspection programs of the large component bodies (pressurizer, steam generator, RPV) is not as rigorous as for piping systems. For smaller non-piping components such as steam generator tubes and CRDMs, improved inspection methods and mitigation programs are expected to reduce future failure frequencies. For steam generators, improved eddy current techniques have been incorporated in inspection programs and more degradation resistant materials are replacing older materials. Also, as mentioned above, better water chemistry control is possible for the secondary side of the steam generator. Similarly for CRDMs, ongoing head replacement programs with more resistant Alloy 690 materials and better and periodic inspection programs are being followed to minimize the occurrence of further PWSCC cracking.

6.3.5 Effect of Operating Time on LOCA Frequencies

Unabated aging mechanisms typically lead LOCA frequency increases with continued operating time. However, nearly all panelists expect that the NRC and industry will aggressively respond to emerging mechanisms. Some historical examples cited to support this contention include the IGSCC cracking issue in BWR plants in the late 1970s and early 1980s and the PWSCC issue currently being addressed in PWR plants. Overall, the participants believe that maintenance and mitigation will offset the natural tendency for LOCA frequencies to increase due to aging. Most of the participants think that the various compensating factors will balance such that the LOCA frequencies will remain relatively constant over the future operating period, up to 35 years from today.

Furthermore, there are several factors which will promote LOCA frequency decreases in the future. A number of the panelists expect that inspection techniques will continue to improve and result in greater knowledge of existing plant degradation and allow degradation evolution to be more accurately monitored. This will allow the effectiveness of mitigation strategies to be more effectively evaluated. Other mitigation strategies are expected to improve and mature as well. In addition, the infant mortality rates typically associated with new systems and technologies have already occurred in the service history. Thus, the fabrication-related defect failure rates should continually decrease in the future.

Conversely, the panelists identified several factors associated with degradation mechanisms which promote future LOCA frequency increases. Thermal fatigue, SCC, and FAC challenges are all expected to increase. Thermal fatigue issues will increase near the end of the life extension period from the high usage factors at many plant locations. One panelist expressed concern about future IGSCC failure rates from potential sulfate increases resulting from infrequent ion-exchange filter replacement. Also, over the next 10 to 15 years, a number of the participants believe that PWR LOCA frequencies will increase until appropriate PWSCC mitigation strategies have been developed and implemented. Once these strategies are in place, then the associated LOCA frequencies would naturally diminish. Panelists also tempered their expectations about possible future LOCA decreases based on the possibility of new degradation mechanisms emerging. While no specific mechanisms were identified, the service history experience has shown that previously unseen mechanisms have periodically surfaced in the fleet. The expectation of many panelists is that new mechanisms will continue to arise periodically and present new LOCA challenges.

6.3.6 Uncertainty in LOCA Frequency Estimates

The panel members generally expressed greater uncertainty in their predictions as the LOCA size (or category) increases and as the future operating time period increases. These trends are natural because assessment of larger and more distant LOCAs requires greater extrapolation of existing service experience data. There was also generally more uncertainty associated with the PWR LOCA frequency assessments. One reason is that a number of panelists believe that the uncertainty associated with BWR piping is less than with PWR piping. The important BWR LOCA issues largely remain IGSCC in the future, but the BWR plants have more experience mitigating these degradation effects. Similar experience does not exist for PWR plants, which leads to greater uncertainty about the possible effects of degradation. However, there is also an expectation that PWR plants will gain experience from addressing current PWSCC piping concerns. At this point in time, the associated PWR and BWR uncertainties may again be similar. Another reason for increased PWR uncertainty is that PWR plants contain more LOCA-sensitive non-piping components. Therefore, the uncertainty associated with assessing their LOCA frequency contributions is naturally greater.

6.4 References

6.1 Snaider, R., "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking," NUREG-0619, April 1980.

6.2 Generic Letter 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping," January 25, 1988.

6.3 Hazelton, W. S., "Technical Report on Material Selection and Processing Guidelines for BWR Cooling Pressure Boundary Piping," NUREG-0313, Rev. 2, June 1986.

7. QUANTITATIVE RESULTS

The analysis procedures of Section 5 were applied to process the results from each panelist and aggregate the individual results to generate LOCA frequency estimates which reflect the estimates of the entire panel. As described in Section 5.3.1, each panelist's input (Section 5.2) is processed using the assumed split lognormal structure of each response to obtain frequency estimates for each contributing piping system and non-piping sub-component. These individual contributors are then summed (Section 5.3.4) to obtain total LOCA estimates for each panelist corresponding to BWR and/or PWR plant types as appropriate. The individual estimates are then aggregated as described in Section 5.4 to estimate central group opinion and evaluate panel diversity.

This section summarizes the LOCA frequencies resulting from the expert elicitation process and the chosen analysis procedure. The detailed quantitative responses to the elicitation questions from each of the individual members of the elicitation panel are available with this report at: http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1829. The effects of safety culture on the LOCA frequencies are described in Section 7.1. Then, the total LOCA frequencies are documented along with the relative piping and non-piping contributions and the effect of operating time in Sections 7.2 – 7.4. While the elicitation structure defined the LOCA sizes by leak rate categories, the LOCA categories are correlated to break sizes as described in Section 3.7 and the results are presented with respect to break size in Section 7.2. Correlating the LOCA flow rate categories to physically equivalent break diameters is helpful for identifying major systems and components contributors. Nevertheless, the results for each applicable LOCA category include contributions for all the variously-sized systems and components.

Individual panelist uncertainties and the variability among the panelists' responses are discussed (Section 7.5). In addition, numerous sensitivity analyses were performed to examine the robustness of the analysis techniques and understand the impact of various analysis choices on the calculated LOCA frequencies. The results of these analyses are presented in Section 7.6. The results are then compared to the LOCA frequency estimates developed in previous studies (Section 7.7). These sections generally provide an overall synopsis of the results. Appendix L (Detailed Results) provides additional detailed results and discussion, including the LOCA frequency estimates for each panelist. The general approach and philosophy used by each panelist in developing these estimates is provided in Appendix K (General Approach and Philosophy of Each Expert).

The quantitative estimates presented and discussed herein are supported by the qualitative technical insights in Section 6. These qualitative insights are generally not replicated in this section, although it is necessary to understand them to evaluate the quantitative results and trends. Specifically, the underlying damage mechanisms (Section 6.3.2) are fundamental to the quantitative BWR and PWR LOCA frequencies. The Section 6 location of other supporting qualitative rationale is referenced whenever possible in this section. Many figures in this section contain straight lines connecting the LOCA categories. These lines exist for clarity to indicate trending and do not represent a physical relationship between LOCA categories assessed in the elicitation.

7.1 Safety Culture

Figures 7.1 and 7.2 illustrate the effect of the industry and regulatory safety culture, respectively, on the ratio of the LOCA frequency in the future to the LOCA frequency at 25 years (i.e., LOCA

Ratio). The Category 1 LOCA Ratio results are shown using box and whisker plots of the responses[1]. Similarly, Figure 7.3 shows the effect of the industry safety culture on the LOCA Ratio for Category 4 LOCAs. Only the Category 4 LOCA results are shown because they are representative of the results for all larger LOCA categories. Qualitative information provided by the panelists supporting the results in these figures is provided in Section 6.2.

In these figures, LOCA Ratios less than 1.0 indicate an expected LOCA frequency reduction as a result of safety culture improvements. The panel members generally expect (Figures 7.1 – 7.3) that the safety culture will either improve slightly, or remain constant, up to the end of the license extension period. The interquartile regions (area represented by the box) for all plots span are bounded by 0.5 and approximately 1.0. In fact, the median response in all these figures is 1.0. The implication is any future LOCA frequency reduction is expected by the majority of the panel to be less than a factor of two. There is also nearly universal agreement that larger LOCA frequencies will not be altered by future safety culture changes. Only the smaller LOCA break sizes are expected by some panelists to benefit from an improved safety culture. Nearly all panelists also expressed the opinion (Section 6.2) that the industry and regulatory safety culture are highly positive correlated. Therefore, regulatory and industry changes in safety cultures are not separable and will vary similarly in the future.

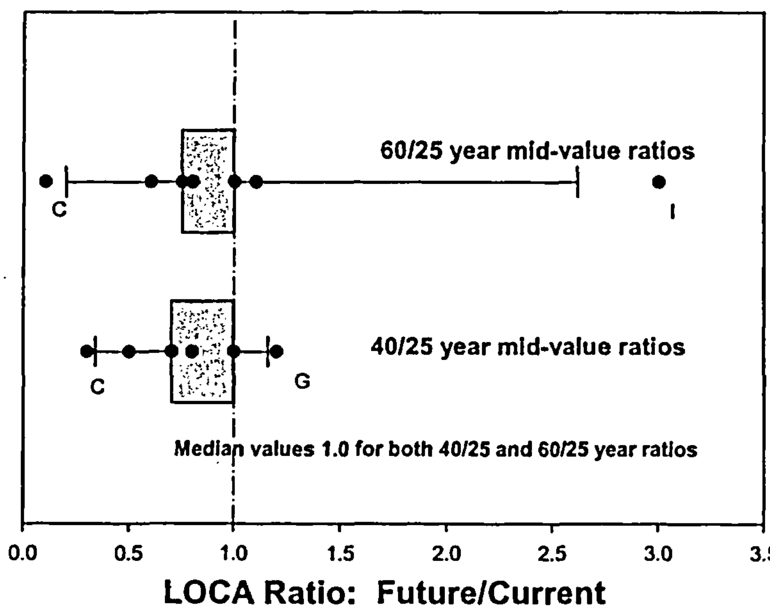

Figure 7.1 Effect of Utility Safety Culture on Category 1 LOCAs

[1] For a discussion of how to interpret the box and whisker plots in Figures 7.1 through 7.3, the reader is referred to Appendix L.

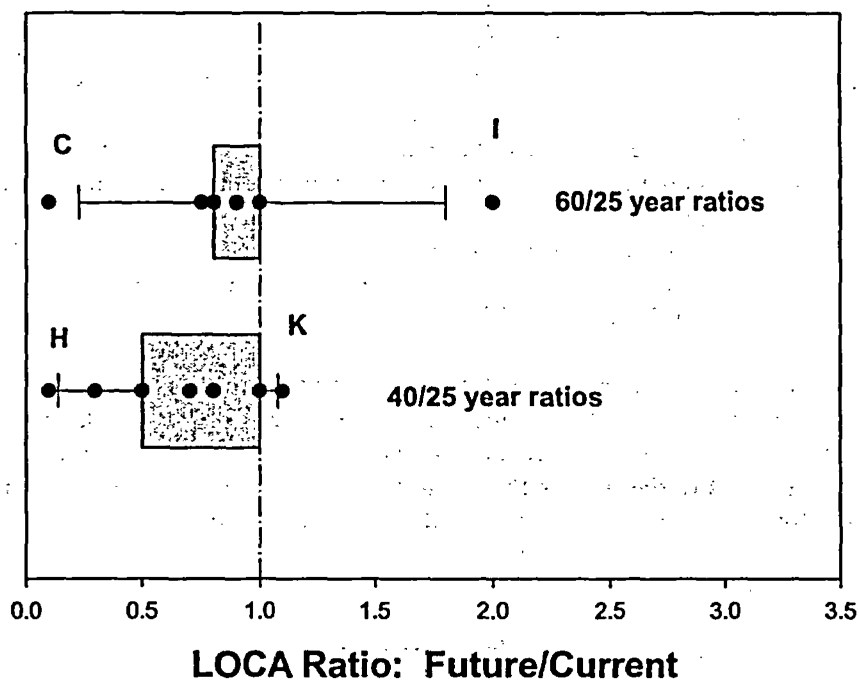

Figure 7.2 Effect of Regulatory Safety Culture on Category 1 LOCAs

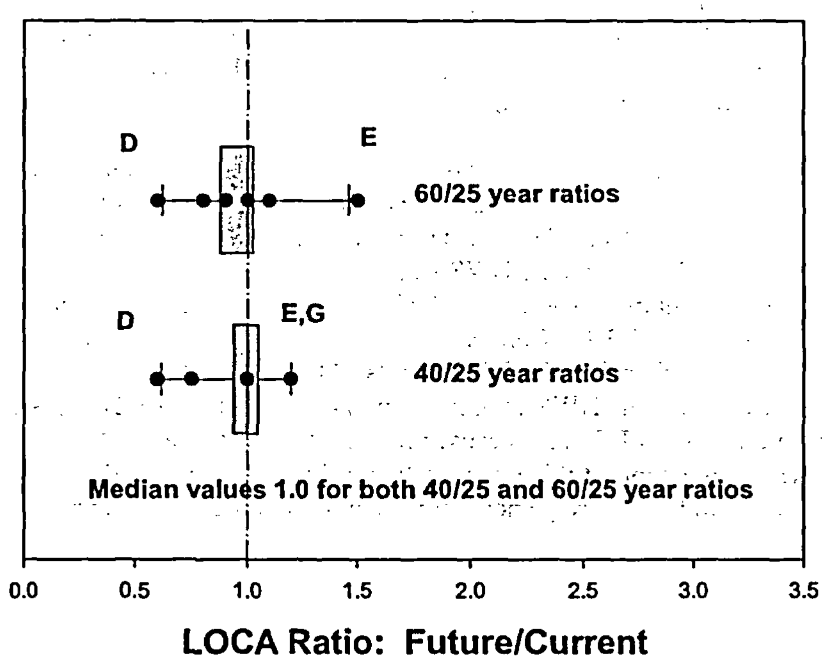

Figure 7.3 Effect of Utility Safety Culture on Category 4 LOCAs

However, there are a few concerns that are counter to the expectation that the safety culture of the entire industry is generally expected to be stable. First, many panelists are concerned that LOCA frequencies can be significantly increased by the operating philosophy of a single safety-deficient plant (Section 6.2). The vessel head degradation that occurred at the Davis-Besse plant was frequently cited as an example of this effect (Section 6.2). Several panelists expressed that safety culture deficiencies at a single plant could increase the LOCA frequencies at that plant by a factor of 10 or more. The other concern frequently expressed was that the safety culture may deteriorate near the end of the plant's operating life as the economic incentive to invest in necessary maintenance activities vanishes. Also, staff morale may erode from looming employment losses (Section 6.2). These concerns did cause some panelists to predict higher LOCA Ratios near the end of the current license extension period (i.e., the 60/25 year ratios) when compared with the 40/25 year results in Figures 7.1 through 7.3.

In summary, the two principal conclusions from the safety culture elicitation questions are (1) safety culture effects on future LOCA frequencies are expected to be minimal, and (2) utility and regulatory safety culture are highly correlated. **Because of these findings, no modification or adjustment was applied to determine the quantitative LOCA frequencies presented subsequently.** However, the panelists expressed the need for continued vigilance to ensure that deficient safety culture does not exist at individual plants or that safety culture does not erode during possible decommissioning.

7.2 Total BWR and PWR LOCA Frequencies

The total (piping plus non-piping) BWR and PWR passive system LOCA frequencies are provided in Table 7.1. The medians, means, 5th (5th Per.) and 95th percentiles (95th Per.) are the geometric means of the panel members' total BWR and PWR LOCA frequency estimates as described in Section 5.3.5. Estimates are provided for the current day frequencies (25 year average plant life) and frequencies at the end of the current licensing period (40 year average plant life). Generally, the 95th percentiles for both the BWR and PWR plants are between a factor of 2 and 4 higher than the mean values. The LOCA flow rate threshold associated with each LOCA category is also provided in Table 7.1.

Figure 7.4 is a graphical representation of the results from Table 7.1. Included in this figure are the mean and 95th percentile current day (25 years of plant operations) LOCA frequency estimates for both BWR and PWR plants. As illustrated, the PWR Category 1 results are approximately 11 times higher than the BWR Category 1 results. This reflects the contribution of the steam generator tube rupture experience which is the dominant LOCA contributor for PWR Category 1 LOCAs (Section 6.3.1). Conversely, for the Category 5 LOCAs, the BWR results are approximately 11 times higher than the PWR results. This reflects the continued concern of the participants with IGSCC for the BWR recirculation system (Section 6.3.2). This concern disappears for the Category 6 LOCAs because the largest diameter recirculation system piping (28-inch diameter) cannot support this largest category LOCA. For the other LOCA categories, the total LOCA BWR and PWR frequencies are generally comparable and vary by a factor of 5 or less.

Table 7.1 Total BWR and PWR LOCA Frequencies

Plant Type	LOCA Size (GPM)	Eff. Break Size (inch)	Current Day Estimate (per cal. year) (25 yr fleet average operation)				Estimate at End of Plant License (per cal. yr.) (40 yr fleet average operation)			
			5th Per.	Median	Mean	95th Per.	5th Per.	Median	Mean	95th Per.
BWR	>100	½	4.0E-05	3.0E-04	5.5E-04	1.6E-03	3.2E-05	2.6E-04	5.2E-04	1.5E-03
	>1,500	1 7/8	3.6E-06	4.8E-05	1.0E-04	3.2E-04	3.0E-06	4.4E-05	9.7E-05	3.1E-04
	>5,000	3 ¼	7.4E-07	9.7E-06	2.4E-05	7.9E-05	6.7E-07	9.8E-06	2.7E-05	8.8E-05
	>25K	7	1.2E-07	2.2E-06	6.1E-06	2.0E-05	1.1E-07	2.3E-06	7.6E-06	2.4E-05
	>100K	18	1.2E-08	2.9E-07	1.1E-06	3.7E-06	1.1E-08	3.1E-07	1.5E-06	4.6E-06
	>500K	41	9.7E-12	3.0E-10	3.2E-09	7.7E-09	1.2E-11	4.0E-10	4.9E-09	1.1E-08
PWR	>100	½	7.9E-04	3.7E-03	5.9E-03	1.5E-02	4.8E-04	2.5E-03	4.3E-03	1.2E-02
	>1,500	1 5/8	9.9E-06	1.4E-04	4.6E-04	1.4E-03	1.0E-05	1.6E-04	5.6E-04	1.8E-03
	>5,000	3	2.6E-07	3.4E-06	1.3E-05	4.2E-05	5.9E-07	7.6E-06	2.8E-05	9.3E-05
	>25K	7	1.9E-08	3.1E-07	1.2E-06	3.9E-06	3.8E-08	6.5E-07	2.7E-06	8.9E-06
	>100K	14	5.3E-10	1.1E-08	9.9E-08	2.6E-07	1.3E-09	2.7E-08	2.5E-07	6.5E-07
	>500K	31	4.5E-11	1.2E-09	1.5E-08	3.8E-08	1.2E-10	2.9E-09	3.8E-08	9.5E-08

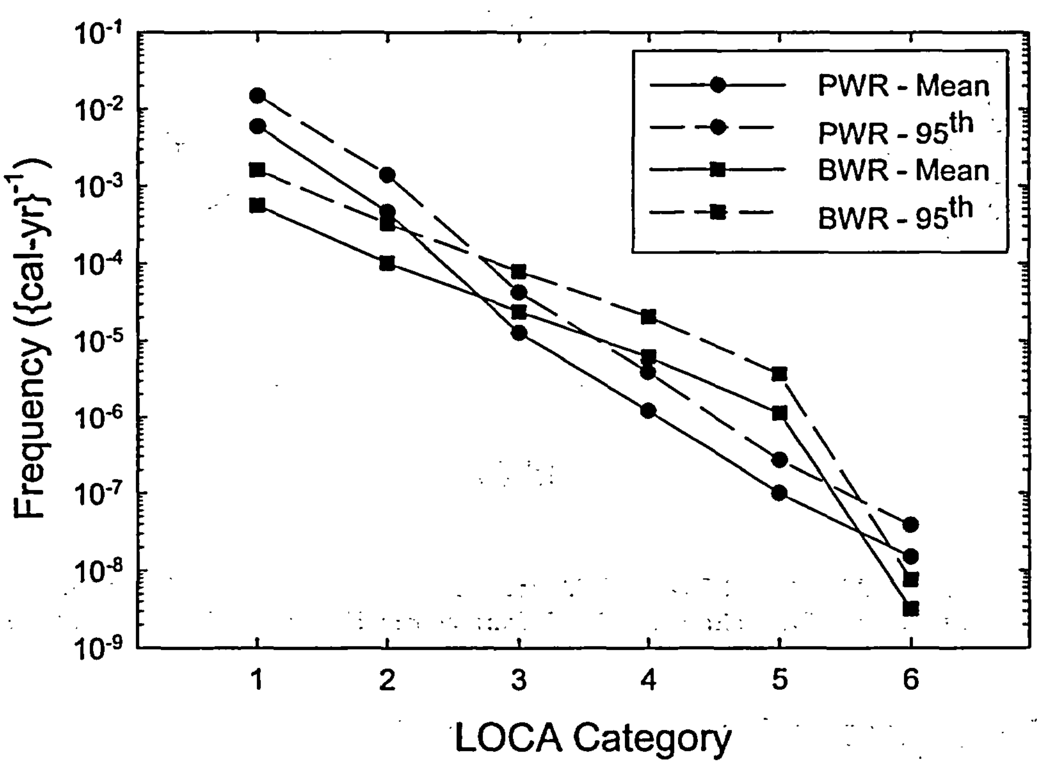

Figure 7.4 Total BWR and PWR LOCA Frequencies (Means and 95th Percentile Values) as a Function of LOCA Category at 25 Years of Plant Operations

Table 7.1 also lists the minimum effective break size corresponding to each LOCA flow rate category using the correlations described in Section 3.7. As described in Section 3.7, the break size represents the equivalent diameter of an associated break area required to result in each of the six threshold flow rates. However, these break sizes do not represent actual pipe diameters. Specifically, there is no 41 inch diameter (Table 7.1) piping in BWR plants. Only the PWR and BWR liquid correlations (Section 3.7) were used in developing the Table 7.1 break size equivalents because the major BWR contributions come from the liquid lines. The main steam line is generally not expected to be a significant contributor to the overall LOCA frequencies. It is worth stressing that most panel members did utilize the effective break size in assessing failure rates for each LOCA category instead of the flow rate definitions. The LOCA frequencies are illustrated as a function of the threshold break diameter in Figure 7.5 for both the PWR and BWR mean and 95th percentile estimates. The minimum break diameter and LOCA category relationship is nearly logarithmic and there is approximately a factor of two difference in each successive minimum break size. Therefore, Figure 7.4 and 7.5 are nearly identical.

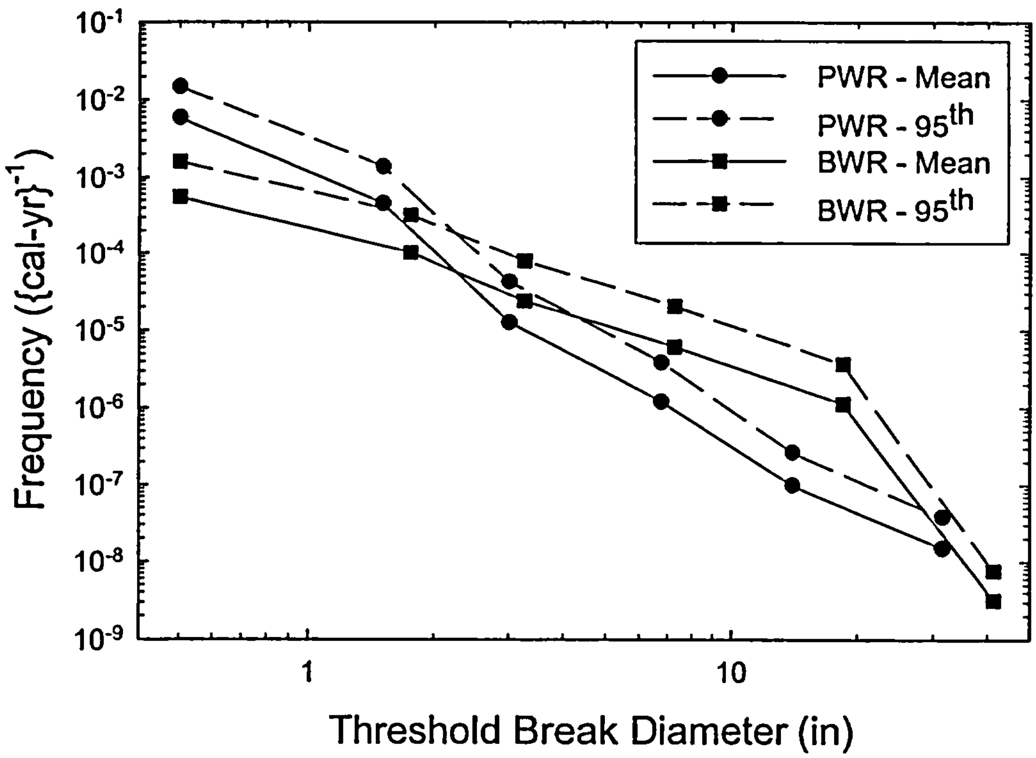

Figure 7.5 Total BWR and PWR LOCA Frequencies (Means and 95th Percentile Values) as a Function of the Threshold Break Diameter at 25 Years of Plant Operations

7.3 Comparison of Piping with Non-Piping Contributions to Total LOCA Frequencies

The relative piping and non-piping current-day mean LOCA frequencies estimates are compared in Figures 7.6 and 7.7 for BWR and PWR plants, respectively. The quantitative differences between the piping and non-piping contributions are summarized in Table 7.2. For the BWR plants, the piping contributions to the Category 1 and 2 LOCA frequencies are a factor of 2 to 5 higher than the contribution from the non-piping components. The BWR primary piping system contributors to these smaller category LOCAs are the small diameter instrument and drain lines

(Section 6.3.1). The main BWR non-piping contributor to these smaller category LOCAs is from vessel penetrations, especially the lower head control rod drive housings (Section 6.3.1), due to current PWSCC concerns (Section 6.3.2).

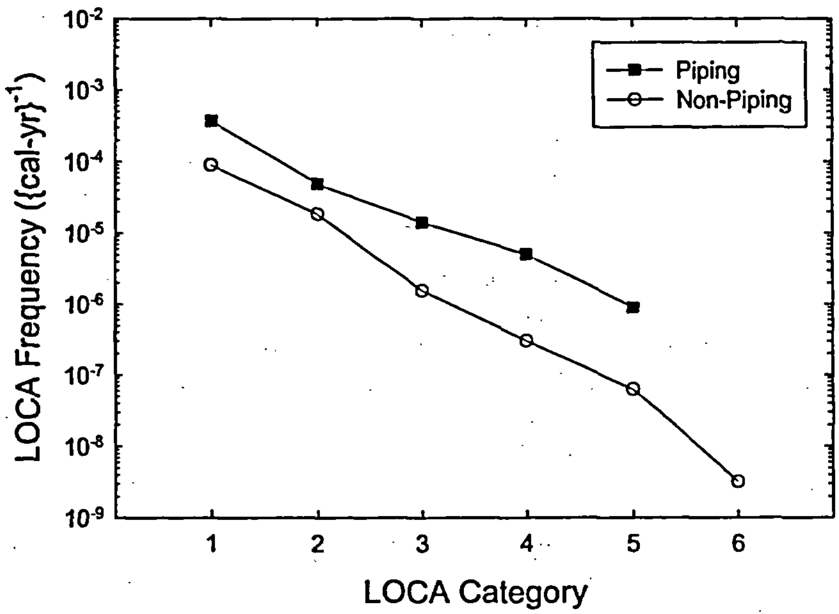

Figure 7.6 BWR Piping and Non-Piping Comparison (Current Day Estimates)

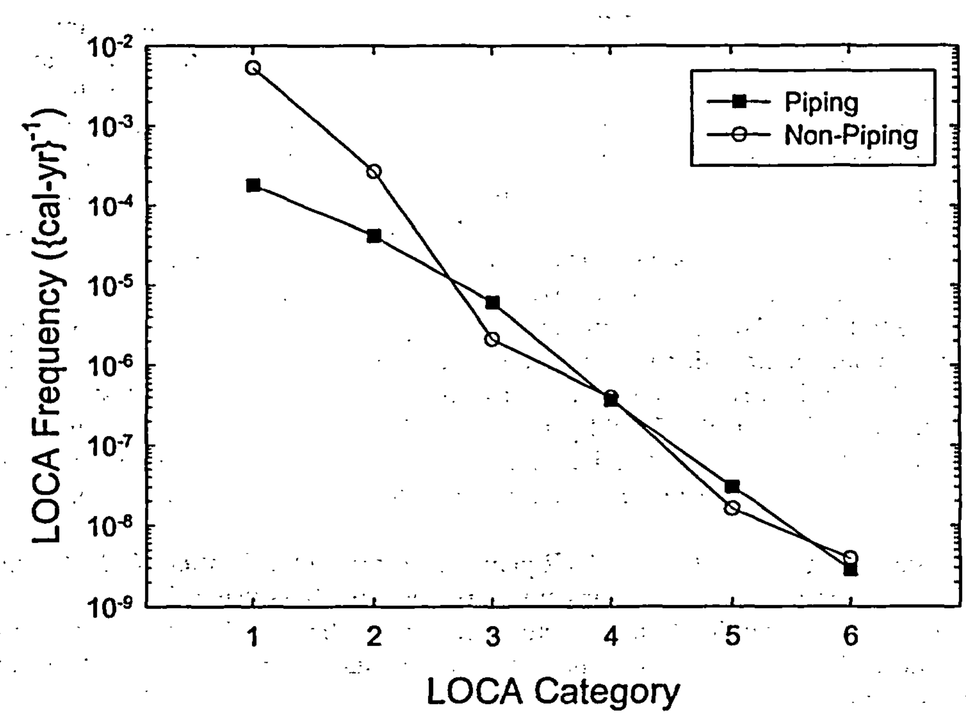

Figure 7.7 PWR Piping and Non-Piping Comparison (Current Day Estimates)

Table 7.2 Ratio of BWR and PWR Non-Piping Contribution to Piping Contribution for Each LOCA Category

LOCA Category	Ratio of BWR Non-Piping Contribution to Piping Contribution	Ratio of PWR Non-Piping Contribution to Piping Contribution
1	0.2	29
2	0.4	7
3	0.1	0.3
4	0.1	1.1
5	0.1	0.5
6	NA	1.4

The major BWR piping system contributor to LOCA Categories 3 - 5 is the recirculation system piping (Section 6.3.1) where IGSCC concerns remain (Section 6.3.2). The panel consensus is that the susceptibility to IGSCC is greatly reduced compared to the past (Section 6.3.2). Mitigation measures such as improved hydrogen water chemistry (HWC), weld overlay repairs, post-weld heat treatment (PWHT), and pipe replacement using more resistant materials have effectively reduced the likelihood of IGSCC. However, there is still residual concern about the failure likelihood of the large recirculation piping material that has not been replaced. The main BWR non-piping contributors to LOCA Categories 3 – 5 are the RPV nozzles and the component bodies (Section 6.3.1). However, the BWR non-piping contributions to the total LOCA frequencies diminish substantially due to presumed robustness of the non-piping components as balanced against the IGSCC recirculation piping concerns. The relative non-piping contribution for all the LOCA categories is generally expected to remain relatively constant in the future, although mitigation of vessel penetration cracking may decrease this specific contribution over the next fifteen years (Section 6.3.4).

The PWR LOCA frequencies (Figure 7.7 and Table 7.2) are dominated by the non-piping contributions for LOCA Categories 1 and 2. The mean non-piping LOCA frequencies are 29 and 7 times greater than the piping contribution for Category 1 and 2 LOCAs, respectively. The major piping contributors for PWR Category 1 and 2 LOCAs are, as was the case for the BWR plants, the instrument and drain lines. The concerns with these systems are similar as in BWR plants (Section 6.3.1). However, the relative non-piping contributions are so much higher for PWR plants because of plant design differences and the increased population of non-piping primary pressure boundary components. The PWR plants operate at higher temperatures and several non-piping components (e.g. pressurizer, steam generator) have experienced service degradation due to PWSCC or actual rupture (e.g. steam generator tubes). For LOCA Category 1, the large non-piping contribution is provided by steam generator tubes, CRDM penetrations, and pressurizer heater sleeves degradation concerns (Section 6.3.1). For LOCA Category 2, the principal contributing factor is the CRDM penetrations (Section 6.3.1). The non-piping contributions for LOCA Categories 1 and 2 are expected to decrease somewhat in the future as the PWSCC degradation is mitigated and as steam generator replacement programs proceed.

The relative non-piping contribution for PWR plants drops precipitously after LOCA Category 2, and the piping and non-piping contributions are nearly the same (less than a factor of 3 difference) for LOCA Categories 3 - 6. The steam generator tube, CRDM, and pressurizer heater sleeve components do not substantially contribute anymore due to size limitations. The major PWR piping system contributions include various auxiliary piping systems for LOCA Category 4; the hot leg, surge line and RHR lines for Category 5; and the hot leg for Category 6 (Section 6.3.1). The major PWR non-piping contributors are nozzles and component bodies for LOCA

Categories 3 and 4; the manways and component bodies for Category 5; and the component bodies for Category 6.

The steam generator tube rupture frequencies are normally separate from other passive system failures in probabilistic risk assessment analyses because they have occurred relatively frequently. As explained in Section 4, an average steam generator rupture frequency for flow rates greater than 100 gpm (380 lpm) of 3.5E-03 $\{\text{cal-yr}\}^{-1}$ has been determined based on operating experience data. This estimate was provided to the panelists for use as a base case. The portion of the total LOCA frequencies (Table 7.2) attributed to steam generator tube ruptures is summarized in Table 7.3. These steam generator tube frequencies have been aggregated using the geometric mean of the individual estimates. This practice is consistent with the total LOCA frequency estimate aggregation discussed in Section 5.4. Category 1 LOCAs are possible with a single tube rupture. Other higher category LOCAs require multiple tube ruptures due to common cause failure. The assessment of multiple tube ruptures was specifically required during the elicitation. The number of panelists who provided steam generator tube ruptures estimates for each LOCA category (# of est.) is also provided in Table 7.3. Only LOCA categories where rupture frequency estimates were provided by a minimum of three panelists are included in this table.

All the panelists who provided PWR elicitation responses identified steam generator tube rupture as a major contributor to Category 1 PWR LOCAs. In fact, the mean and median steam generator frequencies are greater than 55% of the total LOCA frequency estimates in Table 7.1. It is interesting that the mean current day Category 1 steam generator tube rupture frequency estimates are nearly identical to the base case frequency (Section 4.4.1). Therefore, the panelists typically did not believe that these frequencies needed additional adjustment. As mentioned in Section 4.4.1, this frequency is consistent with prior NUREG/CR-5750 estimates. The current day 95th percentile estimates (Table 7.3) are also less than the mean steam generator leak frequency (Section 4.4.1) of 16×10^{-3} $\{\text{cal-yr}\}^{-1}$ which should provide a reasonable bound.

The steam generator contribution to LOCA Category 2 decreases greatly and supplies less than 3% of the mean contribution to the total in Table 7.1. Three panelists provided LOCA Category 3 estimates. Interestingly, the mean Category 3 steam generator tube frequency estimate does not decrease significantly from the Category 2 estimate, and is approximately 40% of the total LOCA frequency estimate. The implication is that the relative importance of steam generator tube ruptures increases again in the opinion of these three panelists. These panelists indicate that multiple tube rupture due to common cause failures is nearly just as likely to result in a Category 3 LOCA as it is a Category 2 LOCA. The aggregated results (Table 7.3) also predict a slight overall decrease in the steam generator tube rupture frequencies over the next fifteen years. This decrease occurs due to expected steam generator replacement and improved degradation mitigation programs.

Table 7.3 PWR Steam Generator Tube Rupture Frequencies

LOCA Size (GPM)	# of Est.	Eff. Break Size (inch)	Current Day Estimate (per cal. year) (25 yr fleet average operation)				Next 15 Year Estimate (per cal. yr.) (End of original license)			
			5th	Median	Mean	95th	5th	Median	Mean	95th
>100	9	½	8.6E-04	2.6E-03	3.4E-03	8.2E-03	4.7E-04	1.5E-03	1.9E-03	4.7E-03
>1,500	7	1 5/8	1.2E-06	4.8E-06	1.1E-05	3.5E-05	4.8E-07	2.3E-06	5.2E-06	1.6E-05
>5,000	3	3	3.4E-07	2.9E-06	5.3E-06	1.8E-05	2.0E-07	2.3E-06	3.6E-06	1.2E-05

7.4 Effect of Plant Operating Time on Total LOCA Frequencies

There is very little difference between the 25 and 40 year estimates (Table 7.1). The 40-year PWR median and mean estimates are less than a factor of 2.6 greater than the 25-year estimates for all LOCA categories. The LOCA Category 1 mean frequencies are expected to actually decrease slightly (by 30%) due to expected improved mitigation practices for steam generator tube ruptures and CRDM cracking (Section 6.3.4). The modest PWR LOCA frequency increases are largely attributed to continuing PWSCC concerns in other components. The 40 year BWR LOCA mean frequencies are even more stable, and differ by less than 30% from the 25 year estimates for LOCA Categories 1 - 5. This stability reflects the opinion that IGSCC (Section 6.3.2) susceptibility is expected to remain constant in the near term. The BWR mean Category 6 LOCA frequency only increases by 50% over the next fifteen years. This minor increase results from increased RPV failure rupture frequencies due to continued radiation embrittlement (Section 6.3.1).

The panelists expect bigger LOCA frequency differences between the 60 and 25 year estimates. The 25 and 60 year mean frequency estimates are illustrated in Figure 7.8. The BWR differences between the 25 and 60 year estimates are smaller than the PWR differences. In fact, for LOCA Categories 1 – 5, the median BWR LOCA frequency 25 and 60 year estimates differ by less than 15%. The BWR mean frequency increases (Figure 7.8) are largely attributable to increases in the 95^{th} percentile estimates supplied by each expert. These increases are largely due to uncertainty about the future and the concern that new mechanisms could arise in the operating fleet (Section 6.3.4). The only significant frequency increase is predicted to result from continued RPV aging and only affects the BWR LOCA Category 6 frequency (Section 6.3.1). Even this increase results in less than a factor of 10 increase in the current day Category 6 LOCA frequency estimates over the next 35 years.

PWR mean and median frequencies increase more substantially at 60 years. Ratios of the 60 to 25 year median frequency estimates are less than a factor of 3 for LOCA Categories 1 and 2, a factor of 6 for LOCA Categories 3 and 4, and a factor of 10 to 13 for LOCA Categories 5 and 6. Similar increases are predicted at 60 years for the 95^{th} percentile estimates. The increases in the median and 95^{th} percentile estimates have a more significant impact on the mean frequencies given the skewed nature of these distributions. The PWR mean frequencies (Figure 7.8) increase by a factor of less than 4 for LOCA Categories 1 and 2, 8 to 10 for LOCA Categories 3 and 4, and 15 to 18 for LOCA Categories 5 and 6. Therefore, larger LOCA sizes are expected to be impacted more severely by possible future mechanisms.

The result (Figure 7.8) is that at 60 years, the PWR mean frequencies are expected to be greater than the BWR frequencies for LOCA Categories 1, 2, 3, and 6 and slightly less than the BWR frequencies for LOCA Categories 4 and 5. This general frequency increase reflects greater concern about the uncertainty surrounding the mitigation effectiveness of the current PWSCC problem and the effect of future degradation mechanisms on PWR LOCA frequencies compared with BWR plants. While no specific new degradation mechanism has been identified for PWR plants, this increase reflects an underlying sense that these plants are more susceptible than BWR plants to future degradation.

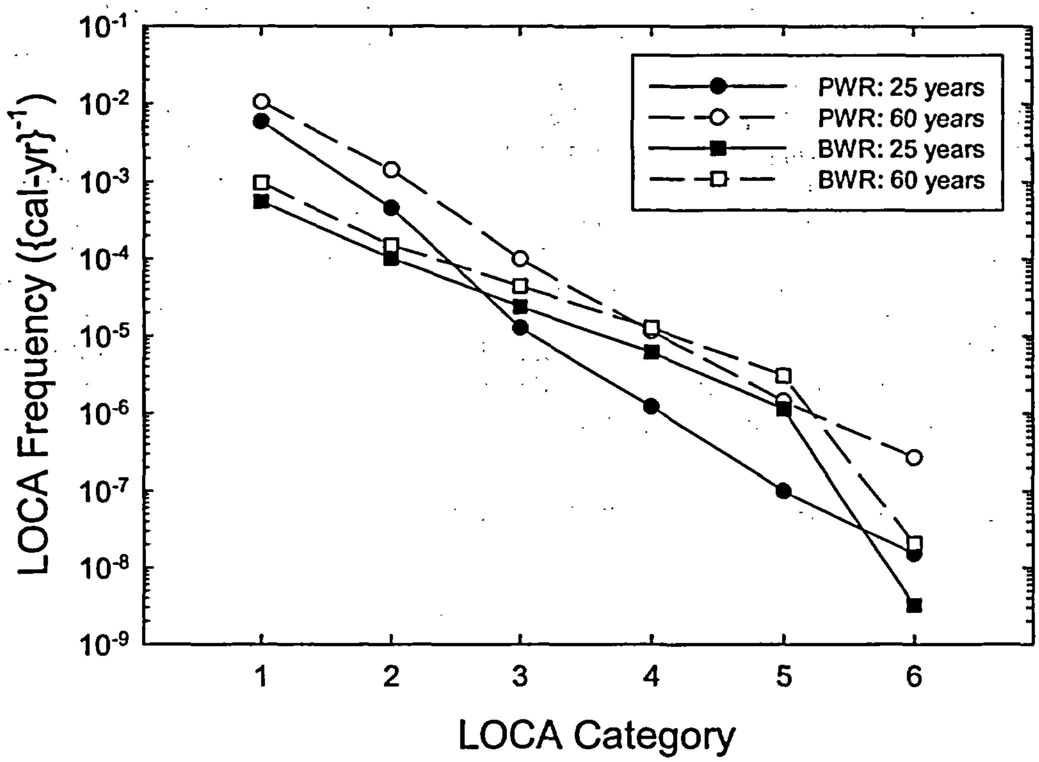

Figure 7.8 Comparison of BWR and PWR Group Combined LOCA Frequencies at 25 and 60 Years

7.5 Panel Member Uncertainty and Group Variability

7.5.1 Panel Member Uncertainty

The BWR Category 1 and 3 LOCA frequencies are illustrated in Figures 7.9 and 7.10, respectively, for the various participants who answered the BWR elicitation questions. Figures 7.11 and 7.12 are similar plots for the PWR Category 1 and 6 LOCA frequencies, respectively. In each of these plots, the individual panel members' medians and 5th and 95^{th} percentile estimates are shown. The letter designators by the horizontal lines in these plots are the reference letters which identify each panelist. For the BWR plants (Figures 7.9 and 7.10), the uncertainty associated with most of the panel member's responses (i.e., the difference between their 5^{th} and 95^{th} percentiles) were comparable for their Category 1 and 3 responses. The exceptions to this are Panelists A, E, and F who all expressed almost twice as much uncertainty with their Category 3 responses as they did in their Category 1 responses. The uncertainties in the larger LOCAs, Categories 4 – 6 continue to increase with successively larger LOCA sizes. This is evident in the individual error factor which is defined as the UB/MV responses (Sections 5.6.2.2 and 7.7.2.2) and is a measure of panelist uncertainty. The geometric mean of the individual error factors increases from a factor of about 7 for LOCA Category 1 to 30 for Category 6. However, the differences in relative uncertainty among the individual panelists remain consistent to the LOCA Category 3 (Figure 7.10) trends. That is, the ratio between any two expert's error factors is relatively constant for LOCA Categories 3 through 6.

The panelist PWR LOCA frequency estimates (Figures 7.11 and 7.12) exhibit many similar features to the BWR estimates. As before, the participants express more uncertainty as the LOCA size increases. For the PWR Category 1 responses (Figure 7.11), the panelist's uncertainty varied from less than 1 order of magnitude for Panelists A, B, and L to 2 orders of magnitude for Panelist H. This relative high level of certainty for LOCA Category 1 reflects the fact that steam generator rupture is a significant contributor to this LOCA category and there is actual rupture data to support the estimates. Conversely, for the PWR Category 6 LOCAs, the level of uncertainty ranges from approximately two order of magnitude for Participants A, B, H, and L to greater than four orders of magnitude for Participants C, E, and J. This level of uncertainty is a reflection of the fact that more uncertainty is naturally related to very low frequency events, like a Category 6 LOCA (Section 6.3.6). The mean error factors vary from approximately 5 to 26 for Categories 1 and 6 respectively (see Section 7.6.2.2). This is comparable to the increase for the BWR results. The most drastic increases are for Participants C and J who expressed three orders of magnitude more uncertainty in their Category 6 responses than they did in their Category 1 responses.

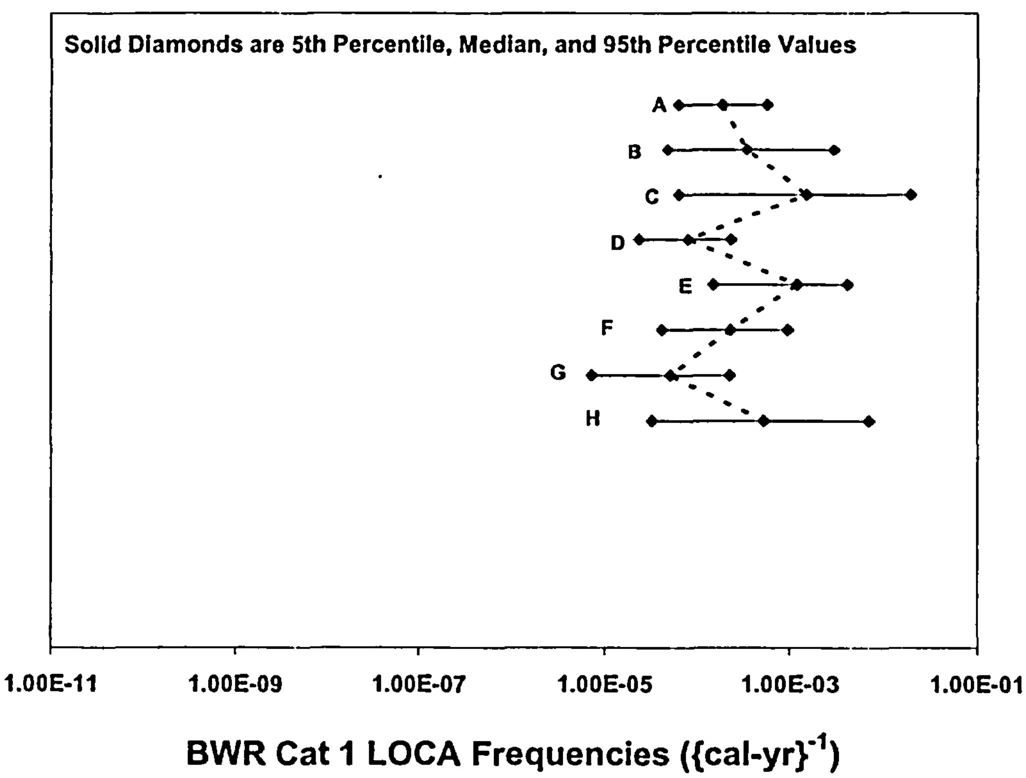

Figure 7.9 Individual Panel Members Estimates of BWR Category 1 LOCA Frequencies Showing Their Mid-Value Estimates as Well as Their 5 and 95% Bounds

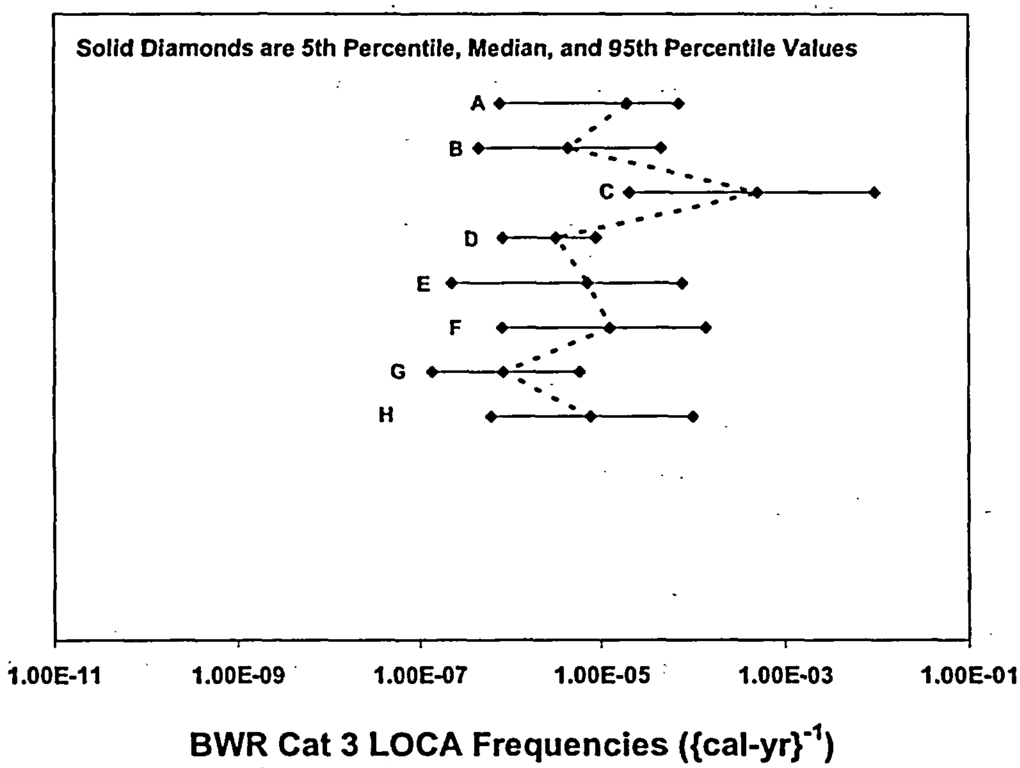

Figure 7.10 Individual Panel Members Estimates of BWR Category 3 LOCA Frequencies Showing Their Mid-Value Estimates as Well as Their 5 and 95% Bounds

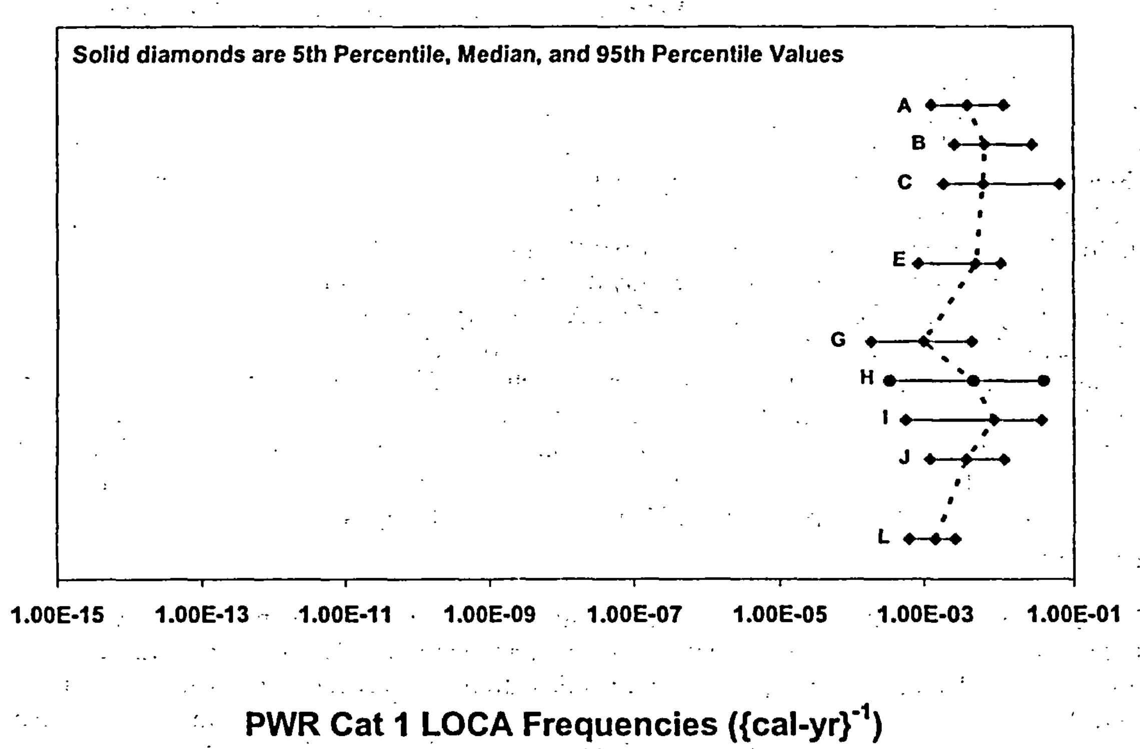

Figure 7.11 Individual Panel Members Estimates of PWR Category 1 LOCA Frequencies Showing Their Mid-Value Estimates as Well as Their 5 and 95% Bounds

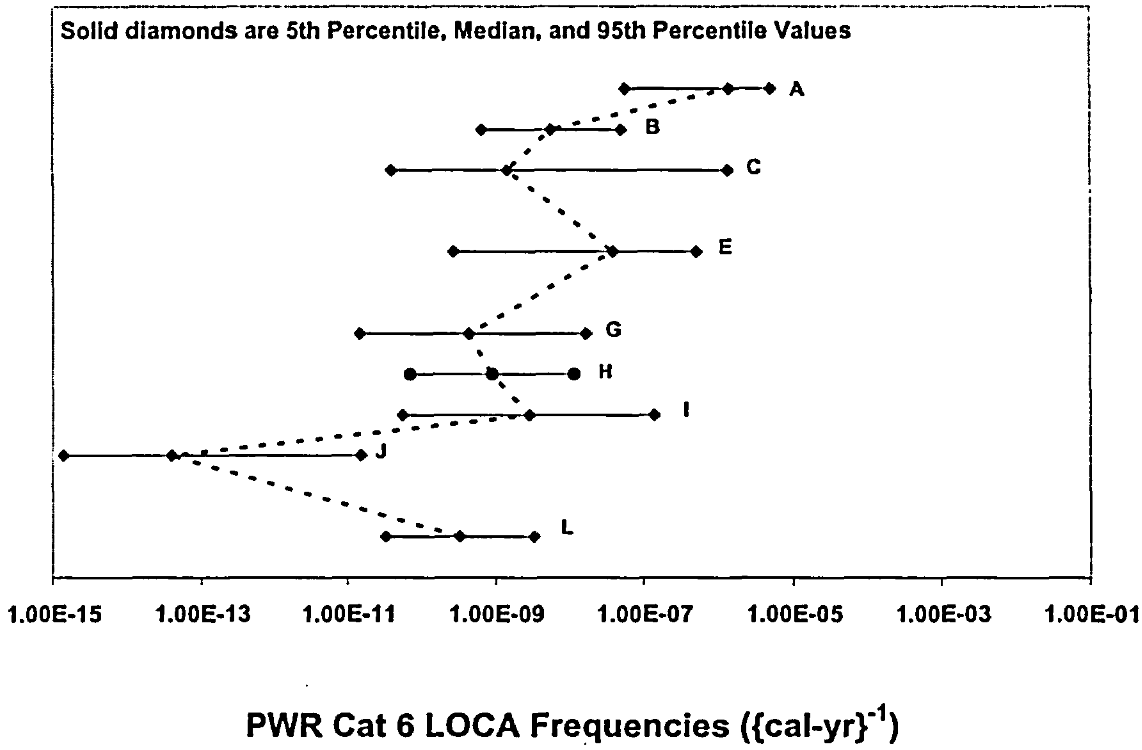

Figure 7.12 Individual Panel Members Estimates of PWR Category 6 LOCA Frequencies Showing Their Mid-Value Estimates as Well as Their 5 and 95% Bounds

7.5.2 Group Variability

There is considerable variability among the panelists' opinions. There is approximately a factor of 30 difference between the lowest and highest BWR Category 1 median frequency estimates (Figure 7.9). This difference is relatively small compared to the factor of 600 difference in the medians between the extreme estimates (Panelists C and G) for BWR LOCA Category 3 (Figure 7.10). Panelist C also provides the highest BWR estimates for LOCA Categories 1 – 4. However, Panelist C's estimates increasingly diverge from the rest of the group for each successively higher LOCA category. For LOCA Category 4, the difference between Panelist C and the minimum estimate (Panelist G) is a factor of over 1,700. Additionally, Panelist C's mid-value response is higher than many of the other panelist's 95th percentile estimates. The reason for Panelist C's disparity stems from the expectation that IGSCC is nearly as likely to result in a Category 4 LOCA as it is in a Category 1 LOCA in the primary recirculation piping. While other panelists largely agree that IGSCC of the main recirculation piping is important (Section 6.3.1), the expectation that Category 1 and 4 LOCAs are nearly as likely is not shared by the other panelists.

There is more variability in the 95th percentile estimates than in the medians for LOCA Categories 1 - 4. For instance, while the minimum and maximum Category 1 mid-value estimates vary by a factor of 30, the 95th percentile extreme values vary by a factor of 80. For LOCA Categories 5 and 6, there is less disparity among the 95th percentile estimates than either the mid-value or 5th percentile estimates. This trend occurs because all panelists expect a much lower frequency associated with the rarest large piping and non-piping failures and there is simply more disagreement about how low these frequencies could be. Conversely, the 5th

percentile estimates exhibit the least disparity for LOCA Categories 1 – 4 and the greatest differences for Categories 5 and 6.

The PWR, like the BWR, results also generally exhibit increasing variability as the LOCA size increases. The minimum and maximum median Category 1 estimates vary by only a factor of 8 (Figure 7.11). This good quantitative agreement is due to the fact that nearly all panelists expect steam generator tube rupture to be the dominant contributor (Section 6.3.1). However, differences between the minimum and maximum panelists' median estimates increase to between 3 and 4 orders of magnitude for LOCA Categories 2 – 4. While this difference is comparable to the BWR estimates, the overall variance associated with these PWR estimates is greater. Only Panelist C's BWR estimates are significantly different than rest of the group while the PWR estimates are more uniformly spread over the range. This greater variance reflects the greater disagreement in the contributing factors to PWR LOCAs of this size (Sections 6.3.1 and 6.3.2).

For the Category 5 and 6 LOCAs (Figure 7.12), the differences between the minimum and maximum panelist estimates are greater than 5 and 7 orders of magnitude, respectively. This primarily results from the low frequencies expected by Panelist J. Panelist J's predicted 95th percentile estimate for LOCA Category 6 is equivalent to or less than the 5th percentile estimates for all of the other panelists (Figure 7.12). If Panelist J's responses are discarded, the disparity among the maximum and minimum median estimates for the remaining panelists decreases significantly to a factor of 750 and 4000 for LOCA Categories 5 and 6 respectively. This reduced LOCA Category 5 range also becomes consistent with the LOCA Categories 2 – 4 ranges. The LOCA Category 6 range is not significantly higher. Panelist A's PWR LOCA Category 6 median value also appears to be an outlier (Figure 7.12) as it is greater than the 95th percentiles of the remaining panelists. Eliminating Panelist A's estimates as well decreases the range of median estimates to a factor of only 100. For LOCA Categories 5 and 6, the quantitative differences among the panelists mainly result from the uncertainty associated with estimating rare events (Section 6.3.6) and not any significant differences in the contributing factors (Sections 6.3.1 and 6.3.2). Panelist J simply predicts that large component failure is much less likely while Panelist A expects it is much more likely than the other panelists.

The variability among the 5th and 95th percentile estimates displays similar trends as the BWR estimates when Panelist J's estimates are discarded. The minimum and maximum 95th percentile estimates exhibit the greatest range for LOCA Categories 1 – 2, while the 5th percentile and median estimate variability is much less. However, for LOCA Categories 5 – 6, the 5th percentile and median estimates differ by the greatest amount while the 95th percentile estimates are less variable. The variability among all the percentiles is similar for LOCA Categories 3 and 4. When Panelist J's results are also considered, the 5th percentile estimates exhibit the greatest variability for LOCA Categories 2 through 6.

It is also interesting to note that almost all of the Category 1 LOCA estimates (Figure 7.9 and 7.11) are fairly symmetric. Therefore, the median response falls nearly in the middle of the coverage interval. The implication is that the panelists expect that the true LOCA mean frequencies are equally likely to be a fixed ratio above or below the median. However, a few of the participants exhibit significantly skewed distributions for the higher LOCA categories. For instance, Panelists' A and E median estimates are closer to the 95th percentiles for both BWR Category 3 (Figure 7.10) and PWR Category 6 (Figure 7.12) LOCAs. This implies that these experts expect that the actual LOCA frequencies could only be a little higher, but significantly lower then their median estimates. Conversely, the Category 6 PWR LOCA responses for Participants C and J have median values closer to the 5th percentile and substantial LOCA frequency increases above the medians would not be surprising to these panelists.

7.5.3 Confidence Intervals

The statistical confidence intervals associated with the total LOCA frequency estimates (Table 7.1, Figures 7.4 and 7.5) are presented in Figures 7.13 and 7.14 for BWR and PWR plants, respectively. The median, mean, and 95th percentiles that have been aggregated from the individual panelists' responses (Section 5.4) are illustrated along with the associated 95% confidence intervals. The confidence intervals have been calculated as described in Section 5.4 and are identified by the error bars about the central estimate data point in each figure. The confidence intervals are a measure of the group diversity of opinion associated with each of the base-line total LOCA frequency parameters. That is, they reflect the variability among the panelists of the median, mean and 95th percentile estimates. Based on the assumed lognormal model for the individual estimates, each confidence interval has a 95% chance of covering the median (estimated by the geometric mean of the individual estimates) of the assumed lognormal distribution. The variability measured by the confidence intervals is distinct from the uncertainty expressed by individual panelists as reflected in the difference between the aggregated median and 95th percentile estimates.

The confidence intervals are symmetric about the aggregated group estimates as a result of the assumed lognormal distribution of the panelist estimates (Section 5.4) and they generally increase with increasing break size as would be expected given the difficulties in estimating infrequent events. This reflects the increasing variability within the entire group as discussed in the previous section. The confidence intervals span less than an order of magnitude for the BWR and PWR Category 1 LOCA estimates. The BWR confidence interval differences then increase to a factor of approximately 10 for LOCA Categories 2, 3, and 5, a factor of 30 for LOCA Category 4, and over two orders of magnitude for LOCA Category 6. The PWR confidence interval differences increase to a factor of approximately 20 for LOCA Categories 2-4, and over two orders of magnitude for LOCA Categories 5 and 6. The differences between the BWR and PWR confidence intervals and the trends with increasing break size are similar to the previously discussed trends on panel diversity (Section 7.5.2)

The difference between the median and 95th percentile aggregated curves also increases with break size. This trend and the magnitude of this increase is consistent with the trends among the individual panelists' uncertainties previously discussed (Section 7.5.1). Additionally, the mean values also become closer to the 95th percentile with increasing break size which also reflects the increased individual uncertainty and the assumed lognormal distribution of the underlying individual estimates.

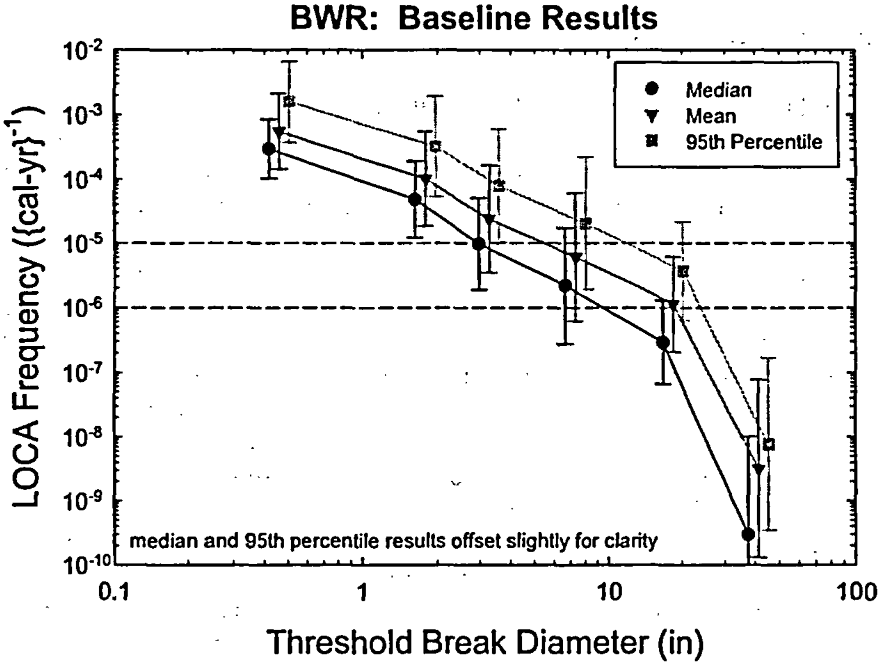

Figure 7.13 Total BWR LOCA Frequencies (Means, Medians, and 95[th] Percentile Values) and Associated 95% Confidence Intervals as a Function of the Threshold Break Diameter at 25 Years of Plant Operations

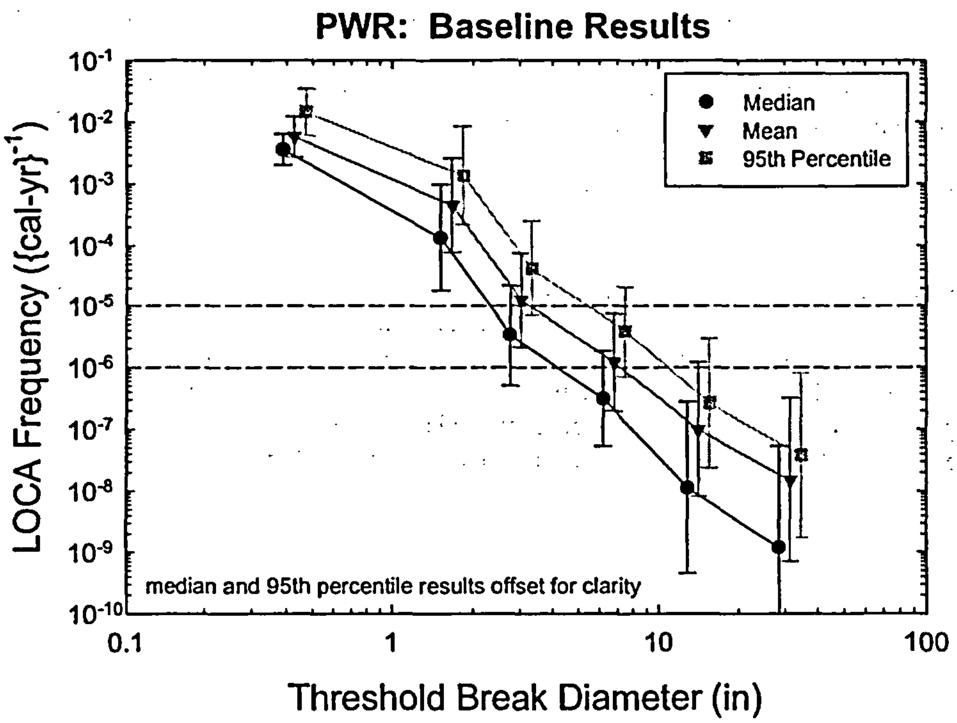

Figure 7.14 Total PWR LOCA Frequencies (Means, Medians, and 95[th] Percentile Values) and Associated 95% Confidence Intervals as a Function of the Threshold Break Diameter at 25 Years of Plant Operations

7.6 Sensitivity Analyses

There are several types of sensitivity analyses that have been conducted to examine the assumptions, structure, and techniques used to process the elicitation responses to develop the baseline LOCA frequency estimates. The characteristics of the baseline estimates have previously been defined (Section 5.5), but they are summarized here for convenience. There are six key elements of the baseline analysis framework:

(i) The mid-values, upper bounds and lower bounds supplied by the panelists for each elicitation question are assumed to correspond to the median, 95th percentile, and 5th percentile, respectively, of a split lognormal distribution with the means calculated assuming the upper tail is truncated at the 99.9th percentile.

(ii) Panelist responses are not adjusted to account for possible overconfidence in the uncertainty estimates for each elicitation response.

(iii) Split lognormal distributions are summed by assuming perfect rank correlation among the individual terms.

(iv) Aggregation of the individual estimates into a single group estimate is performed using the total LOCA frequency estimates determined for each panelist.

(v) The group estimate of the total LOCA frequency parameters (i.e., median, mean, 5th percentile, and 95th percentile) is defined using the geometric mean of the individual estimates.

(vi) Panel diversity is characterized by using a two-sided 95% confidence interval based on an assumed lognormal model for the individual estimates.

These baseline results have previously been presented and discussed in Sections 7.2 – 7.5.

This section quantifies the impact that other assumptions or methods of calculation have on the baseline frequency estimates. The sensitivity analyses fall into one of the following five broad categories as discussed in Section 5.6: mean determination, panelist overconfidence adjustment, correlation structure, individual response aggregation, and panel diversity.

In each case, a single change in the baseline analysis procedure is made and the impact is quantified. While many sensitivity analyses could be performed on these results, only representative and important results are contained herein to examine result differences associated with each of the four broad categories. A summary of the sensitivity analyses conducted is provided in Table 7.4 for the five general analysis categories. The corresponding sections containing the approach and results of each analysis is also provided. More details on the approaches used to develop the sensitivity analyses are contained in Section 5.6.

Table 7.4 Sensitivity Analysis Summary Matrix

Analysis Category	Analysis Type	Corresponding Section
Mean Determination	Effect of Distribution Shape on Mean	7.6.1
Overconfidence Adjustment	Blanket Overconfidence Adjustment	7.6.2.1
	Targeted Overconfidence Adjustment	7.6.2.1
	Error Factor Adjustment	7.6.2.2
Correlation Structure	Perfect Rank Correlation vs. Independence	7.6.3
Response Aggregation	Measures of Group Opinion	7.6.4.1
	Aggregation Point	7.6.4.2
	Aggregation Parameters	7.6.4.3
	Mixture Distribution	7.6.4.4
Panel Diversity	Confidence Bound and Quartile Comparison	7.6.5

7.6.1 Mean Determination

Each panel member provided three estimates for each elicitation response: the mid value, lower bound, and upper bound (Section 3.8). These estimates were assumed in the analysis to represent the median, 5th and 95th percentiles of an underlying distribution (Section 5.1). A distribution shape must be assumed, however in order to combine distributions and determine the associated mean value. The multiplicative structure of the elicitation (Section 3) and the character of the elicitation responses suggest that the distributions should have a split lognormal form. The baseline analysis calculated the mean for each individual elicitation response by replacing the corresponding split lognormal by a split lognormal that is truncated at the 99.9th percentile (Section 5.6.1). The remaining 0.1 percent of each distribution's area is concentrated at the 99.9th percentile. In order to retain the relationship between the median and the 95th percentile estimates that were provided for each elicitation response, no normalization of the probability density function (PDF) below the 99.9th percentile was done. This truncation point was chosen for the baseline analysis to be reasonably conservative. Furthermore, because the panel members provided no information beyond the 95th percentile, any truncation point beyond the 95th percentile is consistent with their responses.

This sensitivity analysis analyzes the effects that other distributional shapes have on the calculated means for the individual elicitation responses. There are several other plausible choices as outlined in Section 5.6.1: the split lognormal, the split log-triangular, a simple lognormal with the mean calculated from the upper tail (upper tail lognormal), or a normalized split lognormal truncated at the 99.9th percentile. The effect of the truncation point for the split lognormal was also examined by truncating at the 95th, 96th, 97th, 98th, 99th, and 99.99th percentiles instead of the 99.9th percentile as in the baseline methodology. The mean was calculated for each of these distributions for surrogate median, 5th and 95th percentile estimates corresponding to a median value of 1, a lower error factor of 1000, and upper error factors ranging from 10 to 1200. This surrogate upper error factor range encompasses the full range of actual elicitation responses. The surrogate lower error factor was fixed because it has very little effect on the mean. Any elicitation response can be modeled using these surrogate parameters by simply multiplying by the elicitation response mid value and using the error factor corresponding to the elicitation response upper bound.

The means calculated for selected distributional shapes are provided in Figure 7.15 for responses with upper error factors up to 100. Figure 7.16 illustrates the means calculated for large error factors up to 1200. The split lognormal, upper tail lognormal, and split log-triangular distributions were selected to encompass the range of plausible distributions for calculating the means. The split lognormal truncated at the 98^{th} percentile is displayed because the calculated means are similar to those using the split log-triangular distribution. The normalized split lognormal distribution truncated at the 99.9^{th} percentile is used for comparison with the baseline truncated distribution, which concentrates the highest 0.1 percent of the PDF area at the 99.9^{th} percentile. These figures do not show distributions that are truncated at less than the 98^{th} percentile. Such distributions do not provide appropriate estimates of means.

Some general trends are readily apparent in Figures 7.15 and 7.16. First, the means calculated using either the split lognormal distribution or a traditional lognormal where the error factor is determined by the ratio of the 95^{th} percentile to the median (upper tail lognormal) are virtually identical. (Their plotted curves are superimposed on each other in both figures.) This verifies that the lower halves of these distributions have a minimal effect on the mean. This finding was verified for larger lower bound error factors than 1000 as in Figures 7.15 and 7.16. The lognormal distributions also, as expected, provide upper bound estimates. The effect of normalizing the truncated distribution versus concentrating all the PDF area beyond the truncation point at the truncation point (baseline method) is minimal. Differences between these two truncation schemes are less than 35%, even for error factors of 1200. The baseline truncation method, however, always results in slightly higher calculated means than does normalization. It is also interesting to note that the means calculated using the 98^{th} percentile truncated distribution are very similar to those determined using the split log-triangular distribution. These distributions provide lower bound estimates for all the distributions in Figure 7.15 and 7.16.

The calculated means are within 30% for the distributions illustrated when the error factor is less than 20 (Figure 7.15). Up to error factors of 50, differences remain less than a factor of two. An error factor of 20 bounds almost all of the elicitation responses for LOCA Categories 1 – 3 while an error factor of 50 bounds most of the Category 4 – 6 responses. The responses for a few panel members are associated with error factors above 50, especially for Category 5 and 6 responses, which represent the most severe and rare LOCA events. Some error factors are as large as 1000 for the rarest Category 6 events.

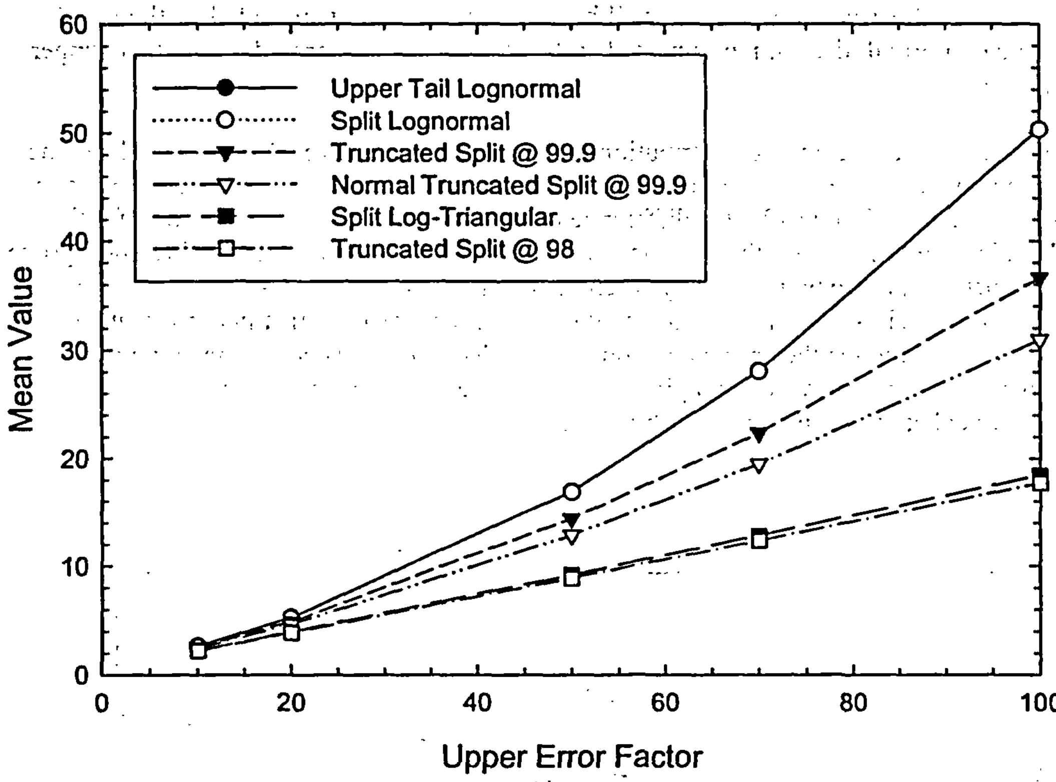

Figure 7.15 Mean Values for Selected Distributions by Upper Error Factor (All Distributions Have Median = 1 and Lower Error Factor = 1000; Baseline is Truncated Split at 99.9%)

The effect of the distribution shape on the mean becomes significant for error factors greater than 100 (Figure 7.16). Beyond this point, the differences among the means for the various distributions increase from a factor of 2.5. At an error factor of 1000, the difference between the split lognormal and split log-triangular means is a factor of 28. The mean for the lognormal truncated at the 99.9th percentile is a factor of 5 less than the split lognormal mean for an error factor of 1000, and is therefore slightly closer to this distribution than the split log-triangular. The choice of the truncation point is also significant for large error factors. The difference in the means between truncating at the 98th or 99.9th percentile when the error factor is 1000 is approximately a factor of 5.5. However, the difference when truncating at the 95th or the 99.99th percentile (not shown) is a factor of 30. The difference between truncating at the 95th percentile and the 99.9th percentile (baseline) methodology is a factor of 15 for this error factor.

The objective in calculating the mean for the elicitation responses was to select a distributional shape that reflects the panelists' responses and also results in a reasonable value for the mean. Because the mean can be a sensitive function of the tail of a distribution, the chosen distribution should not unduly influence the calculated mean. Because the panelists provided no information beyond the 95th percentile, non-truncated lognormal distributions should not be used because their means are overly dependent on the area beyond the 95th percentile for large error factors. This is the rationale for using truncated lognormal distributions. The truncation point should be well beyond the 95th percentile, because truncating the distributions just past the 95th percentile effectively treats the upper bound estimates as absolute bounds. This would be counter to the instructions given to the panelists, which asked them to provide 95th percentiles rather than absolute upper bounds. Truncating at the 99.9th percentile only affects the top 0.1% percent of the PDF. Because this truncation point is well beyond the information provided by the panel members, the lognormal distribution truncated at the 99.9th percentile provides a realistic upper bound for the mean.

Conversely, the split log-triangular distribution provides a realistic lower bound. The difference in the means between these two distribution shapes is only a factor of 5.5 when the error factor is 1000. For error factors below 120, the difference is less than a factor of 2. Therefore, for the bulk of the elicitation responses, the mean is not sensitive to the selection of possible appropriate distributions between these bounds. Only when the error factor is large can the differences become significant. However, the baseline method mean remains realistically conservative for large error factors, which is the primary reason for its selection. This choice appears justified based on this sensitivity study. For simplicity and consistency, the means for all elicitation responses were calculated using this distributional shape.

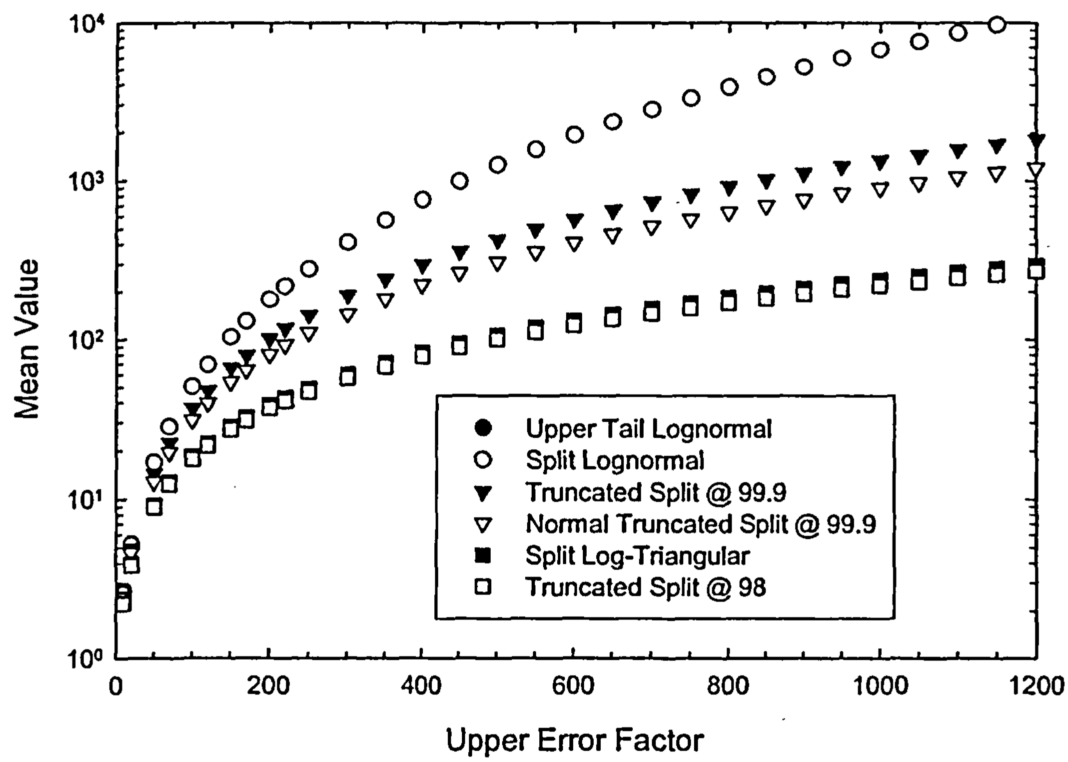

Figure 7.16 Effect of Assumed Distribution Shape on Mean Value for Larger Error Factors: Assumed Distribution with Median = 1 and Lower Error Factor = 1000; Baseline is Truncated Split at 99.9%

7.6.2 Overconfidence Adjustment

Sensitivity analyses were conducted to investigate the effect of assuming that the panelists provided uncertainty intervals with coverage less than the nominal 90% (i.e., the difference between the 5th and 95th percentiles) as requested in the elicitation responses. As noted in Section 5.6.2, nominal 90% coverage intervals provided by experts have been typically shown to have actual coverage between 30% and 70% [7.1]. Two different types of adjustment schemes were used in this analysis. The first scheme adjusted the coverage of the intervals supplied for each individual elicitation question. This is called a coverage interval adjustment. The second scheme adjusted the error factors of the total LOCA frequency estimates calculated for each panelist. This is called an error factor adjustment. Section 5.6.2 has more details on the approach of each adjustment. A summary of the approaches and the results are presented below.

7.6.2.1 Coverage interval adjustments - These overconfidence adjustments evaluated the impacts of modifying the coverage interval ranges associated with each of the panelist's elicitation responses. Nominally, each question required a mid-value, upper bound, and lower bound response. The upper and lower bound responses were associated in the baseline analysis with a 90% coverage interval which is consistent with the instructions provided to the panel. As described in Section 5.6.2.1, the coverage interval adjustments associated the upper and lower bound responses with smaller coverage intervals to examine the quantitative impact of different overconfidence levels. The mid-value responses were not adjusted.

There were five separate coverage interval adjustments performed: two blanket adjustments and three targeted adjustments. The blanket adjustments adjusted all panelists by the same amount. The targeted adjustments used a two-level adjustment. Those panelists expressing larger uncertainty were adjusted by a lesser amount, while those panelists with smaller uncertainty were adjusted by a greater amount. The various overconfidence adjustment schemes are summarized in Table 7.5. The Blanket 1 (B1) adjustment was the most severe adjustment because the responses which nominally represent 90% coverage intervals were assumed to represent only 50% coverage intervals. The Blanket 2 (B2) adjustment provided a smaller correction by assuming that the responses represent 60% coverage intervals. The targeted adjustment uncertainty ranges divided the panelist responses into two groups using an upper error factor (ratio of 95th to median) of 20 for LOCA Category 6 as the demarcation point. The first targeted adjustment (T1) did not adjust the results of the panelists having error factors greater than 20 (4 panelists) and adjusted the coverage interval to 50% for all remaining panelists. The second targeted adjustment (T2) adjusted the more uncertain panelist responses to 80% while the remaining panelists were adjusted to 60%. The third targeted adjustment (T3) is the least severe. The less uncertain panelists were adjusted from a 90% to a 60% coverage interval while the more uncertain panelists were not adjusted.

Table 7.5 Elicitation Response Adjustment Schemes

Overconfidence Adjustment	Coverage Interval 1	Population	Coverage Interval 2	Population
Blanket 1	50%	All	NA	NA
Blanket 2	60%	All	NA	NA
Targeted 1	90%	4	50%	remaining
Targeted 2	80%	4	60%	remaining
Targeted 3	90%	4	60%	remaining

Representative results of this analysis for Category 3 LOCAs in PWR plants are summarized in Figures 7.17 and 7.18. The B1 and T1 corrections are compared to the unadjusted results in Figure 7.17. Several features of the B1 and T1 adjustments become readily apparent. Because the distribution shape is assumed to be a lognormal, these adjustments modify the means more significantly than the 95th percentiles. This is clear when comparing the interquartile widths among the adjusted and unadjusted means and 95th percentile estimates. The mean adjusted widths are markedly bigger than the 95th percentile adjusted widths even though the unadjusted mean and 95th percentile interquartile widths are similar (Figure 7.17). Similarly, the adjustment can greatly skew the distributions. Several of the mean results are higher than the 95th percentile estimates. This skewness is not an accurate characterization of the supportable elicitation uncertainty for this LOCA category. Similar results are apparent for the other PWR LOCA categories and the BWR results.

The blanket adjustment results also suffer from the fact that, for smaller LOCA categories, the adjusted results are not supported by the operating experience data. For instance, the median of the B1 adjusted data is greater than 10^{-3} {cal-yr}$^{-1}$ and several of the frequencies are greater than 10^{-2} {cal-yr}$^{-1}$ (Figure 7.17). Other similar unsupportable results are apparent for other LOCA Categories 1 – 3. The B2 and T2 corrections are less severe, but even the modest adjustment of the coverage interval for those panelists with high uncertainty leads to similar inconsistencies as in the B1 and T1 adjustments.

The T3 correction (Figure 7.18) is the minimal overconfidence adjustment that appears supportable by the elicitation results. For this adjustment, the coverage intervals for those panelists with the least amount of uncertainty (error factor < 20) are only moderately transformed from 90% to 60% (Table 7.5). These are the responses that should be adjusted because the associated uncertainty is less than the group average. The difficulty of these LOCA estimates does not warrant this associated confidence. However, it is difficult to determine the magnitude of the adjustment which is warranted. The 60% coverage interval adjustment is within the wide range supported by research (7.6). The remaining panelists have much greater uncertainty, so there is justification for not adjusting these results.

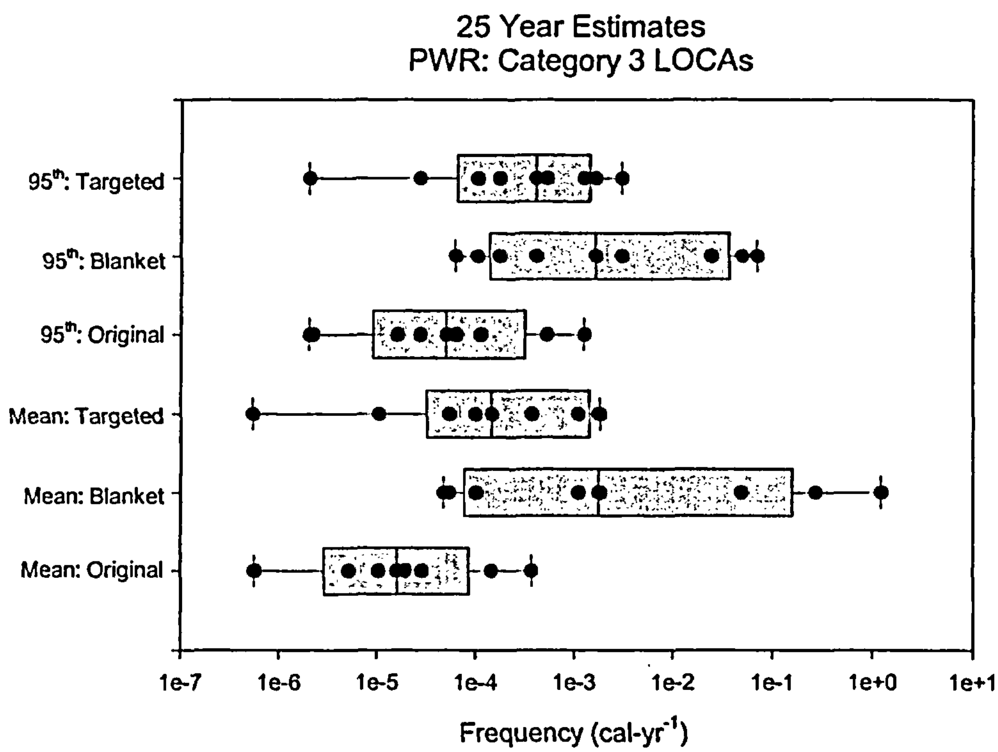

Figure 7.17 Blanket (B1) and Targeted (T1) Elicitation Response Adjustments

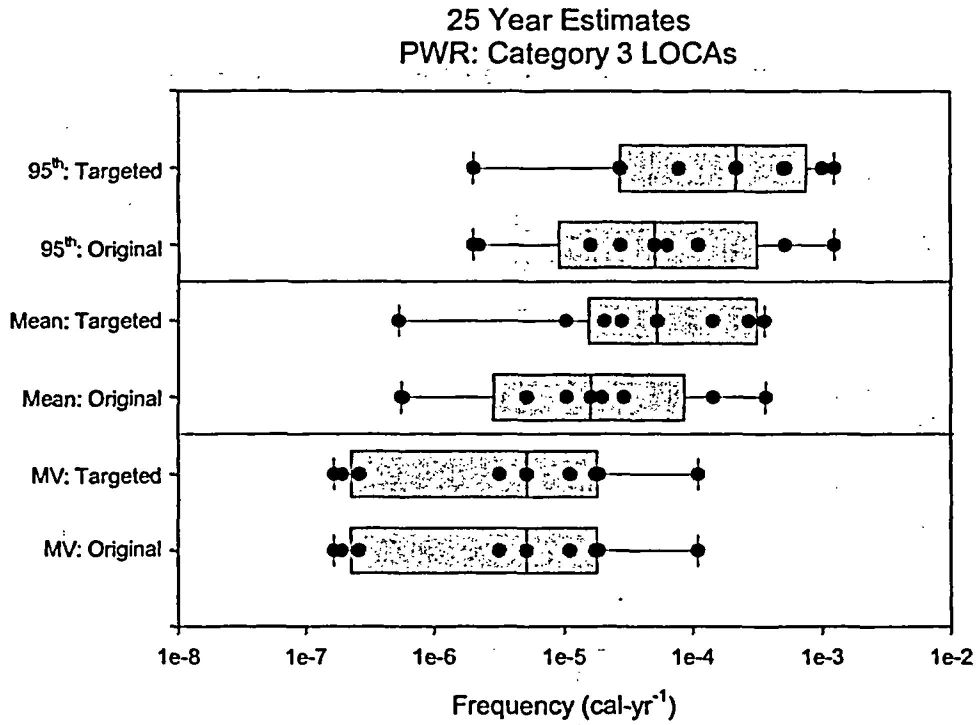

Figure 7.18 Target (T3) Elicitation Response Adjustments Compared to Unadjusted Results

The T3 adjustment results in a modest increase in the median values associated with the mean and 95th percentile estimates (Figure 7.18). Surprisingly, the range between the minimum and maximum responses is unaffected by the correction. It just so happens that the panelists at the extremes did not also have high confidence and their responses are not corrected. The mean and 95th percentile interquartile widths also surprisingly do not vary much from the original, unadjusted responses. What does happen is that the interquartile range for both the mean and 95th percentile estimate is much closer to the maximum value. The interquartile regions for the unadjusted responses are more symmetrically distributed between minimum and maximum responses. The median results (Figure 7.18) are unchanged by the adjustment as previously discussed. These adjusted results, unlike the previous blanket and targeted adjustments, also appear to be physically supportable by the service history, the various predicted base case frequencies, and the elicitation testimony.

The total BWR and PWR LOCA frequencies resulting from the T3 adjustment are illustrated in Figures 7.19 and 7.20. The associated confidence bounds are also shown. A quantitative comparison between the T3 adjustment and baseline results is provided in Table 7.6. The ratio between the targeted (T3) adjustment and the unadjusted results (Table 7.6) illustrates the uniformity of this adjustment. The BWR and PWR means and 95th percentile estimates are only shifted between approximately 2 and 5 times the original results. The mean value is shifted more than the 95th percentile estimate due to the skewness of the underlying distributions, but not dramatically. The PWR results are adjusted slightly more than the BWR results because more PWR panelist responses are adjusted (five vs. four), and the error factors of the unadjusted PWR responses are slightly higher than the unadjusted BWR responses. Hence, a bigger adjustment occurs. Regardless, both the BWR and PWR results are only mildly corrected for overconfidence using the T3 adjustment.

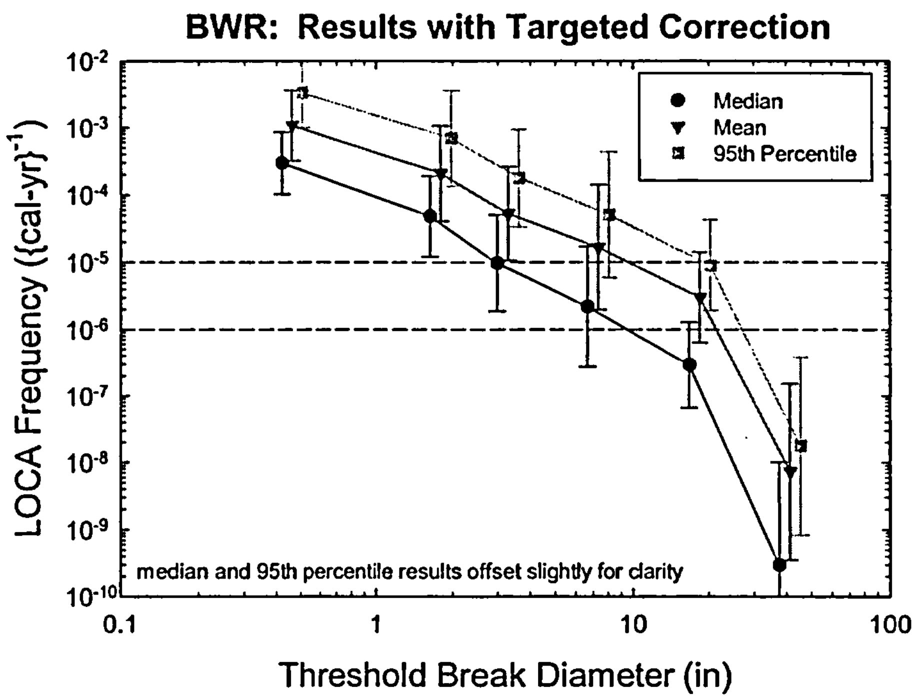

Figure 7.19 BWR LOCA Frequencies with T3 Adjustment

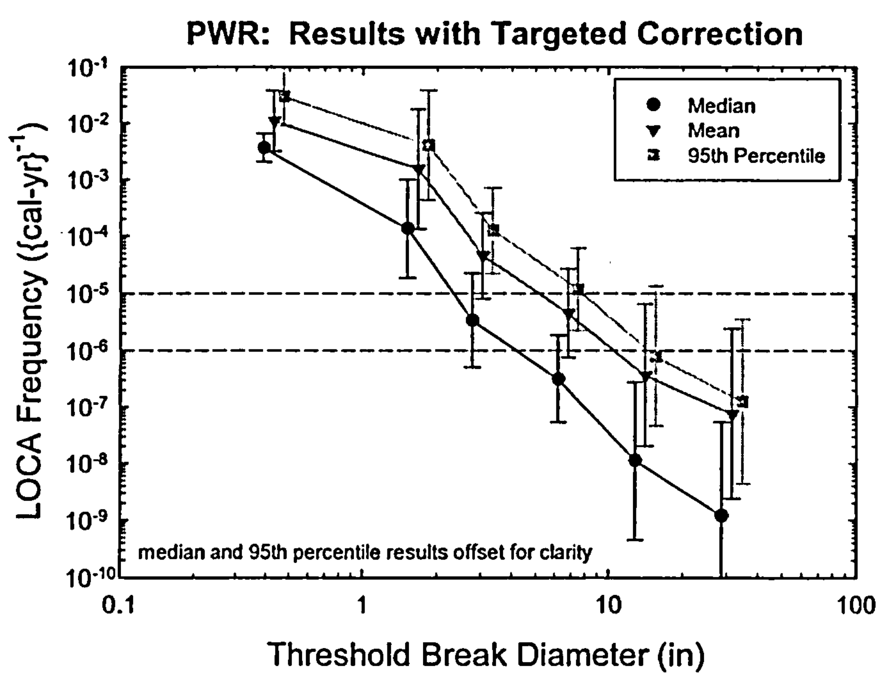

Figure 7.20 PWR LOCA Frequencies with T3 Adjustment

Table 7.6 Targeted (T3) Overconfidence Adjustment Compared to Unadjusted Results (Baseline)

BWR: Current Day			PWR: Current Day		
LOCA Category	Mean Ratio	95th Percentile Ratio	LOCA Category	Mean Ratio	95th Percentile Ratio
1	1.9	2.1	1	1.9	2.0
2	2.0	2.1	2	3.4	3.0
3	2.2	2.3	3	3.6	3.0
4	2.7	2.5	4	3.7	3.0
5	2.6	2.4	5	3.7	3.0
6	2.3	2.3	6	5.0	3.3

The sensitivity of the high uncertainty results to any overconfidence adjustment implies that these panelists did not underestimate the uncertainty associated with their elicitation responses as was initially expected. It may be that the elicitation training and the instruction provided to panelists during the elicitation appropriately sensitized the panelists to the likelihood and expectation that their uncertainty could be underestimated and they adjusted their responses accordingly. It may also be that the lognormal structure of the elicitation may have reduced the natural tendency toward overconfidence.

7.6.2.2 Error Factor Adjustment - The other type of overconfidence correction evaluated in this study (Section 5.6.2.2) adjusts the error factors associated with each panelist's total LOCA frequency estimate. First, the geometric mean of all the individual BWR and PWR unadjusted error factors was determined for each LOCA category and operating time period evaluated during the elicitation. Then, those panelists with error factors below the geometric mean for a given estimate have been adjusted up to the geometric mean. Those panelists with error factors above the geometric mean have not been adjusted. This approach leads to a variable adjustment as a function of plant type, LOCA category, and operating time. The geometric means (GMs) of the unadjusted error factors are summarized in Table 7.7. The upper error factor is the ratio between the 95th percentile and median estimates while the lower error factor is the ratio between the median and the 5th percentile estimates. The lower error factor GM tends to be larger than the upper error factor GM, except for PWR LOCA Categories 5 and 6. The implication is that the aggregated group opinion predicts a wider range of lower LOCA frequencies. The lower error factor GMs are similar for the BWR and PWR plant types. However, the PWR upper error factor GMs are somewhat larger than the BWR upper error factor GMs, reflecting more uncertainty about the effects of PWSCC (Section 6.3.2).

The current day BWR and PWR total LOCA frequency results arising from this error factor adjustment are illustrated in Figures 7.21 and 7.22, respectively, while the median, mean, 5th, and 95th percentile values are tabulated in Table 7.8 for both the current day and end of plant licensing period. The ratios of the adjusted error factor results to the baseline results are provided in Table 7.9. This table reports the ratio of the adjusted error factor mean and 95th percentile estimates along with the 97.5% upper confidence bound associated with these estimates. The adjusted error factor and baseline results are not markedly different. Additionally, the ratio of the adjustment to the baseline results for each of the parameters in Table 7.9 is similar for both BWR and PWR plant types. However, the difference between the adjusted and baseline results increases with increasing LOCA category because there are greater differences among the panelists' expected uncertainty. The BWR adjustment is less than a factor of 2 for both the mean and the 95th percentile for LOCA Categories 1 – 5, and the increase in these parameters is less than a factor of

3 for LOCA Category 6. Similarly, the PWR adjustment for the mean and 95th percentile are less than a factor of 2 for LOCA Categories 1 – 4, and less than a factor of 3 for LOCA Categories 5 and 6. The 95th percentile estimates are adjusted slightly more than the mean results. Similarly, the high confidence bound estimates for each parameter are adjusted slightly more than the group aggregated geometric mean values (Table 7.9). The PWR baseline and error factor adjusted means are compared in Figure 7.23. The modest deviation from the baseline estimates is apparent in this figure.

Table 7.7 Geometric Means of the Panelists' Error Factors (EF)

LOCA Category	BWR Plants		PWR Plants	
	Lower EF	Upper EF	Lower EF	Upper EF
1	7	5	5	4
2	13	7	14	10
3	13	8	13	12
4	18	9	16	12
5	24	13	21	24
6	30	26	26	32

Table 7.8 Total BWR and PWR LOCA Frequencies (After Overconfidence Adjustment using Error Factor Scheme)

Plant Type	LOCA Size (GPM)	Eff. Break Size (inch)	Current Day Estimate (per cal. year) (25 yr fleet average operation)				Estimate at End of Plant License (per cal. yr.) (40 yr fleet average operation)			
			5th Per.	Median	Mean	95th Per.	5th Per.	Median	Mean	95th Per.
BWR	>100	½	3.1E-05	3.0E-04	6.4E-04	2.1E-03	2.6E-05	2.6E-04	6.0E-04	2.0E-03
	>1,500	1 7/8	2.7E-06	4.8E-05	1.2E-04	4.1E-04	2.2E-06	4.4E-05	1.1E-04	4.1E-04
	>5,000	3 ¼	5.6E-07	9.7E-06	2.8E-05	1.0E-04	4.9E-07	9.8E-06	3.2E-05	1.2E-04
	>25K	7	9.6E-08	2.2E-06	7.3E-06	2.7E-05	8.7E-08	2.3E-06	9.3E-06	3.4E-05
	>100K	18	7.2E-09	2.9E-07	1.5E-06	5.4E-06	6.2E-09	3.1E-07	2.1E-06	7.3E-06
	>500K	41	5.6E-12	3.0E-10	6.4E-09	1.6E-08	6.7E-12	4.0E-10	1.0E-08	2.5E-08
PWR	>100	½	6.0E-04	3.7E-03	6.4E-03	1.8E-02	3.5E-04	2.5E-03	4.7E-03	1.4E-02
	>1,500	1 5/8	7.0E-06	1.4E-04	6.2E-04	2.2E-03	7.6E-06	1.6E-04	7.6E-04	2.7E-03
	>5,000	3	2.0E-07	3.4E-06	1.6E-05	5.8E-05	4.5E-07	7.6E-06	3.6E-05	1.3E-04
	>25K	7	1.3E-08	3.1E-07	1.6E-06	5.7E-06	2.6E-08	6.5E-07	3.6E-06	1.3E-05
	>100K	14	3.8E-10	1.1E-08	1.9E-07	5.2E-07	9.2E-10	2.7E-08	4.6E-07	1.3E-06
	>500K	31	3.3E-11	1.2E-09	3.1E-08	7.8E-08	8.2E-11	2.9E-09	8.1E-08	2.0E-07

Table 7.9 Ratio of Error Factor Adjusted to Baseline Results

	LOCA Category	Mean	Mean: 95% confidence	95^{th} Percentile	95^{th} percentile, 95% confidence
BWR	1	1.1	1.0	1.4	1.1
	2	1.1	1.0	1.4	1.1
	3	1.1	1.1	1.4	1.1
	4	1.2	1.1	1.4	1.2
	5	1.5	1.3	1.9	1.5
	6	2.0	2.3	2.3	3.0
PWR	1	1.1	1.1	1.3	1.2
	2	1.3	1.2	1.6	1.4
	3	1.2	1.3	1.5	1.5
	4	1.4	1.5	1.7	2.0
	5	1.8	2.1	2.1	2.7
	6	2.1	2.6	2.1	2.8

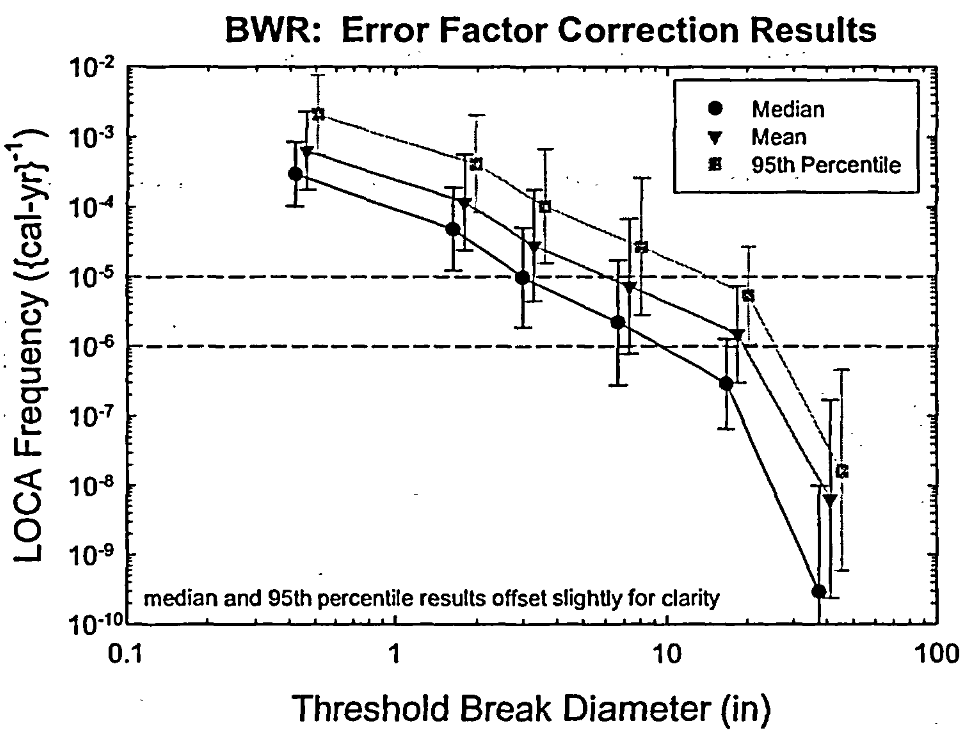

Figure 7.21 BWR LOCA Frequencies with Error Factor Adjustment

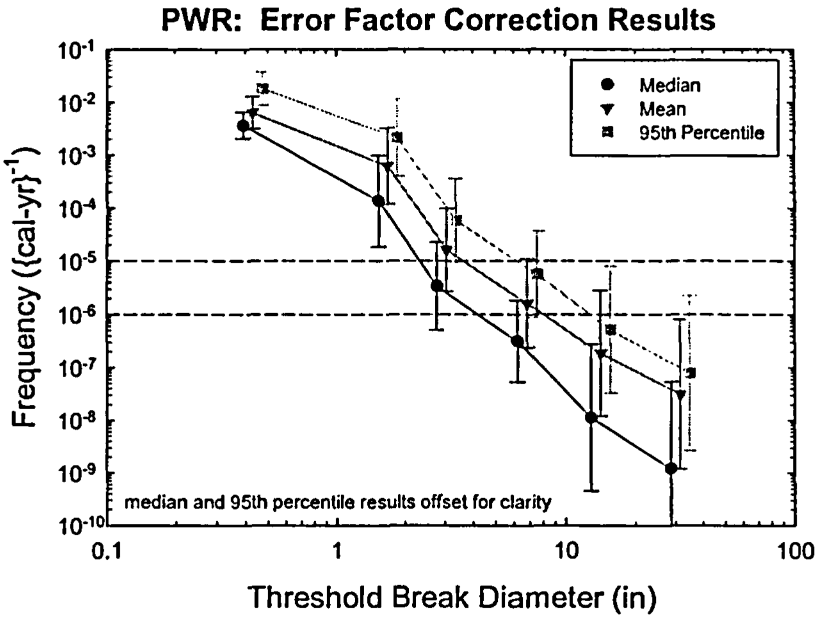

Figure 7.22 PWR LOCA Frequencies with Error Factor Adjustment

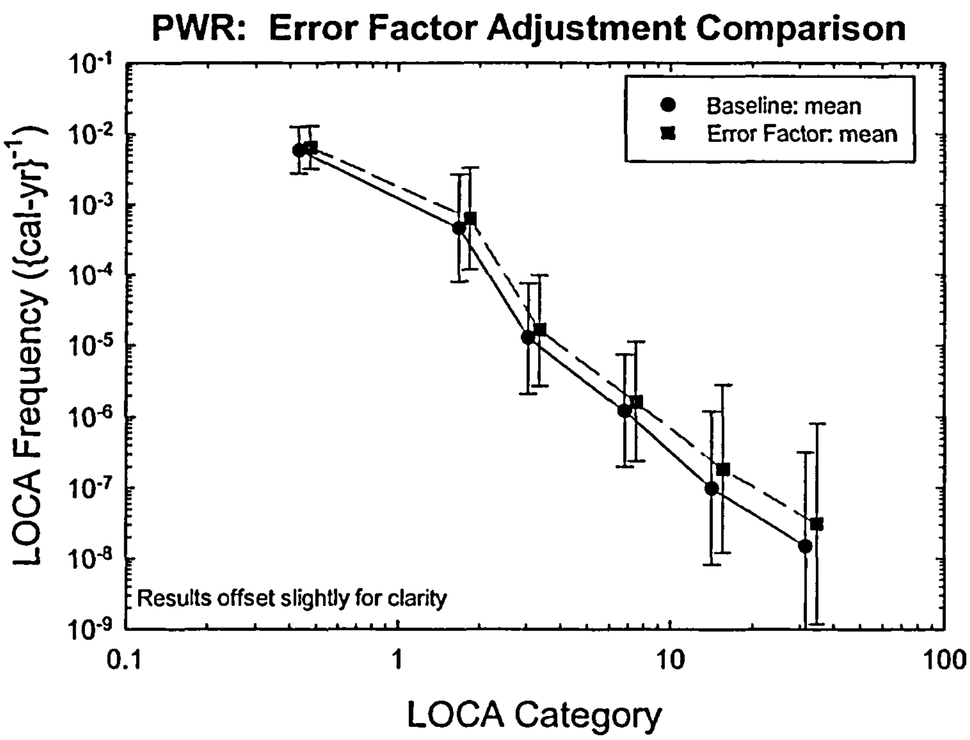

Figure 7.23 Comparison of Baseline Results and Error Factor Adjustment

The error factor adjustment also compares well with the targeted (T3) adjustment discussed earlier. The mean and 95th percentile estimates from these two adjustments differ by less than a factor of 2.8 for all LOCA categories and plant types. Differences are not obviously correlated with LOCA category. However, the adjusted error factor results are always less than the T3 adjusted results. The implication of this is that the error factor adjustment modifies the panelists' coverage intervals from 90% to less than 60% (the value for the T3 adjustment) for those affected panelists. An effective coverage interval adjustment could be determined for the affected panelists' bottom line estimates, but it is more difficult to determine the corresponding adjustment for each individual elicitation response. The elicitation responses were directly modified in the T3 adjustment (Section 7.6.2.1).

The error factor adjustment is preferable to the coverage interval adjustment because the error factor adjustment varies as a function of the difference between the individual and group (geometric mean) estimates for each parameter. Additionally, the error factor adjustment requires no ad hoc determination of which panelists to adjust and the level of the adjustment as was the case for the coverage interval adjustment. Also, because group uncertainty increases with LOCA size, the error factor adjustment is tailored to increase with LOCA size. This is warranted because larger LOCA sizes deviate further from the operating experience failure and precursor data. Overconfidence for estimating frequencies associated with smaller LOCAs is therefore less likely.

7.6.3 Correlation Structure

The percentiles of the total BWR and PWR LOCA frequency distributions are a function of the correlation structure of the distributions associated with the various LOCA-sensitive piping systems and non-piping subcomponents and which are summed to obtain the total frequency distributions (Sections 5.3.4 and 5.6.3). While it is plausible to assume that the components of the sums are positively correlated, the correlation structure is otherwise unknown. However, the correlation structure can be bounded by assuming that the piping systems and non-piping subcomponents are either independent (zero correlation) or have perfect rank correlation (maximum achievable correlation). The baseline analysis (Section 5.5) assumes perfect rank correlation to calculate percentiles during any step which requires the summation of distributions. This sensitivity analysis examines the effect of assuming independence when summing distributions.

Monte Carlo simulation was used for evaluating differences between assuming either a perfect rank correlation or independent correlation structure. Because of the very large number of individual distributions that comprise the elicitation responses, Monte Carlo trials were conducted to calculate the bottom line estimates only for selected panelist responses. Ten simulation trials were selected to span several important variables as summarized in Table 7.10. First, both the panelists and the range of LOCA plant type combinations (i.e., PWR piping, PWR non-piping, BWR piping, and BWR non-piping) and time periods were sampled. Second, a number of distributions representing contributing piping components (or non-piping subcomponents) which must be summed to develop the bottom-line estimates were used. Third and most important, the Monte Carlo trials were selected to span the range of distinguishing characteristics representative of the elicitation responses. These distinguishing characteristics (Table 7.10) indicate whether the elicitation responses are generally symmetric (S in Table 7.10) or asymmetric (U in Table 7.10), and the relative magnitude (small, moderate, large) of the upper or lower error factors (EF in Table 7.10). The characteristics are representative of the variability among the entire population of elicitation responses.

Table 7.10 Summary of Monte Carlo Trials

Trial Number	Number of Distributions	Panelist	LOCA Plant Type/Time Period	Distinguishing Characteristics
1	12	A	BWR-1 Piping @ 25 yrs	S, small EF
2	12	A	BWR-2 Piping @ 25 yrs	U, small upper EF
3	2	C	PWR-6 Piping @ 25 yrs	U, large upper EF
4	4	C	BWR-3 Piping @ 25 yrs	S, moderate EF
5	14	G	PWR-5 Non-Piping @ 60 yrs	U, large lower EF
6	5	C	BWR-3 Non-Piping @ 25 yrs	U, large lower EF
7	8	J	PWR-5 Non-Piping @ 25 yrs	S, large EF
8	7	I	PWR-4 Piping @ 25 yrs	S, moderate EF
9	4	E	BWR-4 Non-Piping @ 25 yrs	U, large lower EF
10	9	B	PWR-3 Non-Piping @ 25 yrs	S, small EF

Each of the individual piping component (or non-piping subcomponent) distributions is the product of two independent distributions for the base case and the relative ratio elicitation responses. For the perfect rank correlation structure (PRCS), all elicitation responses were simulated in the Monte Carlo trials by split lognormal distributions truncated at the 99.9th percentile. These individual piping component distributions were determined by multiplying independent base case and relative ratio distributions (Section 5.3.3). In each Monte Carlo trial, the distribution of each such product was determined by generating 10,000 independent pairs of the base case and relative ratio distributions, multiplying the paired values and sorting them into ascending order. To obtain the distribution of the sum, S, of any number of such products under the PRCS assumption, the rank 1 values of the products were summed to obtain the rank 1 value of S, the rank 2 values of the products were summed to obtain the rank 2 value of S, and so on. The 10,000 ordered values of S is the bottom-line distribution under the assumption of a perfect rank correlation structure. The required percentiles (median, 5th and 95th are then read off the distribution of S.

The results obtained by the Monte Carlo procedure are somewhat different than the results obtained by the baseline calculation methodology described in Section 5.3.5 because this technique uses an approximation to the assumed truncated split lognormal distributions to calculate the percentiles. Figure 7.24 compares the 5th, 50th, and 95th percentiles associated with the final bottom line distributions calculated using the approximate calculation methodology and the Monte Carlo simulations under the assumption of a perfect rank correlation Structure. This figure illustrates the ratio of the Monte Carlo to the calculated elicitation result. The 5th and 95th percentile elicitation calculations differ by less than 10% from the Monte Carlo simulations for all the trials. The median calculations also differ by less than 10% except for trials 2 and 9 where the elicitation results are approximately 30% and 20% higher, respectively, than the Monte Carlo estimates. It is somewhat surprising that the largest differences are apparent for the median estimates. However, it is interesting to note that the distributions in trials 2 and 9 are asymmetric and the lower tail error factor is much bigger than the upper error factor. As expected, the mean ratio of the two results (not illustrated) is always approximately 1 since both the Monte Carlo and elicitation responses assume identical distributions for calculating the mean. Based on these results, it is concluded that the differences between the approximate calculation methodology and the Monte Carlo analysis are generally insignificant. The largest differences are expected for the median estimates for distributions with large symmetric lower tails. In these cases, the approximate median estimates will be marginally conservative. However, differences in the individual bottom line estimates will be reduced after aggregating all the responses since the individual estimates have quite different characteristics and do not all have large asymmetric lower tails.

For an assumed independent correlation structure, no closed-form methodology for approximating the percentiles was developed as it was for an assumed PRCS. Therefore, the effect of the correlation structure on the bottom-line aggregated estimates cannot be directly determined by comparing the results of the two correlation structures. However, a comparison of the bounds calculated for the selected simulations (Table 7.10) can be made. The independent correlation bounds were determined as follows. The calculation starts with the individual piping component (or non-piping subcomponent) distributions that have been determined by independently combining the base case and relative ratio distributions using Monte Carlo simulation as previously described. These are the identical distributions used for the perfect rank correlation simulation. Only the summation step differs. The product distributions in each sum are independently sampled and then added to determine the total piping (or non piping) distribution. This calculation is then repeated to obtain 10,000 samples of the total distribution. The summed frequencies are then ordered and assigned rank probabilities to determine the associated cumulative distribution function for comparison with the rank correlation estimate.

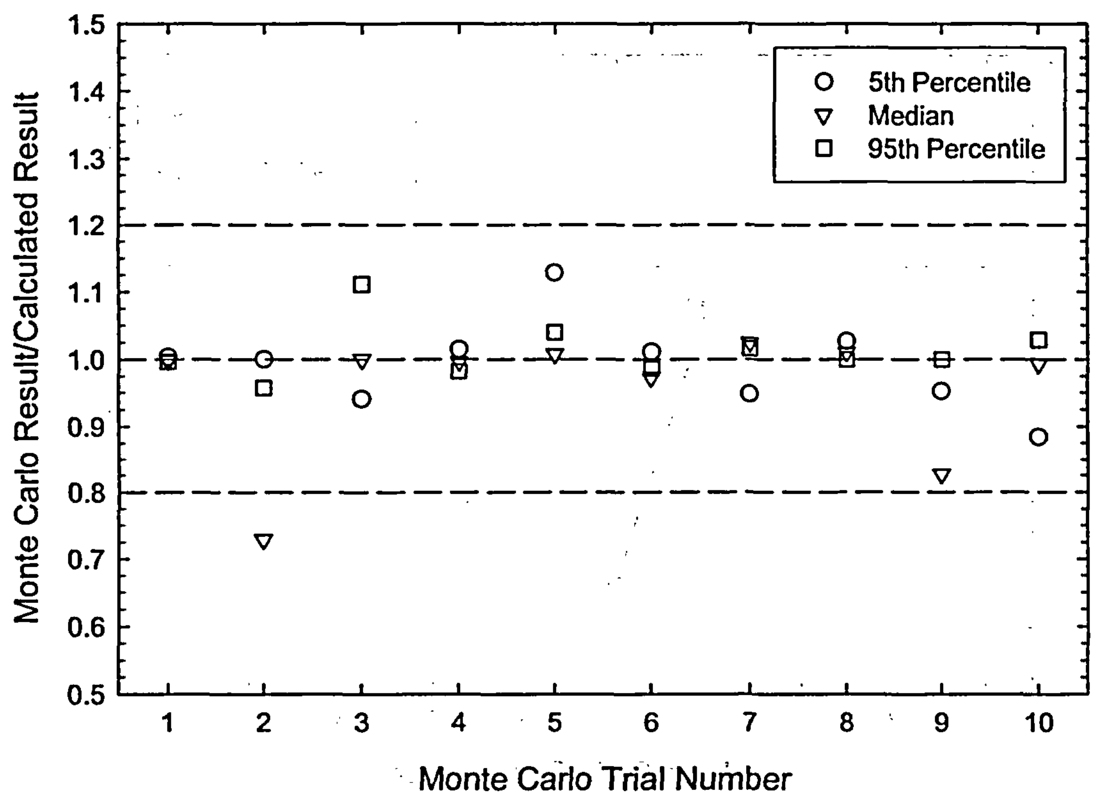

Figure 7.24 Comparison of Monte Carlo Results to Elicitation Calculations Assuming a Rank Correlation Structure

Figure 7.25 illustrates the ratio of the median, 5[th] and 95[th] percentile estimates determined by assuming a perfect rank correlation structure to those calculated assuming an independent correlation structure as a function of the trial number (Table 7.10). The mean is not illustrated because it is unaffected by the correlation structure. For these 10 trials, the median and 5th percentile estimates based on the independent correlation structure are consistently larger than those based on the PRC structure. The PRC estimates of the 95th percentile are usually larger than the independent estimates with the exception of trial numbers 3 and 7. These distributions are characterized by large error factors (up to 1000). Therefore, in general, the PRC structure usually results in larger differences between the 5th and 95th percentiles. This is expected since the maximum correlation is obtained using the perfect rank correlation structure.

It is also apparent (Figure 7.25) that differences resulting from the correlation structure decrease as the percentiles of the combined distributions increase. Differences in the 5th percentile can be as much as nearly two orders of magnitude. The largest differences occur in trials 2, 5, 7, and 9 which consist of distributions with the largest lower tail error factors. Differences in the median estimates are less then a factor of two for all the Monte Carlo trials except 3, 5, and 7 which are comprised of distributions having the widest error factors. The largest difference in the median estimates is more than an order of magnitude and occurs for trial 7. This trial requires combining several high error factor distributions. Differences between the 95th percentile estimates calculated by the PRC and independent correlation structures are generally less than a factor of two. The largest differences occur for trials 3 and 7 and stem from the large error factors associated with the trial distributions.

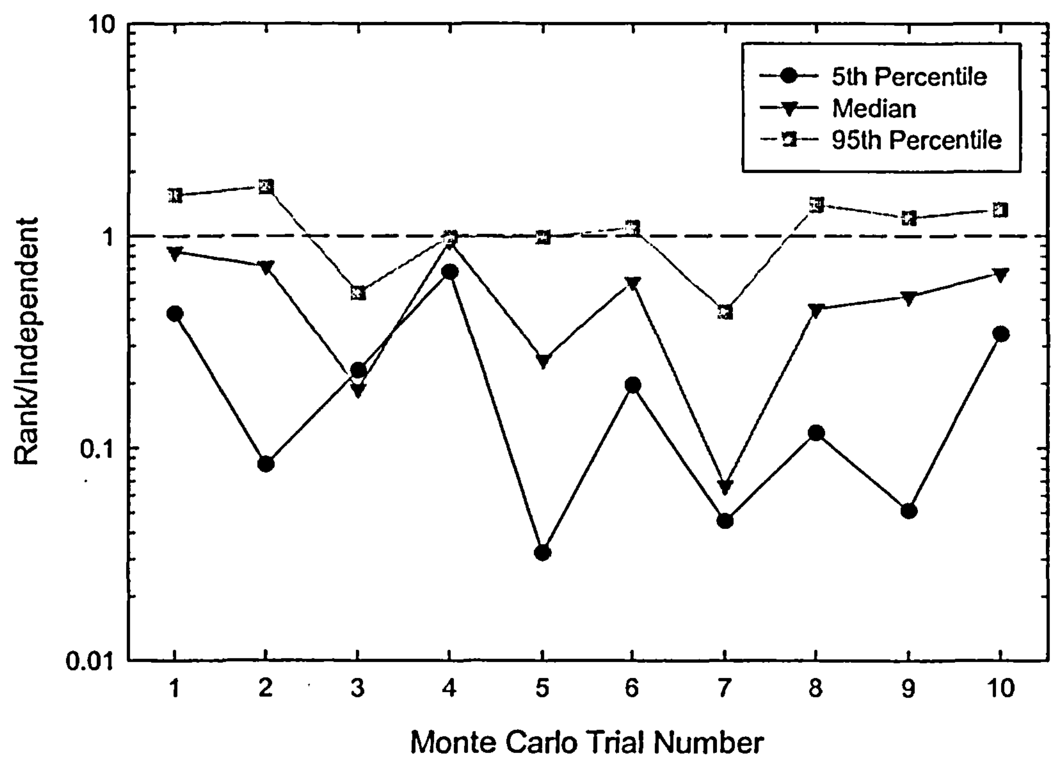

Figure 7.25 Ratio of Rank to Independent Correlation Structures for Combining Selected Elicitation Responses

The baseline calculation method assumes a perfect rank correlation structure. This assumption is justified because the elicitation responses tend to be more positively correlated than uncorrelated. Most panelists noted that degradation mechanisms that significantly contribute to the LOCA frequencies affect many systems similarly. That is, if conditions lead to higher failure probabilities in one piping system due to a specific degradation mechanism, the failure probabilities of other susceptible systems will also increase. Additionally, the rank correlation structure is justified because the associated 95th percentile estimates are usually upper bounds, except for extreme distributions with large upper tail error factors (1000). In all cases, there are relatively insignificant differences in the 95th percentile estimates determined using either correlation structure. Therefore, the upper percentiles of these distributions are not particularly sensitive to the correlation structure.

The correlation structure more significantly affects the lower percentiles. The perfect rank correlation structure leads to lower bound estimates of the median and 5th percentile. Differences related to the correlation structure increase as the error factor and number of contributing distributions increase. This point should be considered when interpreting results utilizing parameters other than the 95th percentile or mean. However, it should be stressed that the Monte Carlo trials were conducted using distributions that exhibit bounding characteristics for the entire population. The largest differences exhibited in Figure 7.25 are not representative of the majority of the elicitation distributions. Therefore, when the individual panelist results are aggregated using the baseline methodology (i.e., calculating the geometric mean of the individual panelist estimates), any large differences between the bounding lower percentile estimates will be reduced.

7.6.4 Aggregation

There were several analyses performed to examine the sensitivity of the baseline results to other possible aggregation schemes for combining the panelists' responses. Studies evaluated the effect of using different measures of group opinion other than the geometric mean (Section 5.6.4.1), the effect of aggregating panelist responses at different stages of the analysis (Section 5.6.4.2), and the sensitivity of the results to the method of calculating parameters using the lognormal parameterization (Section 5.6.4.3). Aggregation by averaging the individual panelists' distributions to create a mixture distribution was also evaluated (Section 5.6.4.4) More detail on each of these analyses is available in Section 5.6.4.

7.6.4.1 Measures of Group Opinion - The technique used to determine group opinion can have a significant effect on the aggregated LOCA frequency estimates. The objective of the elicitation (Section 2) is to determine a central estimate of group opinion. Several different central estimate measures were used to determine group opinion from the individual panelist's responses: geometric mean (GM), trimmed geometric mean (TGM), median, and arithmetic mean (AM). The baseline LOCA frequency results use the geometric mean (Section 5.5).

There is little significant difference between the median, GM, and TGM estimates of central group opinion (Figure 7.26). In this figure, the median, GM, and TGM measures of the mean PWR LOCA frequency estimates at 25 years of plant operation are illustrated. The differences among these three measures are typically less than a factor of 3 for all the bottom line parameters (i.e., mean, median, 5th, and 95th percentiles). There was some initial concern that the very low frequency estimates provided by Panelist J for the PWR Category 5 and 6 LOCA frequencies may significantly affect the GM estimates. However, as seen in Figure 7.26, this is not the case. Differences among these measures are slightly less than a factor of 2 for the BWR LOCA frequency estimates, except for LOCA Category 6 where the median and geometric mean estimates of the group mean and 95th percentiles vary by a factor of 6. The closer agreement for LOCA Categories 1 - 5 stems from the reduced variability among the individual panelists for the BWR LOCA frequency estimates. Hence, all these estimates provide similar measures of central group opinion. There was much more variability among the panelists for the PWR Category 6 estimates and this led to increased differences among these various central value estimates.

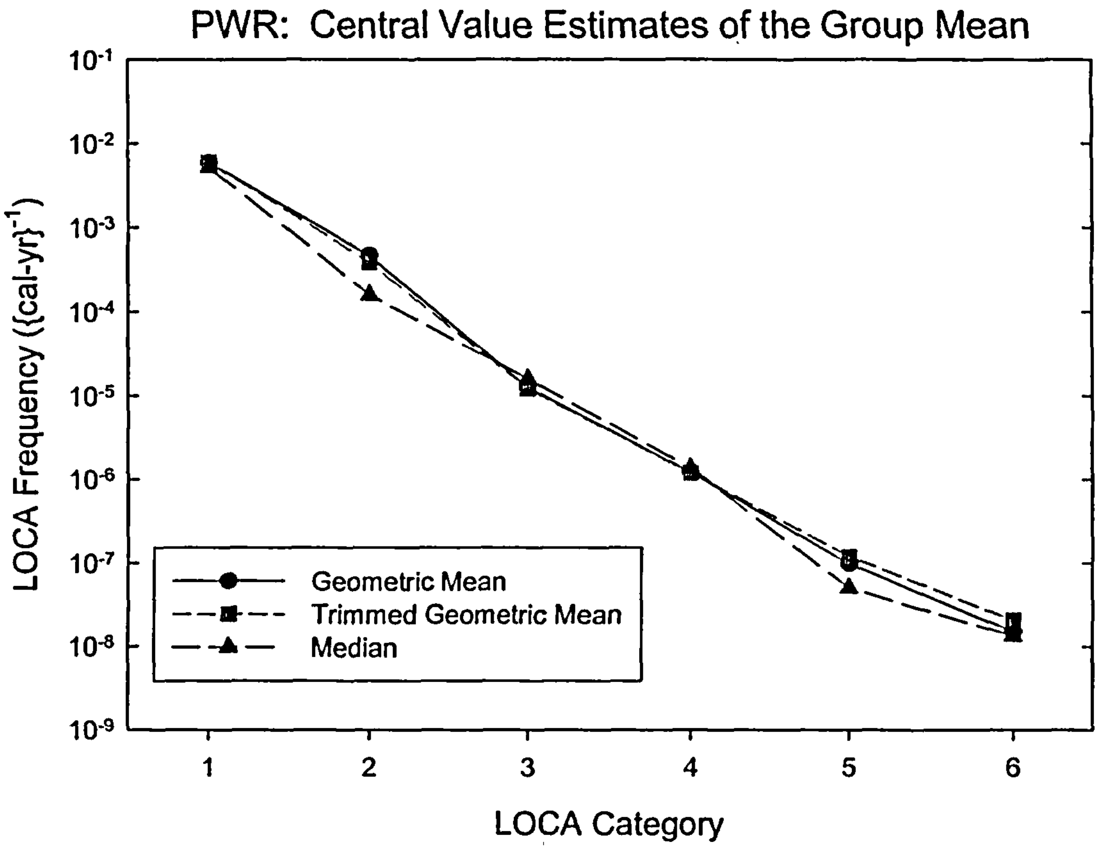

Figure 7.26 Various Central Value Estimates of the PWR LOCA Frequencies at 25 Years

The arithmetic mean and geometric mean estimates of the BWR and PWR LOCA mean frequencies are compared in Figures 7.27 and 7.28, respectively. These figures indicate that the arithmetic mean LOCA frequency estimates are significantly higher than the other estimates. Because the arithmetic mean is always larger than the geometric mean, the arithmetic mean estimates always result in increased LOCA frequency estimates. The amount of the increase gets larger with the difference between the upper bound estimates and the estimates from the remaining panelists. The ratios of the arithmetic mean to geometric mean LOCA frequencies are provided in Table 7.11. The largest differences are greater than a factor of 10 and occur for the BWR Category 3 and 4 and the PWR Category 5 and 6 LOCA estimates. In each of these instances, there are one or two panelists' estimates that are significantly higher than the others. The central estimate of group opinion is then dominated by these high values.

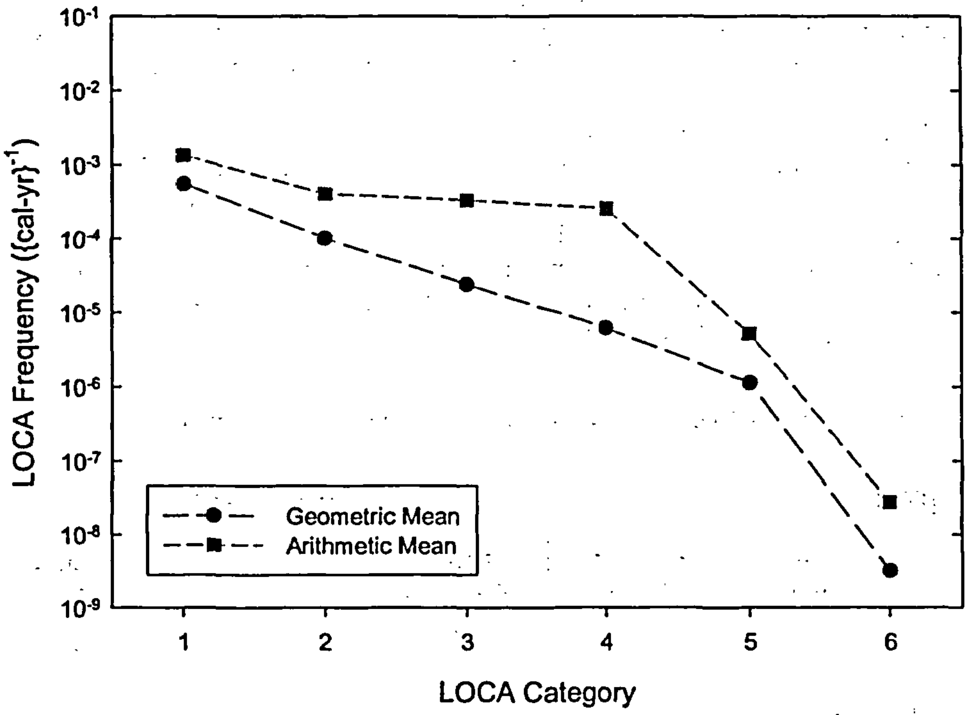

Figure 7.27 Comparison of Arithmetic Mean and Geometric Mean Estimates of the BWR LOCA Frequencies at 25 Years

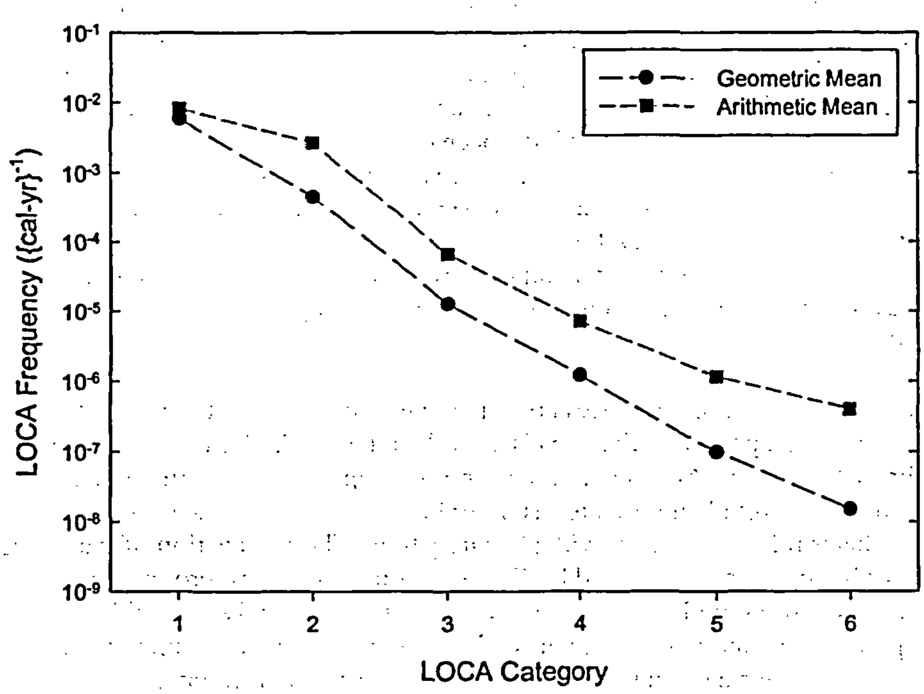

Figure 7.28 Comparison of Arithmetic Mean and Geometric Mean Estimates of the PWR LOCA Frequencies at 25 Years

A single panelist, C, expects the Category 2 – 4 LOCA frequencies for BWR plants to be essentially constant due to IGSCC concerns in large recirculation system lines and the expectation that failure of these lines will propagate directly to a larger LOCA (Section 7.5.2). The expected frequency of these failures is approximately two orders of magnitude higher than other estimates for Category 4 LOCAs. The arithmetic mean estimations largely reflect this single panelist's expectation for the Category 3 – 4 LOCAs. The same panelist also has large uncertainty about the Category 5 and 6 PWR LOCA estimates. While this panelist's median values are consistent with the remainder of the group, the large uncertainty leads to mean values that are significantly higher than most other panelists. Also, Panelist A's median Category 5 and 6 LOCA estimates are significantly higher than the remaining panelists (Section 7.5.2). These high median failure expectations lead to comparatively high mean frequency estimates. These two panelists' estimates cause the increases in the PWR Category 5 and 6 arithmetic mean estimations when compared to the geometric mean estimates.

Table 7.11 Ratio of Arithmetic Mean to Geometric Mean (Baseline)

BWR: Current Day			PWR: Current Day		
LOCA Category	Mean Ratio	95^{th} Percentile Ratio	LOCA Category	Mean Ratio	95^{th} Percentile Ratio
1	2	3	1	1	2
2	4	4	2	6	6
3	14	17	3	5	5
4	42	50	4	6	5
5	5	5	5	12	7
6	9	6	6	27	20

7.6.4.2 Aggregation Point - While the measure used to determine group opinion can be important, the stage at which the results are combined is also an important consideration (Section 5.6.4.2). There are three natural aggregation points in this elicitation structure. One is after the total LOCA frequencies have been calculated for each panelist. Another is after each panelist's piping and non-piping LOCA frequencies have been determined. The last is at the most fundamental level of the elicitation responses, before any analysis is conducted. However, this last aggregation point is problematic because the panelists' elicitation responses did not have the same basic structure and the same level of completeness. Furthermore, additional assumptions would be required to impute missing or incomplete responses in order to aggregate at this most fundamental level. Therefore, aggregation at this level was not performed for the current elicitation analysis.

However, the current elicitation responses are suitable for aggregation at the other two points. The baseline aggregation point (Section 5.5) is after the total LOCA frequencies have been determined for each panelist. At this point, the central group opinion is estimated using the geometric mean of the individual estimates as described previously for the baseline analysis (Section 5.5). This is referred to as individual aggregation. The aggregation of the piping and non-piping contributions is accomplished by first estimating the central group opinion associated with the piping and non-piping LOCA frequency parameters separately. As with the baseline aggregation approach, the individual estimates of these parameters are assumed to be lognormally distributed and the geometric mean is used to determine the central estimate. The piping and non-piping central estimates are then summed using the procedures described in Section 5.3.4. This aggregation scheme is referred to as group aggregation.

The LOCA frequency differences resulting from the chosen aggregation point are summarized in Figure 7.29 for the PWR current day results. This figure illustrates median, mean and 95th percentile LOCA frequency estimates for both individual and group aggregation as described above. The differences are small, usually less than a factor of two for all the bottom-line parameters, although the differences do increase as the LOCA size increases. The differences in the BWR results are similar, but they are not a function of the LOCA size. For both plant types, the baseline (individual) aggregation results lead to slightly higher frequencies. Therefore, while the results are not particularly sensitive to the elicitation point, the baseline methodology results in slightly more conservative frequencies.

Individual aggregation is also the best choice based on the elicitation structure. This aggregation point is not affected by panelist differences in the piping and non-piping boundary definitions which could affect whether contributions (e.g. CRD failures) were counted in piping or non-piping categories by an individual panelist. Individual aggregation also ensures that the relative piping and non-piping contributions and their associated uncertainties reflect the correlations expressed by the individual panelists. For example, a panelist might expect that piping is the dominant contributor to the total LOCA frequency estimates, but could be much less confident (large uncertainty) about the non-piping frequency estimates. For this panelist, the total LOCA frequency estimates would be dominated by the piping contribution and its associated uncertainty. Thus, individual aggregation aggregates only the important contributors for each panelist. In contrast, with the group aggregation approach, the more uncertain, but smaller non-piping contribution for this panelist is explicitly considered in the non-piping aggregation. Because these contributions have a direct effect on the combined results, their actual values become important with group aggregation, while individual aggregation more importantly captures the opinion of this panelist that the non-piping contribution should be negligible. In short, individual aggregation after the total LOCA frequencies have been calculated (baseline methodology) ensures that the results are most consistent with the panelists' opinions and that only the most important contributors identified by each panelist factor into the group total LOCA frequency estimates.

7.6.4.3. Aggregation Parameters - This sensitivity analysis compares the bottom-line mean estimates determined from direct aggregation with estimates calculated from the other aggregated percentiles. In the baseline methodology, the mean estimates are calculated directly from the elicitation responses for each individual panelist by summing the means of the piping component and non-piping subcomponent contributions (Section 5.5). The baseline methodology also aggregates the other percentiles (i.e., median, 5th, and 95th) directly from the individual panelist responses (Section 5.5). An alternative approach is to calculate the bottom-line mean estimates from the aggregated percentile estimates by assuming a distributional relationship for the percentile estimates.

This alternative scheme is called the median and error factor (MEF) approach. In this approach, the 5th and 95th percentile estimates are calculated from the aggregated median and error factor values instead of being directly aggregated as in the baseline methodology. The bottom-line mean estimates are then calculated from the median, 5th and 95th percentile values by assuming that the underlying distribution is a split lognormal which is truncated at the 99.9th percentile. This is the same distributional form assumed for calculating the mean estimates from the individual elicitation responses in the baseline methodology (Section 5.5).

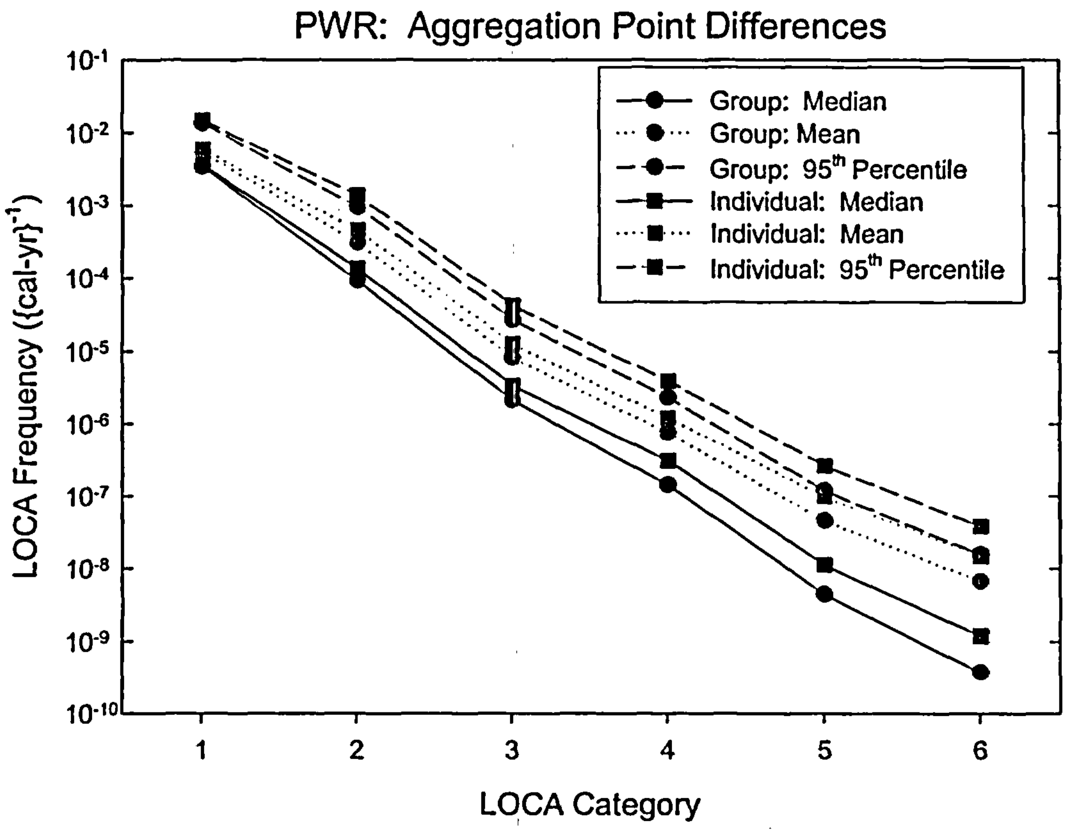

Figure 7.29 Effect of Aggregation Point on LOCA Frequency Estimates: PWR LOCA Frequencies at 25 Years of Operation

The 5th and 95th percentile estimates determined by the MEF approach are identical to the estimates determined by direct aggregation since all the individual panelist estimates can be equivalently represented by the 5th, 50th, or 95th percentiles (baseline methodology) or by the median and upper and lower error factors (MEF approach). Therefore, the MEF approach was used to calculate the percentiles simply to validate the baseline calculations. However, the mean can be determined at several points in the analysis and differences do result from this choice. Another possible calculation point is after the bottom-line percentile estimate for each individual panelist is determined. It is postulated, however, that the MEF approach will maximize differences in the mean estimates because the characteristics of the aggregated percentiles are likely to differ most from the individual elicitation responses due to all the intermediate processing steps (Section 5).

Representative results from this sensitivity analysis are illustrated in Figure 7.30 for the current day PWR results. In this figure, the MEF approach (MEF) and baseline results for the mean and 95th percentile are separately illustrated. As expected, the 95th percentile estimates (as well as the 5th and 50th percentiles) from both approaches are identical. In general, the mean estimates determined using both approaches are also consistent. Differences do increase with increasing LOCA size, but the mean estimates vary by less than 35% for the PWR results (Figure 7.30), and less than 40% for the BWR results (not shown). Therefore, the approach used to determine the aggregated means has little influence on the mean estimates. However, the baseline approach of directly aggregating the mean estimates from the elicitation responses always results in slightly higher frequencies.

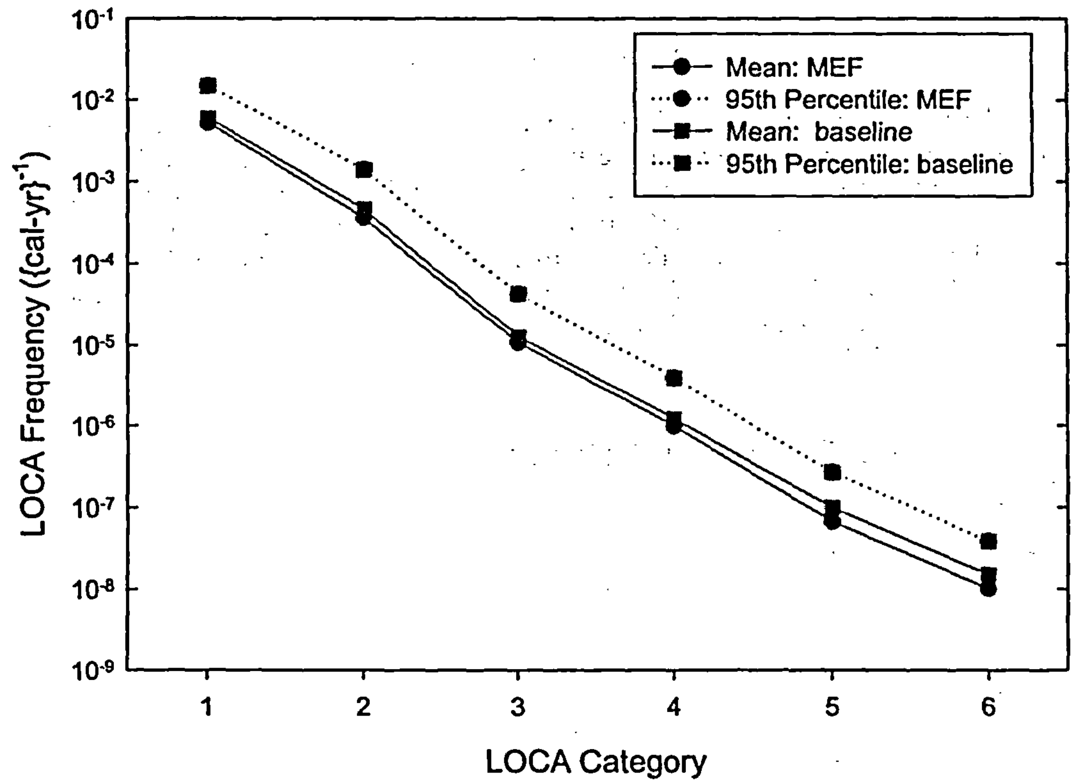

Figure 7.30 Comparison of Lognormal Aggregation Parameters: PWR LOCA Frequencies at 25 Years

7.6.4.4 Mixture Distribution Aggregation - An alternative method of aggregating the individual panelist bottom-line estimates is to use a mixture distribution (Section 5.6.4.4). The mixture distribution used is the average of all the individual panelist distributions for each LOCA category, plant-type (i.e., BWR or PWR), and operating time period. This aggregation methodology assumes that the elicitation responses come from equally credible models and that the responses provided by the experts are representative of the entire population of experts (Section 5.6.4.4). The mixture distribution approach does not attempt to develop aggregated estimates that represent the central group opinion as does the baseline methodology, but rather attempts to exhibit the full range of variability among the panelist responses. More information on the philosophy of the mixture distribution approach is provided in Section 5.6.4.4.

Mixture distributions were developed for all BWR and PWR LOCA categories for the current day time period using the following procedure. Starting from the three estimated bottom-line percentile estimates (median, 5th, and 95th percentiles) for each panelist; it was assumed that these estimates come from a split lognormal distribution. Percentiles below the median are determined using the lower tail error factor. Percentiles above the median value are determined using the upper tail error factor. The cumulative distribution function (CDF) of the mixture distribution for each plant type and LOCA category is the average of the corresponding split lognormal CDFs of all panelists who provided bottom-line estimates. A frequency range for each mixture distribution was determined by setting its lower end equal to the 0.1th percentile of the individual panelist CDF with the smallest 5th percentile and setting its upper end equal to the 99.9th percentile of the individual panelist CDF with the largest 95th percentile. This frequency range was divided into 1000 equal intervals and the individual panelist CDFs were calculated and averaged at each of the 1000 points in the frequency range. This procedure yielded a CDF evaluated at 1000 points for each plant type and LOCA category.

The 5th, 50th, and 95th percentiles were then determined for each mixture distribution. Because the mean of a mixture distribution is equal to the average of the means of the individual panelist distributions (Section 5.6.4.4), the mean of each mixture distribution is equal to the arithmetic mean of the individual mean estimates (Section 7.6.4.1). A summary of the mean, median and 95th percentiles for the mixture distributions is provided in Figure 7.31 for the BWR and Figure 7.32 for the PWR LOCA frequencies for the current day time period. Although confidence bounds have not been determined for these distributions, they could be calculated using a bootstrap technique [7.2].

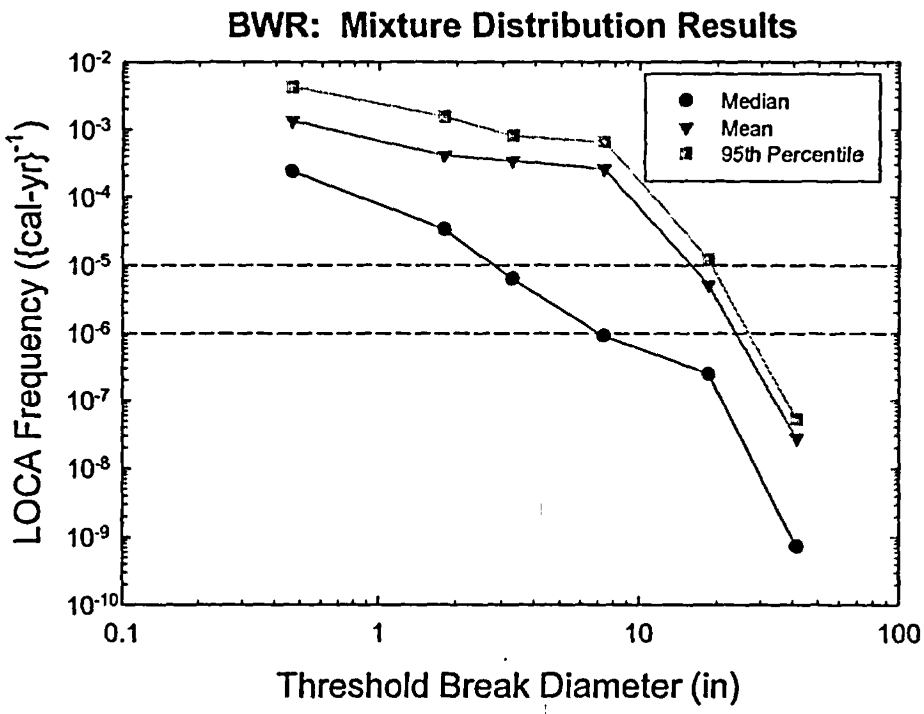

Figure 7.31 BWR LOCA Frequencies Determined by Mixture Distribution Aggregation: Current Day Estimates

Table 7.12 compares the mixture distribution and geometric mean (baseline methodology) aggregation techniques by providing the ratios of the median, mean, and 95th percentile estimates. The median estimates determined from either aggregation techniques are relatively consistent, usually within a factor of 2. As mentioned, the ratios of the mean values are identical to those contained in Table 7.11. The ratios of the 95th percentile estimates in Table 7.12 are also similar to the 95th percentile ratios in Table 7.11. In fact, the 95th percentiles for the mixture distribution are usually within a factor of 2 of the arithmetic mean aggregated estimates.

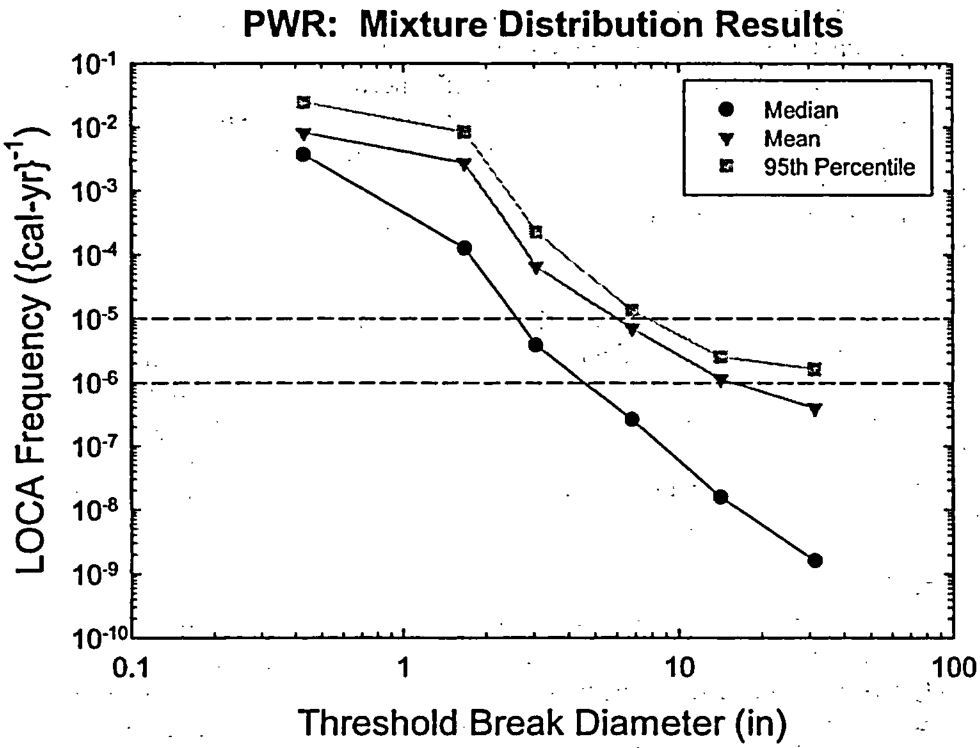

Figure 7.32 PWR LOCA Frequencies Determined by Mixture Distribution Aggregation: Current Day Estimates

Table 7.12 Ratio of Mixture Distribution to Geometric Mean Aggregated Results

		BWR: Current Day			PWR: Current Day	
LOCA Category	**Median Ratio**	**Mean Ratio**	**95th Percentile Ratio**	**Median Ratio**	**Mean Ratio**	**95th Percentile Ratio**
1	0.8	2	3	1.0	1	2
2	0.7	4	5	0.9	6	6
3	0.7	14	10	1.2	5	5
4	0.4	42	32	0.8	6	3
5	0.8	5	3	1.4	12	10
6	2.5	9	7	1.4	27	43

The reasons for the disparity between the mean and 95th percentile estimates in Table 7.12 are identical to those underlying previous discrepancies between the arithmetic and geometric mean aggregation (Section 7.6.4.3). When differences between the mixture distribution and geometric mean aggregated parameters are large, it is an indication that one or two panelists' estimates are significantly greater than the remaining estimates for that category. This is particularly true for BWR LOCA Categories 3 and 4 and PWR LOCA Categories 5 and 6 (Section 7.6.4.3). Smaller differences are indicative of more uniformity among the panelist estimates. This characteristic also explains why the arithmetic mean aggregated 95th percentile estimates are similar to the mixture distribution 95th percentiles. The largest panelist estimate often dominates both aggregation schemes.

The similarity of the parameter estimates aggregated by the arithmetic mean scheme to the mixture distribution parameter estimates decreases as the percentiles decrease. This occurs because the arithmetic mean aggregation is most greatly influenced by the highest individual panelist estimates for each parameter while the mixture distributions become more influenced by the lowest panelist responses as the percentiles decrease. In the extreme, the 5th percentile of the mixture distribution (not shown) can be significantly lower than either the geometric or arithmetic mean aggregated estimates if one or two panelist estimates are significantly lower than the remaining panelist estimates.

The mixture distribution aggregation scheme will always result in higher mean and 95th percentile estimates and lower 5th percentile estimates than the other aggregation schemes. Consequently, the mixture distribution scheme exhibits the greatest difference between the 5th and 95th percentiles. However, the mean and 95th percentile estimates are often dominated by the maximum panelist estimate while the 5th percentile is often dominated by the minimum panelist estimate. This characteristic implies that the extreme individual estimates will often dominate the mean, 5th, and 95th percentile estimates.

7.6.5 Panel Diversity

While the previous section evaluated the effect of different choices to assess central group opinion, another principal objective of this elicitation is to also illustrate panel diversity (Section 3.2). Panel diversity has been quantified using two methods. The baseline method (Section 5.5) calculates two-sided 95% statistical confidence intervals assuming that the individual panelist LOCA frequency estimates of the mean and 95th percentile are lognormally distributed. Under this assumption, the group estimate is the geometric mean and a measure of panel diversity is a confidence interval for the group estimate (Section 5.4).

A sensitivity analysis examines another method (quartile method) to measure panelist diversity. The quartile method does not rely on the assumption of an underlying distributional structure for the individual estimates. For each set of individual panelist estimates, a quartile interval is constructed from the upper and lower quartiles of the individual estimates. The lower and upper quartiles represent the 25th and 75th percentiles, respectively, of the individual panelist estimates. In this method, the median (or 50th percentile estimate) is a natural measure of central group opinion. It has been previously shown that the geometric mean and median (Section 7.6.4.1) provide similar measures of central group opinion. Thus, a quartile interval is a measure of group diversity about the median analogous to a confidence interval about the geometric mean.

The results for the current-day BWR and PWR LOCA frequency estimates are summarized in Figures 7.33 and 7.34, respectively. The BWR baseline confidence intervals and quartile intervals are quite similar for LOCA Categories 1, 2, and 5 for both the mean and 95th percentile estimates. However, the upper baseline confidence bounds are approximately a factor of 5 higher for both the mean and 95th percentiles for BWR LOCA Categories 3 and 4. Once again, this difference reflects the single high panelist estimate for these LOCA categories. The quartiles are not affected by this extreme value while the statistical confidence bounds are. The upper confidence bounds for LOCA Category 6 are also not dramatically higher, but the lower confidence bounds are much lower than the quartile bounds. This difference exists because one panelist's (H) estimates are significantly lower than the remaining panelists' estimates.

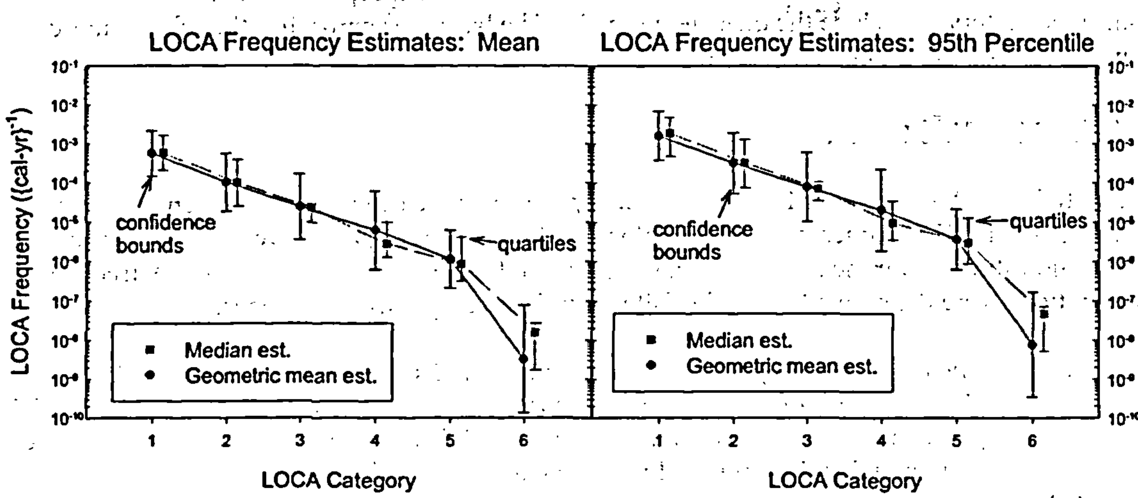

Figure 7.33 Measures of Panel Diversity for BWR LOCA Frequencies at 25 Years

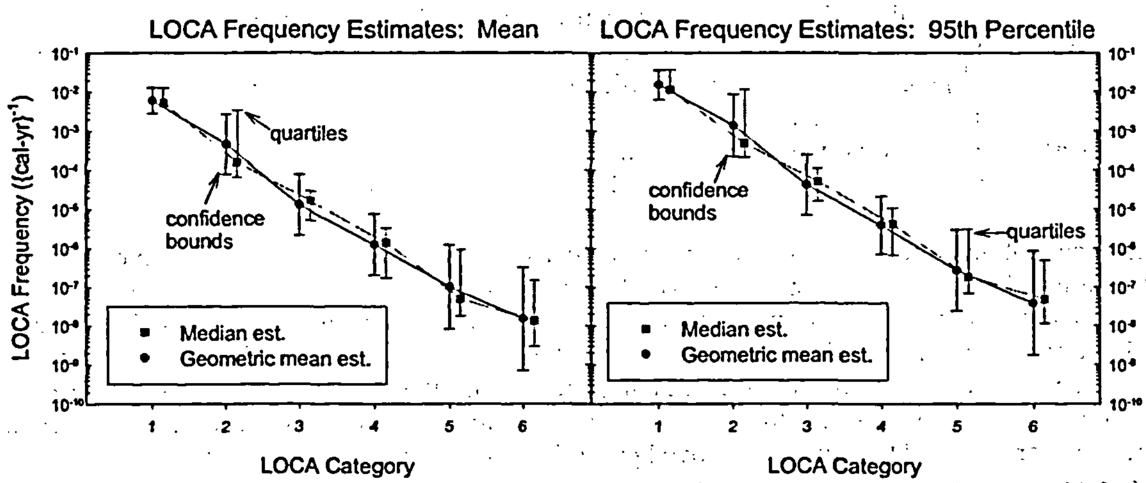

Figure 7.34 Measures of Panel Diversity for PWR LOCA Frequencies at 25 Years.

The PWR mean and 95th percentile diversity measures (Figure 7.34) exhibit many trends similar to the previous BWR results. The baseline confidence bounds are generally larger than the quartile bounds and differences tend to increase slightly with LOCA size as the uncertainty among the individual estimates increases. The only exception to these trends is LOCA Category 2 where the quartile interval is actually larger than the confidence interval. This occurs because two panelists expect relatively high LOCA frequencies, due to PWSCC concerns (Section 6.3.2), when compared with the rest of the group. These two high estimates have a larger impact on the quartile bounds and result in skewed quartiles with respect to the median. The upper quartile bounds for LOCA Categories 5 and 6 are similarly skewed with respect to the median due to two high estimates, but they are not larger than the confidence bounds in these cases.

Figures 7.33 and 7.34 show that, with only a few insignificant exceptions, the statistical confidence intervals include the quartile intervals. The baseline approach is therefore a more conservative measure of panel diversity.

7.7 Summary Results

Estimated LOCA frequencies for the individual panelists were determined and then aggregated to obtain group LOCA frequency estimates, along with measures of panel diversity. The individual and group estimates for the means, medians and 5th and 95th percentiles of the LOCA frequency distributions were determined using the following six assumptions and choices:

- (i) The mid-value, upper bound, and lower bound supplied by the panelists for each elicitation question are assumed to correspond to the median, 95th percentile and 5th percentile, respectively, of a split lognormal distribution, with the mean calculated assuming that the upper tail is truncated at the 99.9th percentile.
- (ii) Some panelist estimates are adjusted to account for possible overconfidence in the elicited uncertainty ranges for the elicitation responses using an error factor adjustment scheme (Section 7.6.2.2).
- (iii) Split lognormal distributions are summed by assuming perfect rank correlation among the individual terms.
- (iv) Aggregation of the individual estimates into a group estimate is performed using the total LOCA frequency estimates determined for each panelist.
- (v) The group estimate of the total LOCA frequency parameters (i.e., median, mean, 5th percentile, and 95th percentile) is defined using the geometric mean of the individual estimates.
- (vi) Panel diversity is characterized by using a two-sided 95% confidence interval based on an assumed lognormal model for the individual estimates.

The resultant individual and group estimates are consistent with the elicitation objectives and structure and are reasonably representative of the panelists' quantitative judgments. In particular, they are not dominated by extreme results, either on the high or low end. In Sections 7.2 through 7.5, the six assumptions and choices above define the *baseline* estimates, with one important difference. Instead of the overconfidence adjustment in (ii), the baseline estimates are determined without any such adjustment. However, the adjusted individual LOCA frequency estimates using (ii) above are deemed to result in improved group LOCA frequency estimates. These improved estimates are referred to as *summary* estimates.

These *summary* LOCA frequency estimates for the current day and end of original license period are provided in Table 7.13 for both BWR and PWR plant types. Table 7.13 is identical to Table 7.8 and is provided as an aide to the reader. The current day median, mean, and 95th percentile summary estimates are graphically presented in Figures 7.35 and 7.36. Figures 7.35 and 7.36 are identical to Figures 7.21 and 7.22 and are provided as an aide to the reader. The 95% confidence intervals calculated for these parameters are also illustrated in this figure. A measure of the individual uncertainty in Table 7.13 and Figures 7.35 and 7.36 is given by the difference between the median and 5th or 95th percentile estimates.

Table 7.13 Total BWR and PWR LOCA Frequencies (After Overconfidence Adjustment using Error Factor Scheme) (reproduced from Table 7.8)

Plant Type	LOCA Size (GPM)	Eff. Break Size (inch)	Current Day Estimate (per cal. year) (25 yr fleet average operation)				Estimate at End of Plant License (per cal. yr.) (40 yr fleet average operation)			
			5^{th} Per.	Median	Mean	95^{th} Per.	5^{th} Per.	Median	Mean	95^{th} Per.
BWR	>100	½	3.1E-05	3.0E-04	6.4E-04	2.1E-03	2.6E-05	2.6E-04	6.0E-04	2.0E-03
	>1,500	1 7/8	2.7E-06	4.8E-05	1.2E-04	4.1E-04	2.2E-06	4.4E-05	1.1E-04	4.1E-04
	>5,000	3 ¼	5.6E-07	9.7E-06	2.8E-05	1.0E-04	4.9E-07	9.8E-06	3.2E-05	1.2E-04
	>25K	7	9.6E-08	2.2E-06	7.3E-06	2.7E-05	8.7E-08	2.3E-06	9.3E-06	3.4E-05
	>100K	18	7.2E-09	2.9E-07	1.5E-06	5.4E-06	6.2E-09	3.1E-07	2.1E-06	7.3E-06
	>500K	41	5.6E-12	3.0E-10	6.4E-09	1.6E-08	6.7E-12	4.0E-10	1.0E-08	2.5E-08
PWR	>100	½	6.0E-04	3.7E-03	6.4E-03	1.8E-02	3.5E-04	2.5E-03	4.7E-03	1.4E-02
	>1,500	1 5/8	7.0E-06	1.4E-04	6.2E-04	2.2E-03	7.6E-06	1.6E-04	7.6E-04	2.7E-03
	>5,000	3	2.0E-07	3.4E-06	1.6E-05	5.8E-05	4.5E-07	7.6E-06	3.6E-05	1.3E-04
	>25K	7	1.3E-08	3.1E-07	1.6E-06	5.7E-06	2.6E-08	6.5E-07	3.6E-06	1.3E-05
	>100K	14	3.8E-10	1.1E-08	1.9E-07	5.2E-07	9.2E-10	2.7E-08	4.6E-07	1.3E-06
	>500K	31	3.3E-11	1.2E-09	3.1E-08	7.8E-08	8.2E-11	2.9E-09	8.1E-08	2.0E-07

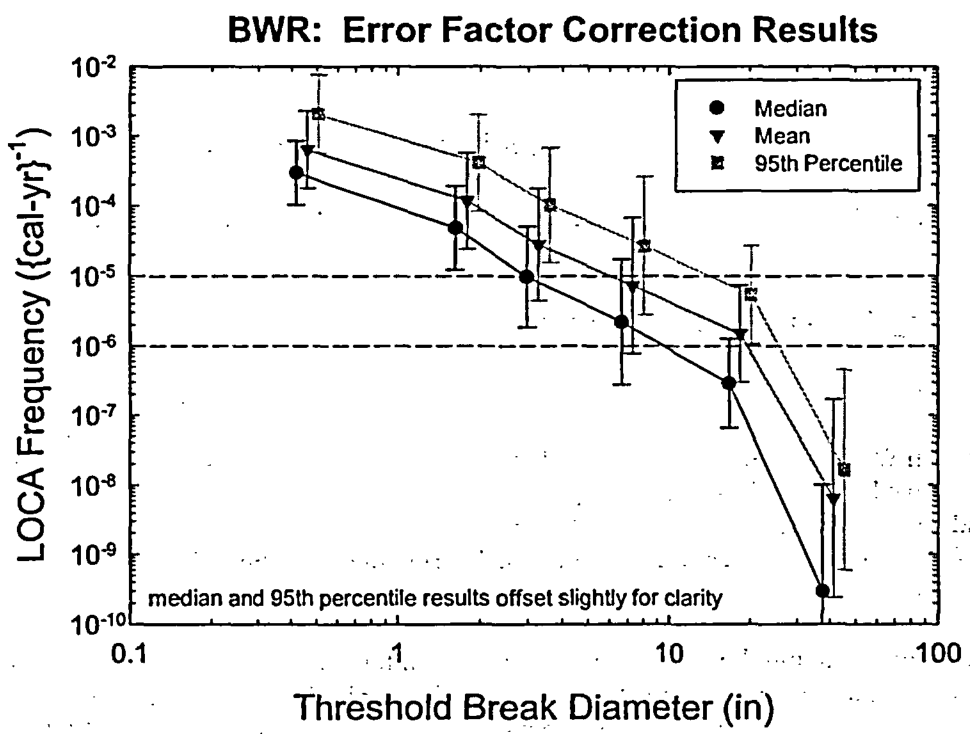

Figure 7.35 BWR LOCA Frequencies with Error Factor Adjustment (reproduced from Figure 7.21)

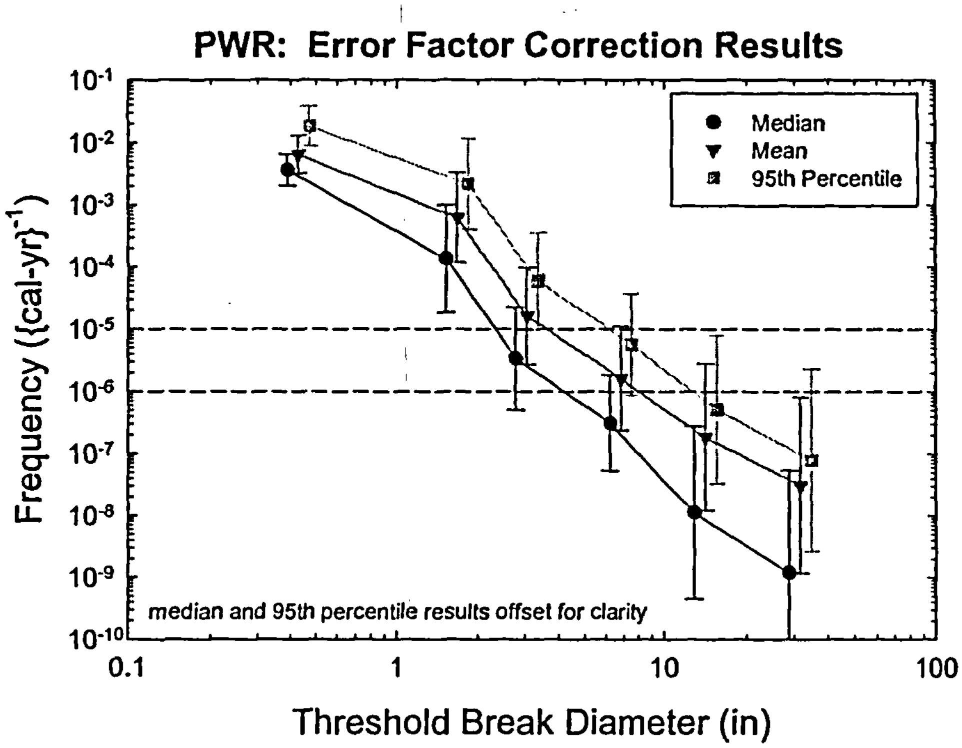

Figure 7.36 PWR LOCA Frequencies with Error Factor Adjustment (reproduced from Figure 7.22)

7.8 Comparisons with Historical Results

In this section, the LOCA frequency results from this elicitation are compared with estimates from past studies. As mentioned in Section 1.2, previous LOCA frequency estimates have been developed in WASH-1400 [7.3], NUREG-1150 [7.4], and NUREG/CR-5750 [7.5]. Additionally, estimates have been developed for each plant type for the individual plant examinations (IPE) [7.6] and determined using an early 1990's piping precursor database as documented in EPRI TR-100380 [7.7]. Prior estimates were also obtained during the pilot elicitation in this exercise as discussed in Section 3.1.

However, before making these comparisons, there are a few important distinctions between these earlier studies and the current estimates which must be emphasized. First, these earlier estimates defined the LOCA range of 100 to 1,500 gpm (380 to 5,700 lpm) for small break LOCAs and 1,500 to 5,000 gpm (5,700 to 19,000 lpm) for medium break LOCAs while the current study defined flow rate thresholds for each LOCA category (Section 3.7). For the purpose of these comparisons, the current summary estimates (Table 7.8) are recalculated to conform to the historical definitions. A more important distinction is that the prior studies generally utilize the generic WASH-1400 and NUREG/CR-5750 definition which distinguishes between medium and large break LOCAs at an approximately 6-inch (152 mm) effective pipe break diameter. Whereas, in this study, Category 3 and 4 LOCAs have minimum effective break diameters of approximately 3 and 7-inches (76 to 178 mm), respectively.

Table 7.14 provides a comparison of the mean values between the current summary elicitation results and the NUREG/CR-5750 results which is the most recent generic study. In this table, the mean results at 25 years from this elicitation are essentially replicated from Table 7.8 after converting the cumulative frequency estimates to represent flow rate ranges consistent with the NUREG/CR-5750 LOCA definitions. The NUREG/CR-5750 ratio values in this table represent the ratio of the NUREG/CR-5750 estimates to the current summary elicitation results. Note that the NUREG/CR-5750 steam generator tube rupture frequency estimates have been added to these results to make them consistent with the LOCA Category 1 elicitation results which include these events.

Table 7.14 also provides a comparison of the current summary elicitation results with the pilot elicitation results for 35 years into the future (i.e., at 60 years of plant life). This was the only time period that was evaluated in the pilot elicitation (Section 3.1). The current elicitation 60-year results are also located in Appendix L. The comparison with the pilot elicitation is reflected by the ratio values of the pilot elicitation results to the current summary elicitation results (Table 7.14). The large break estimates from NUREG/CR-5750 and the pilot elicitation are compared with both the elicitation Category 3 and 4 results because the break size definitions are not consistent.

Table 7.14 Comparison of Current Summary Elicitation Results with Selected Past Studies

Plant Type	Current Summary Elicitation Results			NUREG/CR-5750 Results (25 years of plant operation)			Pilot Elicitation (60 years)	
	LOCA Intervals	Mean Freq. (25 yrs)	Mean Freq. (60 yrs)	LOCA Size	Mean Freq.	Ratio	Mean Freq.	Ratio
BWR	1 – 2	5.2E-04	1.1E-03	SB	4.0E-04	0.77	1.5E-03	1.41
	2 – 4	1.1E-04	1.8E-04	MB	3.0E-05	0.27	9.1E-05	0.50
	> 3	2.8E-05	6.1E-05	LB	2.0E-05	0.71	5.2E-05	0.85
	> 4	7.3E-06	1.8E-05	LB	2.0E-05	2.75	5.2E-05	2.93
PWR	1 – 2	5.8E-03	1.1E-02	SB	7.4E-03	1.27	1.5E-03	0.14
	2 – 4	6.2E-04	1.8E-03	MB	3.0E-05	0.05	6.1E-05	0.03
	> 3	1.6E-05	1.5E-04	LB	4.0E-06	0.25	7.2E-06	0.05
	> 4	1.6E-06	1.8E-05	LB	4.0E-06	2.47	7.2E-06	0.41

Figures 7.37 and 7.38 provide additional selected comparisons of the median, mean, 5th and 95th percentile estimates between the current summary elicitation results and the prior studies. For the PWR medium break (MB) LOCA estimates (Figure 7.37), the current elicitation estimates yield LOCA frequency estimates (mean values) that are very consistent with the WASH-1400 [7.3] and other similar estimates contained in NUREG-1150 [7.4], the IPE submittals [7.6], and EPRI TR-100380 [7.7]. For the BWR large break (LB) LOCA estimates (Figure 7.38), the current summary estimates (mean values) are approximately one to two orders of magnitude less than these prior estimates. As can be seen in Table 7.14, the current summary elicitation estimates are more comparable with the NUREG/CR-5750 estimates [7.5], as the differences in the mean values are generally less than a factor of 4. The one exception is for the current day PWR MB mean value estimate which is about twenty times higher than the NUREG/CR-5750 PWR MB mean value LOCA estimate (Table 7.14 and Figure 7.37). The increase is largely due to current PWR PWSCC concerns (Section 6.3.2) in both piping and non-piping (CRDM) components. However, the expectation that aging will affect intermediate piping to a greater extent than small or large diameter piping (Section 6.3.3) may also contribute to increases in both the BWR and PWR MB LOCA frequencies developed by elicitation (Table 7.14).

In addition, the NUREG/CR-5750 LB LOCA estimates are slightly higher (< than factor of 3) than the current elicitation Category 4 results for both BWRs and PWRs (see Table 7.14). For LB LOCAs, the current elicitation Category 4 results are more compatible with the NUREG/CR-5750 results because the effective break sizes are closer. The Category 3 LOCA frequencies represent break sizes greater than 3 inches. The overall good agreement between the NUREG/CR-5750 and current elicitation estimates is a bit surprising given the markedly different methodologies used to arrive at these results.

These current LOCA frequency estimates are also somewhat comparable to the internal NRC pilot elicitation estimates reported in July 2002. The pilot elicitation used the NUREG/CR-5750 frequencies as the basis for determining the impact of aging on passive system LOCA frequencies over the next 35 years. The pilot elicitation results were between a factor of 2 to 4 times higher than the NUREG/CR-5750 results for all BWR and PWR LOCA sizes except for the PWR small break (SB) LOCAs which are approximately a factor of 5 less than the NUREG/CR-5750 results. However, the PWR SB pilot elicitation results do not include the steam generator tube contribution which is the primary cause for this difference.

For BWRs, the current elicitation SB and LB (Category 4) frequencies are lower than the pilot elicitation results while the current elicitation medium break (MB) frequencies are a factor of 2 higher than the pilot elicitation results. The current elicitation credited the BWR mitigation procedures for IGSCC more so than did the earlier elicitation which may explain why the SB and LB frequencies are lower for the current elicitation. However, the current elicitation concerns for PWSCC cracking in BWR CRDM nozzles results in additional increases that offset the decreases due to perceived IGSCC mitigation effectiveness. Hence, the BWR MB LOCA frequencies for the current elicitation are slightly higher (factor of 2) than the pilot elicitation.

For the PWRs, the differences between the current and pilot elicitation results are generally more substantial. The current elicitation results are consistently higher than the pilot elicitation results for PWRs. The most substantial difference is for the PWR MB LOCAs where the current elicitation results are 30 times higher than the pilot estimates. The current elicitation has the additional benefit of more fully comprehending the severity of CRDM and PWSCC cracking in nickel-based alloys in PWR plants which explains why the current elicitation results are higher than the pilot results.

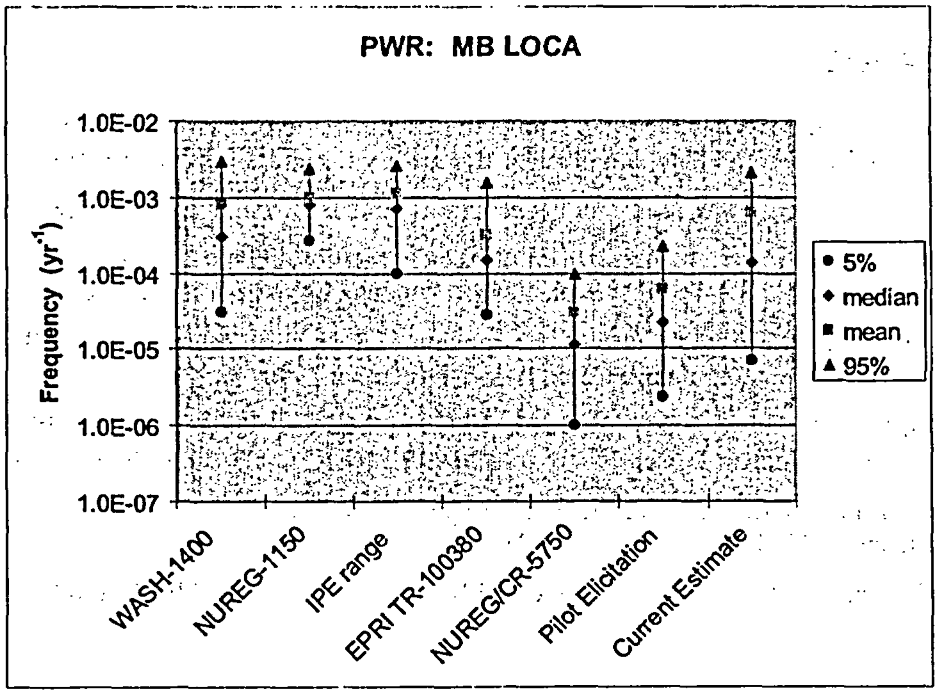

Figure 7.37 Comparison of Current Elicitation Results for PWR Medium Break LOCAs for 25 Years with Results from Selected Past Studies

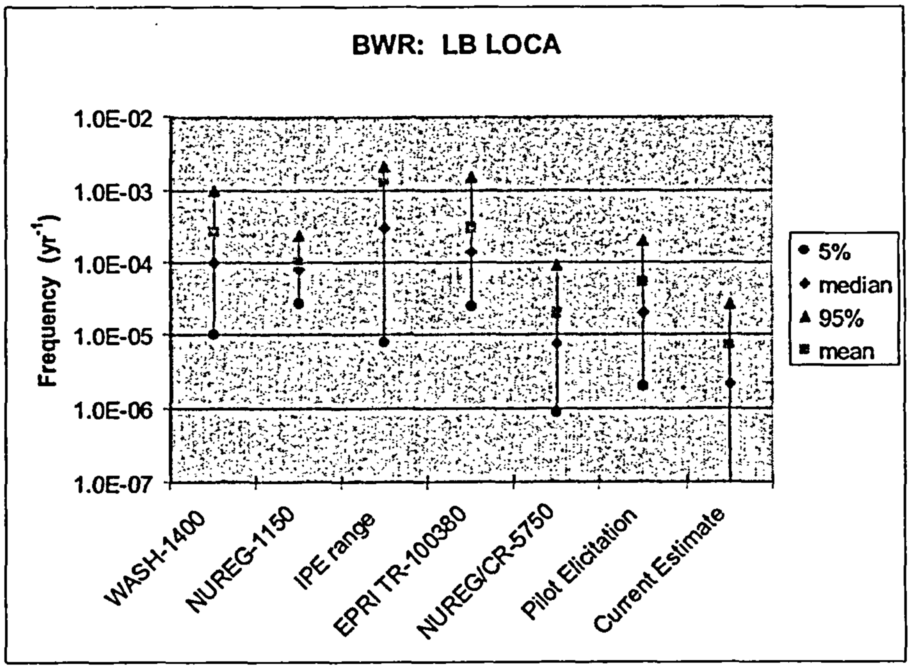

Figure 7.38 Comparison of Current Elicitation Results for BWR Large Break LOCAs for 25 Years with Results from Selected Past Studies

7.9 References

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7.6 "Individual Plant Examination Program: Perspective on Reactor Safety and Plant Performance," NUREG-1560, Vol. 2, Parts 2 – 5, December 1997.

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8. ONGOING WORK

The LOCA frequencies developed in this report will be periodically reevaluated to determine if they need to be updated based on information and knowledge gained subsequent to the expert elicitation. There are several areas of work ongoing or planned for the near future that will support/augment this effort. Those activities include:

(1) updating the pipe failure database as part of the OPDE project,
(2) developing a new probabilistic fracture mechanics (PFM) computer code for predicting the frequencies of various size LOCA events, and
(3) expanding on the efforts of this program through a new international cooperative research program called MERIT (Maximizing Enhancements in Risk Informed Technology).

Additionally, work is ongoing to evaluate the LOCA frequencies due to seismic loading which would be combined with the elicitation LOCA frequencies to gain a more complete understanding of plant challenges. Each of these activities are subsequently described in more detail.

The OECD Pipe Failure Data Exchange (OPDE) project is a collaboration between twelve OECD member countries to capture international nuclear power plant piping event information in a central database. Because of the rarity of piping failures and precursor events, it is crucial to have both an accurate count of these events and a large pool of relevant reactor experience to draw from in order to develop meaningful piping failure frequency estimates. Data captured in the OPDE database allows for examination of LOCA precursor event occurrences versus time directly from operating experience. The database is striving for complete international data starting from 1998. Information from events dating back to the early 1970s will be included for some countries, including the United States. The OPDE database will form part of the technical database used to periodically update LOCA frequency estimates.

As a means of verifying/augmenting the results from this elicitation effort, additional probabilistic analyses will be conducted using a probabilistic LOCA code (PRO-LOCA) that is currently under development. This new code includes many deterministic aspects that have been developed in various NRC piping programs but not included in any of the currently available PFM codes, such as PRAISE, SRRA, or PRODIGAL. The PRO-LOCA code includes improved crack initiation/crack growth models, additional weld residual stress solutions, improved fracture models, improved leak-rate models with newly developed crack morphology parameters, and new methods for addressing multiple crack initiation sites and crack coalescence. While the code currently estimates frequencies through direct Monte-Carlo simulation, new more efficient processing routines are currently under development. Once the development work on this code has been completed and the necessary benchmarking and quality assurance checks completed, the code will be exercised for a series of test cases as a means of validating the results from this elicitation effort. The final objective will be to combine this code, along with the operating experience database, to either directly calculate LOCA frequencies or use as a basis for conducting a subsequent elicitation to update LOCA frequency estimates contained in this report.

Additionally, a new international cooperative research program (MERIT) is being developed as a means of further developing the tools for making these types of probabilistic-based risk-informed decisions. The two main outcomes of the MERIT program as related to the determination of LOCA frequencies are the continued development of the previously mentioned probabilistic LOCA code (PRO-LOCA) and the further assessment of weld residual stresses and their impact on stress corrosion cracking. As part the continued development of the PRO-LOCA code, work within the MERIT program will focus on LOCA frequency contributions from non-piping component degradation. Also, an attempt will be made to reach an international consensus as to a set of standardized procedures for making these types of assessments. With regards to the further assessment of weld residual stresses and their impact on stress corrosion cracking, some of the key activities to be undertaken as part of the MERIT program are: (1) developing new solutions for additional bimetal weld locations and weld procedures, (2) examining the effect of weld repairs on the residual stress state, (3) identifying mitigation techniques/strategies for the re-qualification of leak-before-break (LBB) for bimetal welds susceptible to primary water stress corrosion cracking, and (4) optimizing weld repair techniques to minimize weld residual stresses.

Finally, the LOCA elicitation results were for normal operating conditions and associated transients. Service Level D transients such as seismic loading or severe accident scenarios were not considered in the LOCA elicitation efforts. Consequently a number of different severe accident and Service Level D design (and non-design) conditions were examined by NRC staff. Of those, seismic loading was considered as the most prominent. A seismic LOCA study is underway to combine operating experience, probabilistic risk analysis insights, and piping degradation models to estimate the frequency associated with degraded piping. Initial results have supported the robustness of unflawed piping under seismic loading and indicate that this is not a significant LOCA frequency contributor. However, results also indicate that flaws larger than a certain size may provide significant frequency contributions. This critical size is plant specific due to seismic hazard curve variability and piping system design differences. A possible outcome is that flaws large than a plant's threshold size would need to be repaired or replaced to ensure that the LOCA frequency contributions due to seismic loading are minimized. Additional work is ongoing to decrease the conservatism associated with the piping degradation model so that the critical flaw size can be more accurately determined.

9. SUMMARY OF RESULTS AND CONCLUSIONS

An expert elicitation process has been used to consolidate service history data and insights from PFM studies with knowledge of plant design, operation, and material performance to develop LOCA frequency estimates. The approach allowed the panelists to decompose complex issues which impact LOCA frequencies into fundamental elements which are easier to assess. Quantitative estimates were provided to the panelists for precursor LOCA events associated with degradation in piping and non-piping components. Additional quantification was conducted to develop frequencies associated with well-defined base case conditions. The panelists extrapolated from this information based on their knowledge of passive system component failure to develop LOCA frequency contributions for potential piping and non-piping passive systems failures. The panelist input was processed to develop estimates that reflect individual panelist uncertainty and panel diversity.

The panelists provided qualitative insights on a number of topics including evaluation of base case predictive methodologies; the effect of safety culture on LOCA frequencies; important degradation mechanisms; piping and non-piping LOCA frequency contributors; the effects of component size and operating time on LOCA frequencies; the influence of mitigation and maintenance; and uncertainty in making estimations, to name a few. The insights provided by the panel are reasonably consistent. Most members believe that service history-based analyses provide the most accurate basis for developing current day (i.e., 25 year) frequency estimates associated with the simplified base case conditions. However, probabilistic fracture mechanics (PFM) studies offer valuable insights on the possible future influence of degradation mechanisms. The use of sensitivity analysis, using either method, is also valuable for identifying important contributing issues. Many panelists believe that combining information from both techniques is necessary to make accurate frequency projections, even for the simplified base case conditions.

There are several competing variables that affect the general safety culture of the industry and regulatory bodies. However, the panel members generally expressed the belief that the future safety culture will not differ dramatically from the current culture. In fact, most participants expect a small improvement due mainly to continued experience and continued technology advancements. Because of this general expectation, the facilitation team decided not to adjust the LOCA frequency estimates provided by the panel members to account for safety culture effects. The panelists also generally expressed that utility and regulatory safety cultures are highly correlated. However, many panelists do believe that effects of safety culture are cyclical and safety culture can significantly affect LOCA frequencies at specific plants, either positively or negatively.

Many participants believe that the number of precursor events (e.g., cracks and leaks) is a good barometer of the LOCA susceptibility for the associated degradation mechanism. Welds were almost universally recognized as important due to high residual stress, preferential attack of many mechanisms, and the increased defect likelihood. Nozzle, elbows, and tees were also thought to be important locations. The participants generally identified thermal fatigue, stress corrosion cracking (SCC), and mechanical fatigue as the degradation mechanisms which most significantly contribute to LOCA frequencies in PWR plants. These mechanisms and flow accelerated corrosion (FAC) are important LOCA frequency contributors for BWR plants. The panel consensus is that the susceptibility of BWR piping systems to IGSCC is greatly reduced compared to the past. However, this mechanism remains an important LOCA contributor in many of the panelists' estimates. The panelists were concerned that PWSCC is an important contributor to current day PWR LOCA frequency estimates. However, most panelists expect that this mechanism will be mitigated within the next 15 years.

The panelists generally believe that complete rupture of a smaller pipe or non-piping component is more likely than an equivalent size opening in a larger pipe because of the increased susceptibility to fabrication or service cracking. Additionally, smaller bore piping, compared to larger bore piping, is more likely to have fabrication flaws, has lower inspection quality and quantity, and is more susceptible to external failure mechanisms arising from human error. This is a primary reason why the biggest contributors in each LOCA category tend to be the smallest pipes which can lead to that size LOCA. There is substantially more disagreement about the most likely contributing systems for the intermediate-size (Category 3, 4) LOCA Categories because of the sheer number of possible contributing systems. However, many panelists thought that aging may have the greatest effect on intermediate diameter (6 to 14-inch) piping systems.

Generally, non-piping LOCA frequency assessment is believed to be more challenging than the piping assessments. There are multiple non-piping components to consider, each associated with different operating requirements, designs, materials, and inspection considerations. Furthermore, there is also not the wealth of precursor data available for the non-piping components compared with piping. However, the panelists believe that non-piping components contribute significantly to Category 1 and 2 LOCA frequencies. For PWRs, non-piping components are the dominant contributor to Category 1 and 2 LOCA frequencies due to steam generator ruptures and CRDM cracking concerns. Postulated non-piping failures contribute much less significantly to Category 3 – 5 frequency estimates. Non-piping contributions become important for LOCA Category 6 because there is little (PWR) and no (BWR) piping which can result in this size flow rate at failure. Non-piping contributions are more significant in PWR plants because of the increased number of LOCA-sensitive components and past and current experience with degradation in these components.

The frequency estimates are not expected to change dramatically over the next fifteen years, or even the next thirty-five years. The differences between the current day estimates and estimates for the next fifteen years are generally less than a factor of 3. Over the next thirty-five years, increases in the mean frequencies of less than factors of 10 are expected for LOCA Categories 1 – 4 and less than factors of 20 for LOCA Categories 5 and 6. However, these longer term increases are largely a function of increased uncertainty among the panelists and increases in the median estimates are less. There are several competing factors which affect future trends. LOCA frequency decreases will be driven by expected improvements in mitigation strategies, inspection techniques, material replacement, and the continued reduction in fabrication-related problems. LOCA frequency increases will be driven primarily by continued aging, but will also be exacerbated by improper maintenance and other human errors.

While there is general qualitative agreement among the panelists about important issues, the quantitative estimates were much more difficult to assess because of the underlying scientific uncertainty and the lack of truly relevant LOCA data. The panel members generally expressed greater uncertainty in their predictions as the LOCA size increased. As mentioned, most panelists also believe that uncertainty increases with future operating time. Both trends are expected and justified because of the greater extrapolation required of service data. Panelist uncertainty was generally similar for BWR and PWR plants. There are also significant differences among the panelists' estimates. However, panel variability or diversity as expressed by the quartiles or confidence bounds is generally much less than individual panelists' uncertainties, is fairly constant with LOCA size, and is similar for BWR and PWR estimates.

Baseline LOCA frequency estimates for the 5th percentile, median, mean and 95th percentile were determined from each panelist's elicitation responses. These individual responses were then aggregated by calculating the geometric mean of each of these underlying parameters. Group variability was estimated by calculating 95% confidence bounds for each of the aggregated frequency parameters (i.e., median, mean, 5th and 95th percentiles) assuming that the panelist variability is lognormally distributed. Generally, the baseline elicitation estimates for these frequency distribution parameters are much less than the WASH-1400 estimates and similar to the NUREG/CR-5750 estimates. The elicitation and NUREG/CR-5750 small break BWR and PWR LOCA frequency estimates are similar once the steam generator tube rupture frequencies are added to the NUREG/CR-5750 PWR results. The NUREG/CR-5750 large break LOCA break size of 5 to 6 inches (127 to 152 mm) is most consistent with the elicitation LOCA Category 4 break size of 7 inches (178 mm). The NUREG/CR-5750 LB LOCA estimates are only slightly greater (approximately a factor of 3) than the LOCA Category 4 results.

The largest difference between the NUREG/CR-5750 and elicitation results is for the medium break (MB) LOCAs. The elicitation MB LOCA estimates are higher than the NUREG/CR-5750 estimates by a factor of approximately 4 and 20 for BWR and PWR plant types, respectively. These increases are largely attributable to concerns about the degradation sensitivity and large quantity of intermediate size piping (3" to 14") in current plants. PWR increases are also partly due to PWSCC concerns in piping and non-piping (CRDM) components. The generally good agreement between the NUREG/CR-5750 and current elicitation estimates is somewhat surprising given the markedly different methodologies used to arrive at these results.

A number of sensitivity analyses were conducted to examine the robustness of the quantitative results to the underlying analysis procedure. Sensitivity analyses investigated the effect of distribution shape on the means, and the effects of correlation structure, panel diversity measure, panelist overconfidence and aggregation measure on the estimated parameters. The mean calculation used a split lognormal distribution truncated at the 99.9th percentile to obtain reasonably conservative values. The choice of distribution shape is not significant for distributions when the error factor is less than 100, which represents the bulk of the elicitation responses. Split log-triangular and the baseline truncated split lognormal distributions represent the most credible bounds for characterizing the elicitation responses. The difference in the means calculated for these distributions is less than a factor of 5.5 for distributions with error factors up to 1000.

The correlation structure assumed maximal correlation, which is reasonably representative of the elicitation structure and provides conservative 95th percentile estimates. The mean is unaffected by the assumed correlation structure. However, a bounding independent correlation structure leads to median and 5th percentile estimates that are as much as one and two orders of magnitude higher than the estimates calculated using the assumed correlation structure for selected distributions. These differences increase with the error factor of the underlying distribution and the number of distributions which contribute to the total LOCA frequency estimates for each panelist. However, because the relatively large sensitivity to the correlation structure is manifested for only some of the panelists, the sensitivity to the correlation structure will be less for the aggregated results.

The baseline approach to panel diversity used the 95% confidence bounds for each associated LOCA frequency parameter (i.e., the median, mean, 95th percentile). The confidence bounds are based on the variance of all the estimates and are relatively unaffected by one or two extreme individual results. In general, the confidence intervals are conservatively comparable to the interquartile ranges of the individual results. That is, the 90% confidence intervals based on the 5% and 95% confidence bounds encompass at least 50% of all the individual estimates for each parameter.

Ad hoc overconfidence adjustments are difficult to justify and can result in large, unsupportable increases in the frequency estimates The blanket and more severe targeted adjustments generally lead to greatly skewed estimates (e.g., the mean is much greater than the 95th percentile) that are not consistent with the elicitation structure and panelist response characteristics. Additionally, the blanket adjustments can lead to high mean frequency estimates that are not supported by the operating experience data. The targeted adjustment of only those panelists whose estimates express a relatively large certainty can be supported by the post-adjusted results. All these adjustment schemes require ad hoc assumptions about which panelists to adjust and the appropriate level of adjustment. The error factor adjustment scheme is less subject to these drawbacks. This adjustment scheme provides a variable overconfidence adjustment as a function of LOCA size and plant type, and is greatest for those panelists that expressed the least amount of uncertainty compared to the group average. This adjustment leads to a relatively small increase in the baseline LOCA frequency estimates and the results are comparable to the least severe targeted adjustment.

Finally, there is a large sensitivity to the method used to aggregate the individual panelist estimates to obtain group estimates. A geometric mean aggregation, rather than an arithmetic mean or mixture distribution aggregation, was chosen in this case because it was a better representation of the consensus-type group estimate sought by this study.

There was a wide range of individual responses for the LOCA frequency expert elicitation. Use of the arithmetic mean or mixture distribution aggregation method in this case would produce group estimates which are dominated by the one or two highest individual estimates and would not represent a consensus-type group estimate, which was sought for this study.

For this study, the geometric mean aggregation develops frequency estimates which are a better representation of consensus-type results and better express the expert panel's state of knowledge regarding LOCA frequencies in a manner consistent with the elicitation process used to obtain the panel responses, the nature of these responses and the stated study objectives.

This study does not make a recommendation as to whether the LOCA frequency estimates using the baseline analysis procedure or using a particular sensitivity analysis should be used in any particular application. For the reasons stated above, the geometric mean aggregated results after using the error factor scheme to individually adjust for overconfidence (i.e., the *summary* estimates) are believed to be a reasonable representation of the expert panel's current state of knowledge regarding LOCA frequencies for the stated study objectives. However, sensitivity studies demonstrate that alternative analyses can lead to significantly different LOCA frequency estimates. Therefore, the purposes and context of any application must be considered when determining the applicability of any set of study results. While this places an additional burden on the users of the results, those users are in the best position to judge the extent to which the study results can be used for their particular applications.

APPENDIX A

PANEL MEMBER QUALIFICATIONS

APPENDIX A

PANEL MEMBER QUALIFICATIONS

This appendix contains short resumes/biographical sketches for each of the panel members demonstrating their qualifications and credentials for this expert elicitation.

BRUCE BISHOP
PRINCIPAL ENGINEER
RELIABILITY AND RISK ASSESSMENT GROUP
WESTINGHOUSE ELECTRIC COMPANY'S NUCLEAR SERVICE DIVISION
PITTSBURGH, PENNSYLVANIA

Mr. Bishop has been at Westinghouse for almost 35 years, working primarily in the area of structural reliability analysis, initially on breeder reactor core components and then on light water reactor pressure vessels, piping and other primary loop components. During this time, Mr. Bishop developed and applied the structural reliability and risk assessment (SRRA) models, methods and software for probabilistic fracture mechanics (PFM) analyses supporting a number of risk informed inspection initiatives. Included are the SRRA applications for most plant piping, reactor internals and reactor coolant pump components, and the irradiated belt line region, head penetration nozzles and nozzle inner radius region of the reactor pressure vessel. Mr. Bishop has been the recipient of six George Westinghouse Signature Awards for Engineering Excellence and two Special Recognition Awards from the ASME Pressure Vessel and Piping Division for conference tutorials on Application of Probabilistic Structural Mechanics. He has participated in the development and presentation of numerous Westinghouse technical reports and more than 35 publications on application of structural reliability to risk informed decisions, including a chapter on pressure vessel and piping applications in the *Probabilistic Structural Mechanics Handbook*. Mr. Bishop, a member of the editorial boards for *Reliability Engineering and System Safety* and *Nuclear Engineering and Design,* is currently a member of the ASME Safety Engineering and Risk Analysis Division and the ASME Codes and Standards Working Groups on Implementation of Risk-Based Inspection and Operating Plant Criteria (for reactor pressure vessel integrity issues). He also actively participates in the PWR Materials Reliability Program Issue Task Groups on Reactor Vessel Integrity and Alloy-600 Issues, including the PFM analyses of the reactor pressure vessel during postulated pressurized thermal shock events and the Alloy 82/182 butt welds that are subject to primary water stress corrosion cracking.

VIC CHAPMAN
O J V CONSULTANCY, LTD.
DERBY, ENGLAND

Mr. Chapman has a Diploma in Engineering and an Honors Degree in mathematics. He has worked in the pressure system design and maintenance field for over thirty years. Over this period he has gained experience in the design and fracture analysis of pressure vessels, the study of material properties and the statistical interpretation of data. Over the past twenty-five years he has concentrated on the area of probabilistic fracture mechanics, risk based decision-making, risk based inspection and inspection qualification. He was responsible for the introduction of the Risk-Based In-Service Inspection program into the Royal Naval Nuclear Fleet.

Mr Chapman retired from Rolls Royce (Naval Nuclear Division) four years ago and became an independent consultant, forming his own company, O J V Consultancy Ltd. In addition to his consultancy work, which is funded by many organizations and international bodies, he has played a leading role on several international committees:

He is currently Chairmen of ENIQ-TGR (European Network for Inspection & Qualification-Task Group on Risk).

A member of the European program NURBIM (Nuclear Risk-Based Inspection Methodology).

A member of the ASME Research Committee on Risk Technology.

He chaired the European program EURIS (European Network of Risk-Informed In-Service Inspection).

Was a member of the ASME Research task force on Risk Based Inspection – Developed of Guidelines.

Was a member of the UK Technical Advisory Group on Structural Integrity (TAGSI) sub-committee on defects in welds.

GUY DEBOO
SENIOR STAFF ENGINEER
EXELON NUCLEAR
CHICAGO, ILLINOIS

Mr. Deboo has 28 years experience working in the nuclear power generation field. Mr. DeBoo's recent experience includes fatigue, crack growth and flaw stability analyses necessary to demonstrate operability for most power plant components. These evaluations would include root cause and remaining life determinations. He has extensive experience with IGSCC and other material degradation issues. He also has performed and supervised functionality and operability evaluations of systems and components to address unanticipated operating events or conditions, which do not meet inspection or test requirements.

During his 28 years in nuclear power generation, Mr. DeBoo has worked on three major nuclear projects (six units) including all design, inspection and testing phases leading to commercial operation. This experience included system and component seismic qualification, component fatigue qualification, licensing and design review for fuel load, special analytical assessments of the safety significance of installation discrepancies, and system/equipment/component functionality/operability evaluations resulting from startup and operating test programs.

Mr. DeBoo has extensive experience in the evaluation of fatigue-related problems, material degradation issues and the assessment of remaining life in vessels, piping and supports for nuclear applications. He has performed safety evaluations for unanticipated operating events, and has developed plant-unique acceptance criteria to permit continued operation. Those events include fluid transients, thermally stratified flows, and flow-induced vibration. He supervised the AMSE Class 1 piping fatigue analysis on two boiling water reactor (BWR) units and two pressurized water reactor (PWR) units.

Mr. Deboo has a B.S. in Mechanical Engineering from Northwestern University and a M.S. in Mechanical Engineering from the University of Illinois. He is a member of the American Society of Mechanical Engineers. Currently he is serving on Section XI of the ASME Boiler and Pressure Vessel Code as a member of the Working Group on Pipe Flaw Evaluations and as the Secretary of the Working Group on Flaw Evaluation.

WILLIAM GALYEAN
SENIOR PRA ANALYST
IDAHO NATIONAL ENGINEERING AND ENVIRONMENTAL LABORATORY
IDAHO FALLS, IDAHO

Mr. Galyean has over 25 years experience in performing probabilistic risk assessments (PRA) on commercial nuclear power plants. After earning his Bachelor of Science degree in Physics at Millersville State College in 1976, he went to the University of New Mexico where he obtained his Master of Science degree in Nuclear Engineering in 1978.

Mr. Galyean is a senior PRA analyst at the Idaho National Engineering and Environmental Laboratory (INEEL), which he joined in 1986. In recent years, Mr. Galyean was the Principal Investigator of two large NRC-sponsored PRA-related programs. The latest entailed performing detailed statistical analyses on nuclear power plant reliability data collected from operating experience. This data was then used to estimate both the historical system reliability for operational missions actually performed and the expected reliability for postulated risk-significant missions. The earlier program was a research program aimed at supporting the resolution of U.S. NRC Generic Issue 105, "Interfacing System Loss-of-Coolant Accidents at Light Water Reactors." This program integrated many disciplines in the area of risk analysis, including: human reliability, thermal-hydraulics, consequences, external events, and stress analysis. In addition a number of innovative techniques were developed for generating human error probabilities and fluid system rupture probabilities. Mr. Galyean has also developed and teaches 1-week courses on PRA modeling techniques and on Level-2 PRA, and developed low-power and shutdown (LP/SD) specific PRAs as part of the Standardized Plant Analysis Risk (SPAR) model program. Other activities include serving on an NRC-sponsored expert panel that was formed to produce updated estimates of loss of coolant accident frequencies, supporting the independent validation of a probabilistic fracture mechanics computer code (FAVOR), participating as the INEEL project manager on a multi-company, multi-lab project funded through the DOE NERI program performing a risk-informed assessment of new reactor design requirements. He has also provided significant contributions to a number of other risk/reliability related programs, including NUCLARR (Nuclear Computerized Library for Assessing Reactor Reliability), a PC-based databank of reliability data for both hardware and human actions; a reliability analysis of the INEEL site power distribution system; and an analysis of the risk significance of possible operator actions for managing severe accidents at a commercial nuclear power plant.

Prior to joining INEEL Mr. Galyean worked for Falcon Research and Development Company, Science Application International Corporation, and the NUS Corporation.

Mr. Galyean is a member of the American Nuclear Society, and has served on an International Atomic Energy Agency (IAEA) review teams (International Peer Review Service – IPERS) that reviewed a Level-1 PRA on the Borssele NPP (Dutch) and a Level-2 PRA on the Krsko NPP (Slovenian).

KAREN GOTT
SWEDISH NUCLEAR POWER INSPECTORATE

Dr. Gott studied metallurgy and materials science at Imperial College, London.

During the more than 20 years she has worked in Studsvik Dr. Gott studied many aspects of the environmental effects on structural materials in nuclear power plants, both through contract research projects and failure analysis. She has held a number of different types of position whilst at Studsvik including project manager, marketing manager and manger of the reactor chemistry group. She was also on periodic loan to a US subsidiary in Richland, WA, to help them establish laboratory support for their decontamination services.

The main areas of her research activities were

- Creep crack formation in stainless steels (mechanical testing, electron and light optical metallography)
- Fracture mechanics (corrosion fatigue, residual stress measurement, non-destructive testing)
- Reactor chemistry (PWR and BWR chemistry, activity build-up including field measurements, decontamination)
- Reactor materials (surveillance testing, failure analysis, metallography of Inconel 182)

In her current position at the Swedish Nuclear Power Inspectorate she has continued to work in the field of environmental degradation of nuclear power plant structural materials. The work covers both the regulatory and the research aspects. On the regulatory side she is involved in the development of regulations, inspection and safety evaluations that form the basis for decisions based on Swedish law and regulations. One of her responsibilities includes the management of the materials and chemistry research area for the Inspectorate. In addition she has built a database covering operationally induced failures and damage to mechanical components in the Swedish nuclear fleet and is responsible for its maintenance and the associated analysis of failure cases. In 2003 she was on a six month job rotation to the Materials Engineering Branch of NRR working amongst other things on primary water stress corrosion cracking problems.

She is a member of the international conference committee which arranges the regular water chemistry conferences in the nuclear field, and has also acted on the international committee for the Fontevraud conference in France. She served as chairperson of the steering committees of two large international projects concerning irradiation assisted stress corrosion cracking and the establishment of a pipe failure database.

DAVID HARRIS
PRINCIPAL ENGINEER AND VICE-PRESIDENT
ENGINEERING MECHANICS TECHNOLOGY
SAN JOSE, CALIFORNIA

Dr. Harris is a principal engineer at Engineering Mechanics Technology (EMT), Inc. and has some 30 years of experience in fracture mechanics and solid mechanics analysis and applications. His background is in mechanical engineering, and he has extensive experience in probabilistic structural mechanics, especially as related to fracture mechanics.

Dr. Harris began his career as a mechanical engineer at Lawrence Radiation Laboratory (LRL) in Livermore, California. After several years at LRL, Dr. Harris joined one of the earliest vendors of acoustic emission instrumentation, Dunegan Corporation, as Director of Research. After four years at Dunegan Corporation, Dr. Harris joined Science Application, Inc. (SAI, now known as SAIC) in their Palo Alto office. During his seven years at SAI, Dr. Harris' efforts included performing some of the earliest applications of probabilistic fracture mechanics (PFM) to nuclear reactor piping. He was the principal developer of the PRAISE code, which was developed for the US Nuclear Regulatory Commission. The PRAISE code is based on PFM and is one of the most widely applied tools for evaluation of the reliability of weldments in nuclear reactor piping.

Dr. Harris worked at Failure Analysis Associates for over ten years. During this time he developed and applied fracture mechanics to a wide variety of problems, ranging from railroad wheels to rocket ship engines. These efforts included both deterministic and probabilistic aspects, and involved both computer software development and applications to industrial problems. He was the manager of the Fracture Mechanics section, which included some five engineers involved in fracture mechanics and related finite element stress analysis. He was the principal developer of the NASCRAC code, which is a general purpose code for deterministic analysis of crack growth that was developed for NASA.

Dr. Harris is currently a vice-president and principal engineer at EMT a company that he was involved in founding some seven years ago. EMT is an engineering consulting firm that specializes in fracture mechanics, life prediction and related software – both deterministic and probabilistic. Efforts at EMT include development of the PRAISE code in Windows (WinPRAISE), including enhancements to make the software easier to use in routine applications, and expansion of PRAISE to include crack initiation due to cyclic loading in air and water environments. He was also involved in the development of commercial fracture mechanics software – including linear and nonlinear SmartCrack. BLESS is a code for analysis of reliability of headers and piping in fossil-fired power plants that was developed with support of the Electric Power Research Institute (EPRI). BLESS is a physics-based model that considers both crack initiation and growth due to creep and cyclic loading.

Dr. Harris has been involved in ASME activities related to reliability considerations in design and inspection of nuclear reactor piping. He was an original member of the ASME Research Task Force on Risk-Based Inspection Guidelines, and was the editor of Volume 3 of a series of reports published by this committee. Volume 3 was on applications to fossil fired power plants. He is currently vice chairman of the Risk Technology Committee of the ASME. Dr. Harris is a member of ASTM as well as ASME. He has nearly 100 publications in the open literature, primarily in the areas of acoustic emission and fracture mechanics. He received a B.S. and M.S. in mechanical engineering from the University of Washington and a Ph.D. in applied mechanics from Stanford University.

BENGT LYDELL
SUPERVISOR
ERIN® ENGINEERING AND RESEARCH, INC.
WALNUT CREEK, CALIFORNIA

Mr. Lydell has 30 years of risk and reliability analysis experience. Prior to joining ERIN®, he held positions with the Swedish Nuclear Power Inspectorate (SKI), Pickard, Lowe and Garrick, Inc., and NUS Corporation. Mr. Lydell has extensive, practical experience with applied quantitative risk assessment. In various capacities (systems analyst, human reliability analyst, independent reviewer), he has supported numerous domestic and foreign PSA projects (Level 1 and 2, and internal flooding). As an independent contractor, during the period 1993-99 he performed R&D in piping reliability analysis for the oil and gas and nuclear industries. This work explored field experience data and its role in quantitative piping reliability analysis, including the interfaces between PSA requirements and probabilistic fracture mechanics. The SKI pipe failure database resulted from this work. Under contract to SKI and BKAB (a Swedish utility), during 1998-99 he performed a pilot LOCA-frequency study; a summary report is published as SKI Report 98:30 (May 1999). This particular study was commissioned to address the feasibility of applying BWR pipe service experience data to the estimation of plant-specific LOCA frequencies. The SKI pipe failure database formed the basis for the OECD Nuclear Energy Agency's "OECD Pipe Failure Data Exchange" Project (OPDE), an international forum for the exchange of pipe failure information. Managed by Mr. Lydell, a clearinghouse is operating the OPDE database and provides the quality assurance function.

SAM RANGANATH
XGEN ENGINEERING
SAN JOSE, CALIFORNIA

Dr. Ranganath has spent 30 years working with Boiling Water Reactors (BWR). He spent over 28 years working on BWRs at General Electric before moving to set up a consulting company - XGEN Engineering that provides fracture mechanics, materials and stress analysis services to the power industry. His last position at GE Nuclear Energy was Engineering Fellow and Manager, Hardware Design. He has also taught graduate courses in structural mechanics and materials at Santa Clara University and San Jose State University for over 10 years

Dr. Ranganath has a Ph. D in Engineering from Brown University and a Masters degree in Business Administration from Santa Clara University. He is a Fellow of the American Society of Mechanical Engineers. He has also been an Engineering Fellow at GE Nuclear Energy and was elected to the Engineering Hall of Fame at GE Nuclear Energy.

Dr. Ranganath has been active in the development of the ASME Code for over 20 years. He led the effort on developing flaw acceptance rules for austenitic piping in the ASME Code. He was also played a major role in developing improved rules for seismic design of nuclear power plant piping. He has also been the principal investigator on several materials research programs at the Electric Power Research Institute. He has also been active in the BWR Vessel and Internals Program (BWRVIP) and has been the lead author of several Inspection and Evaluation documents for BWR internal components. His expertise in BWR issues such as IGSCC, corrosion fatigue, fracture mechanics, ASME Section XI and Section III Codes, repair hardware design and BWR design is important in assuring that the LOCA frequency conclusions reflect BWR field experience.

PETE RICCARDELLA
SENIOR ASSOCIATE
STRUCTURAL INTEGRITY ASSOCIATES
GREENWOOD VILLAGE, COLORADO

Pete Riccardella received his PhD. from Carnegie Mellon University in 1973 and is an expert in the area of structural integrity of nuclear power plant components. He co-founded Structural Integrity Associates in 1983, and has contributed to the diagnosis and correction of several critical industry problems, including:

- Feedwater nozzle cracking in boiling water reactors
- Stress corrosion cracking in boiling water reactor piping and internals
- Irradiation embrittlement of nuclear reactor vessels
- Primary water stress corrosion cracking in pressurized water reactors
- Turbine-generator cracking and failures.

Dr. Riccardella has been principal investigator for a number of EPRI projects that led to advancements and cost savings for the industry. These include the **FatiguePro** fatigue monitoring system, the **RRingLife** software for turbine-generator retaining ring evaluation, **Risk-Informed Inservice Inspection** methodology for nuclear power plants, and several **Probabilistic Fracture Mechanics** applications to plant cracking issues. He has led major failure analysis efforts on electric utility equipment ranging from transmission towers to turbine-generator components and has testified as an expert witness in litigation related to such failures.

He has also been a prime mover on the ASME Nuclear Inservice Inspection Code in the development of evaluation procedures and acceptance standards for flaws detected during inspections. In 2002 he became an honorary member of the ASME Section XI Subcommittee on Inservice Inspection, after serving for over twenty years as a member of that committee.

In 2003, Dr. Riccardella was elected a Fellow of ASME International.

HELMUT SCHULZ
DEPARTMENT HEAD – COMPONENT INTEGRITY
GESELLSCHAFT FÜR ANLAGEN- UND REAKTORSICHERHEIT (GRS)
KÖLN, GERMANY

Mr. Schulz has over 35 years of experience in nuclear engineering, structural and fracture mechanics, materials, and nuclear safety. At GRS he is Head of the Department of Components Integrity and was/is a member of various national and international advisory bodies regarding nuclear safety, component integrity, and codes and standards. In this role, he is responsible for the safety assessment of nuclear components and structures, as well as related research and verification of fracture mechanics codes for safety applications.

Prior to joining GRS, he worked for Gesellschaft für Reaktorsicherheit where he held various staff positions and project management responsibilities for PWR safety assessment work. Prior to that he was with United Nuclear Corporation where he did work with fuel element design and inspection of reference elements in US Nuclear Power Plants. He was also on staff with AEG Research Center for which he worked in the area of fuel element design, qualification of fabrication processes, testing programs on fuel elements and inspection of reference elements in nuclear pilot plants. Mr. Schulz holds a B.S./M.S. in mechanical and nuclear engineering and has served on the engineering faculty at Essen.

FRED SIMONEN
LABORATORY FELLOW
PACIFIC NORTHWEST NATIONAL LABORATORY
RICHLAND, WASHINGTON

Since joining the Pacific Northwest National Laboratory in 1976, and before that at the Battelle Columbus Division beginning in 1966, Dr. Simonen has worked in the areas of fracture mechanics and structural integrity. His research has addressed the safety and reliability of nuclear pressure vessels and piping as well as other industrial and aerospace structures and components.

During the 1990's Dr. Simonen was a leader on the behalf of NRC and the American Society of Mechanical Engineers in the implementation of risk-informed methods for the inspection of nuclear piping. Dr. Simonen supported NRC staff by writing all chapters on piping reliability that are now part of DG-1063 Regulatory Guide on Risk-Informed In-service Inspection of Nuclear Power Plant Piping. These chapters provided the first formal set of guidelines to industry for probabilistic structural mechanics calculations for estimating piping failure probabilities. His recommendations impacted the selection of critical piping components that are given high priority for nondestructive examinations. On behalf of NRC and ASME Research, Dr. Simonen led a national effort during 1996-97 to benchmark probabilistic fracture computer codes. The exercise concluded with PNNL performing the first ever statistically based calculations to quantify the uncertainties in calculated piping failure probabilities.

Since the early 1980's he has led several studies for the U.S. NRC on the effects of pressurized thermal shock on the failure probability of reactor pressure vessels. This work has advanced the technology of probabilistic fracture mechanics and methods for estimating the number and sizes of flaws in vessel welds. Dr. Simonen's research has corrected longstanding deficiencies in traditional methods used to estimate the number and sizes of the welding flaws that govern the structural reliability of high-energy reactor piping and vessels.

Dr. Simonen was invited during 1995 and 1998 by the International Atomic Energy Agency to meetings in Russia and Sweden to participate with a group of experts who evaluated the application of the leak-before-break concept to RBMK reactors. The Central Research Institute of the Electric Power Industry invited Dr. Simonen to Japan during 1998 to present lectures on the reliability of reactor piping and methods to quantify the benefits of in-service inspection programs.

Dr. Simonen has published over 200 papers, articles and reports in the open literature. He is a member/fellow with the American Society of Mechanical Engineers and serves on numerous ASME committees and codes and standards bodies, and has been awarded a number of prestigious awards from ASME.

Dr. Simonen holds a PhD. and Masters Degree in Engineering Mechanics from Stanford University and a B.S. in Mechanical Engineering from Michigan Technological University.

GERY WILKOWSKI
PRESIDENT
ENGINEERING MECHANICS CORPORATION OF COLUMBUS
COLUMBUS, OHIO

Dr. Wilkowski is an internationally recognized expert on the fracture behavior of piping in the nuclear as well as oil and gas industries. His areas of expertise include: full-scale pipe and pressure vessel fracture testing, nondestructive examination, J_R-curve testing, high-rate toughness testing, experimental design and instrumentation, elastic-plastic estimation scheme analysis, impact testing, ASME Section XI flaw analyses, leak-before-break analyses, and pipe system fracture behavior under seismic loading.

He was heavily involved in the development and verification of the fracture mechanics analyses for circumferential cracks in nuclear pipe for ASME Section XI. He was also a member of the following review committees:
(1) NRC Pipe Crack Task Group member that developed the NRC LBB procedure,
(2) NRC Peer Review Committee for proposed new seismic design rules for nuclear piping,
(4) NRC CRDM cracking review team member,
(5) NRC Davis-Besse clad integrity review team member,
(6) Consultant to AECB on CANDU pressure tube guillotine break phenomena, and
(7) Member of DOE's Peer Review Groups for: Savannah River plant, New Production Reactor plant, Advanced Neutron Reactor, and uranium hexafloride storage cylinders.

Dr. Wilkowski is a fellow of ASME. Currently he is a member of the following ASME Boiler and Pressure Vessel Code Section XI groups: Plant Operating Criteria Special Working Group, Flaw Evaluation Working Group, and Secretary of the Pipe Flaw Evaluation Working Group. He is the past chairman of the ASME Materials Fabrication Committee, and past chairman of the Pipe and Support Subcommittee of the ASME Operations, Applications, and Components Committee, all of which are part of the ASME Pressure Vessel and Piping Division. He was a coordinator for the 14th, 16th, and 17th Structural Mechanics in Reactor Technology (SMiRT) Conferences. He is a registered professional engineer in the State of Ohio since 1979.

Dr. Wilkowski has more than 200 technical publications, most on piping fracture. He is currently on the Editorial Board of the *International Journal of Pressure Vessels and Piping.* He is a past Associate Technical Editor of the ASME Journal of Pressure Vessel Technology, and guest editor of the Nuclear Engineering and Design journal. He was editor or co-editor of eleven ASME special technical publications. He was co-editor of four NRC Conference Proceeding Reports on leak-before-break.

Dr. Wilkowski has both a B.S. and M.S. degree in Mechanical Engineering from the University of Michigan and a PhD in Nuclear Engineering from the University of Tokyo.

APPENDIX B

MEETING MINUTES FROM GROUP PANEL MEETINGS

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MEETING MINUTES FROM GROUP PANEL MEETINGS

In this appendix the meeting minutes from the three group meetings of the expert panel are presented. First the meeting minutes from the kick-off meeting are presented, followed by the meeting minutes from the base case review meeting. Lastly, the meeting minutes from the wrap-up meeting of the elicitation panel are presented.

MEETING NOTES FROM US NRC LOCA ELICITATION KICK-OFF MEETING
DOUBLETREE HOTEL, ROCKVILLE, MD
FEBRUARY 4 – 6, 2003

Day 1 – Tuesday, February 4, 2003

Welcoming Remarks, Agenda Review, and General Information

Rob Tregoning of the USNRC began the meeting with a review of the agenda and general announcements. In addition, the individuals present were asked to introduce themselves with a short background of their experience related to the issue of LOCA frequency estimations.

Mike Mayfield of the USNRC welcomed the group and offered his perspective on the subject. Hossein Hamzehee of the USNRC also stressed the importance of the LOCA frequency determination for the continuing effort to explore risk-informed revision of 10 CFR 50.46, which govern Emergency Core Cooling System (ECCS) requirements.

The meeting attendance list was provided to all of the meeting participants.

Presentation: Importance of LOCA Distributions to 50.46

The first presentation on the agenda was made by Alan Kuritzky of the USNRC. Alan laid out the importance of LOCA frequency estimates with respect to the 50.46 revision effort. Some of the key points from his presentation and subsequent discussion are outlined below:

- The NRC staff proposed a plan for risk-informing the technical requirements of 10 CFR Part 50 (Option 3) in SECY 99-264.
- Stakeholder input was considered in the recommendation to focus revision on the ECCS requirements.
- These requirements are covered in three regulations: 10 CFR 50.46, Appendix K to 10 CFR 50.46, and General Design Criterion (GDC) 35.
- Potential changes to 10 CFR 50.46 fall in one of the following areas
  - ECCS reliability (one of the focuses of elicitation – due to the simultaneous Loss of Offsite Power [LOOP] requirement)
  - ECCS acceptance criteria
  - ECCS evaluation model
  - ECCS LOCA size definition (another focus of the elicitation)
- The elicitation results will impact changes to the ECCS reliability areas and the ECCS LOCA size definition.
- ECCS reliability is primarily impacted because of the effort to eliminate the simultaneous LOCA-LOOP requirement. This has two pieces:
  - LOCA initiation frequencies and
  - Conditional probability of LOOP given a LOCA.

The focus of the current expert elicitation effort will be to obtain robust LOCA frequencies for use in the LOCA-LOOP evaluation. Interim LOCA frequencies were developed last spring as part of an internal NRC elicitation effort and these have been used to demonstrate the technical feasibility of a change in this requirement. The results of this interim NRC effort will not be made available to this panel until after the wrap-up meeting to ensure that the panel results are independent. The objective is to have this project completed by December 2003. More detail on this topic is provided in Rob Tregoning's subsequent presentation.

For the LOCA size definition effort, a computational code is being developed to incorporate LB LOCA contributions from pipe breaks and other component failures. The results of this panel will be used to normalize the analytical code results and also provide distributions for important input variables. The targeted completion date for this technical feasibility study is July 2004. More detail on this topic is also provided subsequently.

Rob Tregoning indicated that the LOCA frequencies would also be used within the 10 CFR 50.61 risk-informed revision effort (Pressurized Thermal Shock Rule) to ensure that current calculations are acceptable.

Presentation: Current LOCA Frequencies and Failure Mechanisms

The next presentation was made by Bill Galyean of INEEL in which he reviewed Appendix J of NUREG/CR-5750. Some of the key points from his presentation and subsequent discussion include:

- There are a number of varieties of LOCA initiating events, including:
  - Traditional pipe break LOCAs
  - Stuck open PORVs and SRVs
  - Steam generator tube ruptures
  - Reactor coolant pump seal failures
  - Interfacing system LOCAs (ISLOCAs) – where primary system coolant is inadvertently introduced into the secondary side piping and a secondary pipe fails creating a leak path of primary coolant outside containment
  - Reactor vessel rupture
- While service history failure data exists for some of those categories, data for pipe break LOCAs and other similar events simply does not exist because it has never occurred.
- There is methodology for estimating the frequency of an event that has never occurred. A Bayesian update of a non-informative prior can be employed. This assumes that the mean value for the distribution is ½ of a failure over the service life. This can be result in a very conservative estimate because the assumed failure frequency in the prior is so high (pf = 0.5). If the failure rate is not constant over time, one also needs to account for time dependency and this methodology is not equipped for this.
- A primary Appendix J assumption is that you needed a leak before you can get a break. A conditional pipe break probability given a leak was based on the Beliczey-Schulz correlation.
- There was also a presentation of passive LOCA failures that can occur in non-piping systems as well as a list of possible data sources for this information.

Discussion: The elicitation panel discussed the validity of this assumption for degradation mechanisms that result in long surface flaws which are not as likely to leak prior to failure. Also, the expectation is that leaking flaws will be fixed after they are discovered during a plant walkdown or through other leak detection methods.

Discussion: Bruce Bishop indicated the need for very clear definitions of what constitutes a large, medium, small, and very small break LOCA. The concern is that the system response to a double-ended guillotine break (DEGB) where the flow rates can reach 3,250,000 lpm (860,000 gpm) (according to Westinghouse calculations) is very different from a 19,000 lpm (5,000 gpm) leak which is also often characterized as a large break LOCA. Rob Tregoning indicated that clear definitions will be developed as part of this exercise.

Discussion: Gery Wilkowski relayed information provided by Helmut Schulz that the Beliczey and Schulz correlation of conditional probability of a rupture given a leak was developed for cyclic fatigue crack growth.

Discussion: Rob Tregoning emphasized that Bill Galyean's presentation was provided to recap the last NRC-sponsored work in this area. This NUREG/CR-5750, Appendix J approach is not endorsed for the expert elicitation process; however, it represents one manner in which LOCA frequencies have been developed. Tregoning also emphasized that because substantial LOCAs have not occurred, past operating experience data needs to be augmented by information from other areas. If information was available simply from service history experience, there would be no need for the elicitation.

Discussion: The point was also raised that the panel needs to consider LOCA sources other than traditional pipe LOCAs.

Presentation: LOCA Frequency Determination Using Expert Elicitation

Rob Tregoning then made a presentation on LOCA frequency determination using expert elicitation. The objectives of this presentation are to motivate the expert elicitation effort, discuss the limitations of relying solely on past operating experience; present ongoing NRC-sponsored research in this area; define the objective of the expert elicitation; outline the approach; and discuss the structure of the kick-off meeting. Some of the specific key points from his presentation and subsequent discussion are outlined below:

- The large break (LB) LOCA design basis size will be determined by considering all relevant LOCA sources. A probabilistic fracture mechanics (PFM)-based model is under development for predicting LOCA contributions as a function of break size. This code will also account for LOCAs from non-piping sources, and will include contributions from future unknown failure mechanisms. Expert elicitation input from this panel will be used throughout program development.
- The objectives of the elicitation are to
  - Develop future small break (SB), medium break (MB), and large break (LB) loss of coolant accident (LOCA) frequency estimates extending up through the end of license renewal (approximately 35 years).
  - Develop benchmark problems and standardized inputs for conducting probabilistic fracture mechanic (PFM) simulations of LB LOCA events in important BWR and PWR systems.
- The elicitation approach will construct base cases. Quantitative LOCA estimates as a function of break size will be developed for these base cases. Then, important variables and issues will be discussed within the relative framework of the base cases.

Discussion: It was stressed that the base cases will just provide a reference point. Adjustments to the base cases will account for the impact of those issues which contribute significantly to the LOCA frequencies. These adjustments must consider their effect on current LOCA frequencies and their time dependence up through the end of license extension (≈ 35 years).

- The programmatic approach for the elicitation was presented. It consists of the following important areas:
  - Conduct the kick-off meeting
  - Develop elicitation questions
  - Allow individual study to develop answers for elicitation questions
  - Conduct the individual elicitations
  - Analyze results (facilitation team)
  - Conduct a wrap-up meeting

Discussion: It was conveyed that when the facilitation team queries each elicitation panel member, best estimate answers will be sought as well as the uncertainty in the estimates. It was also stressed that each panel member need not answer all questions, but only those that they feel that they are qualified and comfortable with answering. However, people will be encouraged to answer all questions, even in areas outside of their specific expertise. Uncertain knowledge should be reflected in the uncertainty estimates. Rob Tregoning indicated that one job of the facilitation team will be to filter out responses which are not well-founded, and exclude them from the final analysis.

- The principal components of the kick-off meeting were reviewed again. There are several major objectives which need to be accomplished during this kick-off meeting:
  - Present the elicitation objectives and define fundamental terms to ensure common understanding.
  - Undergo elicitation training to understand the process and approach.
  - Construct methodology for developing baseline LOCA estimates. Develop a classification scheme and approach for issues which could affect the baseline LOCA estimates.
  - Identify and classify issues for consideration. Discuss issues as necessary for clarification.
  - Agree on significant issues to include in the elicitation.
  - Determine the structure of the elicitation questions.

Presentation: Expert Elicitation Process

Lee Abramson of the US NRC spoke on the expert elicitation process that will be followed in this exercise. Some of the specific key points from his presentation and subsequent discussion are outlined below:

- Key word is "formal" use of expert judgment. Engineers practice informal expert judgment every day.
- It was emphasized that elicitation is a structured process and that the process requires experienced practitioners to conduct the exercise. This is not a "do it yourself" activity.

Discussion: A question was raised if the results of this elicitation or past elicitations could be used as a baseline for future efforts, in much the same way that Bayesian analysis is performed. Lee Abramson indicated that there is no natural means of updating results from prior elicitations based on recent experience or new data. However, it may be appropriate to use the results of a prior elicitation as starting point for future elicitation.

- The need for comprehensive documentation was also stressed to ensure that the process approach, issues, analysis techniques, results and uncertainties are clear. Additionally, follow-on work to refine the results requires comprehensive documentation in order to understand the basis of the initial study.

- The need for an expert panel with a broad range of expertise and experiences was expressed. Also all of the stakeholders (both utilities and regulators) must be represented.
- There are two methods of elicitation: group and individual. The problem with group sessions (versus individual sessions) is that often group dynamics lead to domination of one or two individual opinions. The results then no longer represent everyone's input.
- Elicitation team for this exercise consists of
  - Normative expert – **Lee Abramson**
  - Substantive experts – **Alan Kuritzky, Ken Jaquay, Rob Tregoning, others?**
  - Recorder – **Paul Scott**
  - Documenter – **Paul Scott (could be same as recorder)**
- Panel members need to provide rationale for answers so others can see why certain experts came up with certain answers. In that way other experts have the option of changing their answers based on feedback from the group. The panel will largely be provided this feedback at the wrap-up meeting. Panel members can revise answers to any question at any time.

Discussion: It was asked if the response will be weighted in any way to account for expertise in a given area. Lee Abramson replied that the analysis will use unweighted responses so that everyone's response is judged equally. With this size of panel, weighting should not substantially affect the final results. The elicitation will also query the panel member's uncertainty for each answer. If inordinate uncertainty exists, then the response ***may be*** downgraded. Also, the rationale provided by each expert will help determine if responses need to be weighted.

- Types of biases present in elicitation processes:
  - Motivational biases (i.e., social pressure or group pressure to make a certain decision). These need to be recognized and avoided at all costs.
  - Cognitive biases — biases can occur when people have developed an initial answer and more data becomes available which require the initial answer to be modified. Typically people underestimate the impact of the new data. This bias is referred to as anchoring. The elicitation structure will be developed in an attempt to minimize these biases. For instance, initial estimates of the total LOCA frequencies will not be asked.
  - Background biases (i.e., what an individual might see as reasonable, or would expect, based on his background.) For example, an experimentalist might see a high probability of failure of a piping based on the number of experiments he has run in which he saw a failure, but typically the test conditions were such that similar conditions in the field are highly unlikely to ever occur. This bias is natural, but it is important to get each individual to consider all variables which affect the result and break them down into meaningful pieces.
- People are more than likely to underestimate the true uncertainty, by a factor of 1/2.
- People are more likely to anchor on median value, not on the extremes.
- Goal is to make the questions as unambiguous as possible (very precise) and to focus questions on the major issues affecting the LOCA analysis.
- The uncertainty range will be queried during the elicitation by asking for the "number" such that there is 5% chance that the true response is less than this number. A separate number will be provided for the upper bound such that there is also a 5% change that the true response is higher than this number. This corresponds to the 90% coverage interval of the variable.
- Purpose of elicitation panel members is to come up with individual answers, not a consensus.

Discussion: There was quite a bit of discussion and confusion about the definition of the coverage interval. Lee Abramson said that the uncertainty range (difference between higher and lower response to a given question) should cover the true number for that variable 90% of the time. The true value should fall below the lower response 5% of the time and the true value should land above the higher response 5%

of the time. However, Lee cautioned against making the coverage interval inordinately large just to capture uncertainty. If this occurs, the coverage interval contains little useable information.

Elicitation Exercise

Each participant filled out an elicitation questionnaire dealing with age related health issues. The results from this exercise were reviewed with the meeting participants on Wednesday morning. As part of this exercise, Lee Abramson indicated that it is usually easier to determine relative rates versus absolute rates. Various absolute and relative questions were posed in order to demonstrate this concept.

Definition of Terms for Elicitation

Terms used during the elicitation must be commonly understood by the group in order to foster discussion, issue development, and subsequent elicitation. Certain key terms must be defined. Rob Tregoning indicated that, for this exercise, all definitions should be kept generic, not plant specific. The first term to be defined is loss-of-coolant accident (LOCA). Rob Tregoning presented the NUREG/CR-5750, Appendix J definition as a starting point. This report defines a LOCA as "an unisolable breach of the Reactor Coolant Primary Boundary (RCPB) requiring Emergency Core Cooling System (ECCS) initiation."

The group felt that the term "unisolable" was not appropriate because the main point is to limit the scope to Class 1 piping. Also, the merits of the phase ECCS initiation were debated because the ECCS response in some plants requires use of normally operating plant equipment. Therefore, some plants might require a large leak before implementation of standby ECCS systems. There was also a discussion on the merits of using break instead of breach, but the term breach was determine to be more generic than break. The addition of the term "sudden breach" instead of just "breach" was also neglected because of the vagueness of the word sudden.

The group agreed to a definition of a general LOCA as follows. **A LOCA is "a breech of the reactor coolant pressure boundary which results in a leak rate beyond the normal makeup capacity of the plant".**

The next definitions are required to determine the size classifications of LOCAs. Once again, Rob Tregoning presented the definitions used in NUREG/CR-5750, Appendix J as a starting point. These definitions were also used in NUREG-1150 and form the basis of plant probabilistic risk assessment (PRA) event trees. This document defined three LOCA size categories: small break (SB), medium break (MB), and large break (LB). The NUREG/CR-5750 definitions are as follows:

- **SB LOCA** - A break that does not depressurize the reactor quickly enough for the low pressure systems to automatically inject and provide sufficient core cooling to prevent core damage. However, low capability systems (i.e., 380 to 5,700 lpm [100 to 1,500 gpm]) are sufficient to make up the inventory completion. For a BWR, this translates to a pipe in the primary system boundary with a break size less than 370 mm^2 (0.004 ft^2), or a 25 mm (1 inch) equivalent inside pipe diameter, for liquid, and less than 4,600 mm^2 (0.05 ft^2), or an approximately 100 mm (4 inch) inside diameter pipe equivalent, for steam. For a PWR, this equates to a pipe break in the primary system boundary with an inside diameter between 13 to 50 mm (½ to 2 inches).
- **MB LOCA** – A break that does not depressurize the reactor quickly enough for the low pressure systems to automatically inject and provide sufficient core cooling to prevent core damage. However, the loss from the break is such that high capability systems (i.e., 5,700 to 19,000 lpm [1,500 to 5,000 gpm]) are needed to makeup the inventory depletion. For a BWR, this translates to a pipe in the primary system boundary with a break size between 370 to 9,300 mm^2 (0.004 to

0.1 ft^2), or an approximately 25 to 125 mm (1 to 5 inches) inside diameter pipe equivalent, for liquid, and between 4,600 to 9,300 mm^2 (0.05 to 0.1 ft^2), or an approximately 100 to 125 mm (4 to 5 inches) inside pipe diameter equivalent, for steam. For a PWR, this equates to a pipe break in the primary system boundary with an inside diameter between 50 to 150 mm (2 to 6 inches).

- **LB LOCA** – A break that depressurizes the reactor to the point where the low pressure system injection automatically provides sufficient core cooling to prevent core damage. For a BWR, this translates to a pipe in the primary system boundary with a break size greater than 9,300 mm^2 (0.1 ft^2), or an approximately 125 mm (5 inch) inside diameter pipe equivalent, for liquid and steam. For a PWR, this equates to a pipe break in the primary system boundary with an inside diameter greater than 150 mm (6 inches).

The elicitation panel questioned the basis of the equivalent pipe diameter relationships to break size provided in the NUREG/CR-5750 Appendix J. Bill Galyean thought that they could be traced back to NUREG 1150 and possibly WASH-1400. It was quickly determined that "break" should be replaced by "breech" everywhere for consistency with the general LOCA definition. Also, the group decided that the formal definitions should be based on leak rate, and not equivalent break area or size.

At this point, the need for additional LOCA size classification was revisited. This request was promulgated by Bruce Bishop based on discussions with the Westinghouse Owner's Group (WOG). The System response and mitigation procedures for a 19,000 lpm (5,000 gpm) LOCA (lower limit LB LOCA within NUREG/CR-5750, Appendix J) and a DEGB of the largest class 1 pipe (flow rate up to 3,250,000 lpm (860,000 gpm) according to WOG) are significantly different. Because, the elicitation results will used in existing PRAs, it was also stressed that the original leak rate classifications in NUREG/CR-5750, Appendix J should also be maintained. While the group also agreed that the leak rate threshold should ideally be based on the equipment needed to mitigate a specific event, this information is highly plant specific and could not be approximated generically.

For the reasons stated in the above paragraph, the group decided to keep the NUREG/CR-5750, Appendix J leak thresholds of 380 lpm (100 gpm), 5,700 lpm (1,500 gpm), and 19,000 lpm (5,000 gpm), but to add several leak rate categories above 19,000 lpm (5,000 gpm). The highest category was set at 1,900,000 lpm (500,000 gpm) to capture the DEGB events of the largest primary system pipes. Additional ranges of 95,000 lpm (25,000 gpm) and 380,000 lpm (100,000 gpm) were chosen to span the range from 19,000 lpm (5,000 gpm) to 1,900,000 lpm (500,000 gpm) in roughly equivalent magnifications. These leak rate categories were also chosen because they tend to group DEGBs by primary system functionality.

The LOCA size classification thresholds adopted by the group are summarized in Table B.1.1[1]. A category 1 LOCA is defined as "a breach of the reactor coolant pressure boundary which results in a leak rate which is greater than 380 lpm (100 gpm). Similarly, a category 6 LOCA is a breach of the RCPB which results in a leak rate which is greater than 1,900,000 lpm (500,000 gpm). It should be stressed that category 1 LOCAs include contributions from all categories. The group preferred the threshold classification of LOCA sizes instead of partitioning the sizes into ranges as in NUREG/CR-5750, Appendix J. Care will be needed during the elicitation to ensure that these definitions are understood.

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[1] The nomenclature for the table and figure numbers is such that the letter B refers to Appendix B, the first number (1 or 2) refers to a figure associated with either the first or second panel meeting, and the second number refers to the numerical sequence of that particular table or figure in the text for the applicable meeting, i.e., either first or second.

Table B.1.1 LOCA Size Classification Thresholds

Category	Leak Rate Threshold (gpm)
1	> 100
2	> 1,500
3	> 5,000
4	> 25,000
5	> 100,000
6	> 500,000

It was determined by the group that these leak rates should be roughly correlated to breach area, and converted into an equivalent pipe diameter so that the upper bound leak rates for various piping systems could be determined. There was some concern about the feasibility of developing generic estimates. It was suggested that equivalent pipe sizes could be based on 1.47 lpm/mm^2 (250 gpm/in^2) for liquid PWR lines and 1.03 lpm/mm^2 (175 gpm/in^2) for liquid BWR lines. However, these estimates did not agree with the Westinghouse equivalent pipe diameter estimates.

Presentation: SKI-PIPE Database: Background - Structure - Status - Applications (1994 - 2002)

This presentation by Bengt Lydell discussed the SKI-PIPE database evolution and background; the database structure and content; current database status; and LOCA frequency estimate conducted with the data base. Some of the specific key points from his presentation and subsequent discussion are outlined below:

- Background: The database was motivated to create a tool that would serve both PRA and the PFM/material science practitioners. It's structured to provide information to completely define the piping systems attributes (design characteristics) and the influence functions (operating history) which govern system failure probability. By thoroughly assessing these features it is possible to determine plant specific estimates of piping system reliability.
- Structure and Content: The database covers pipe failures in commercial nuclear power plants from 1970 to the present.
- It should be stressed that SKI-PIPE only includes failures in piping systems, external to the reactor pressure vessel. Non-piping system failures are not included. Also, SKI-PIPE contains only passive piping failures of metallic piping.
- A pipe failure is defined in the database as any degradation that results in piping repair or replacement.
    - Each record in the database is indexed. References to the original data source (e.g., LER report) and supporting information are provided. All the supporting documented is stored electronically.
    - The database is organized by reliability attributes (i.e. design features such as material, dimensions) and influence factors (i.e. unique service conditions, including degradation susceptibility).
    - When the original record is incomplete (such as an LER), a best effort is made to fill in database gaps by directly contacting the plant operators.
    - It is noted in the database when each record consists of multiple flaws at a single component location. However, subsequent data entries are typically associated with only the largest flaw at that location.

    - The database includes both surface penetrating flaws and non surface penetrating flaws (i.e., embedded flaws).
- Current Database Status:
    - The database is continually being updated.
    - The current OECD-sponsored OPDE project has participants from 12 nations. The first year of the three year effort is concerned with adding and validating database entries for each of the member countries from 1998 through 2001.
    - Raw data is currently obtained from over 40 different sources
- Applications: Two relevant studies are the determination of LOCA frequencies for the Barsebäck-1 plant and examination of intergranular stress corrosion cracking (IGSCC) in Russian graphite moderated reactors (RBMK).
    - The Barsebäck-1 study employed plant-specific attribute and influence functions which were comprehensively developed for all "known and credible" degradation mechanisms.
    - The Beliczey and Schulz conditional rupture probability was not used in the Barsebäck-1 analysis. Instead a Bayesian update of a Jeffrey's modified non-informative prior was employed.
    - The database results have been compared with PFM predictions for welds in certain systems with some success.
    - The RBMK studied indicated that the experience today with IGSCC in Russia is similar to US BWR IGSCC cracking experience in the late 70's to early 80's, before wide-spread mitigation was adopted

Discussion: The panel asked if they could get copies of the SKI-PIPE database. Karen Gott of SKI indicated that it is possible to distribute a non-proprietary version of the database. This non-proprietary version contains piping failures thru 1998.

Baseline LOCA Determination I

Discussion: Rob Tregoning commented that the panel needed to define baseline LOCA frequencies in order to benchmark relative responses during the elicitation. He also mentioned that the SKI-PIPE database could be used to develop baseline frequencies if the group could develop well-defined "base case(s)". The base case(s) will represent a set of conditions and physical phenomena. In theory, the absolute LOCA frequencies associated with each base case are not important for the elicitation session because all elicitation responses will be judged relative to the base case conditions. The absolute frequencies are only required to reconstruct the final results. However, the panel members decided that their elicitation responses might change depending on the exact LOCA frequencies associated with the base case conditions. That is, if a base case frequency was 10^{-8}/year, the elicitation responses might be quite different than if the frequency was 10^{-2}/year. **The group therefore agreed that they will define rigorous conditions for each base case and also associate absolute LOCA frequencies with these conditions.**

<u>Day 2 – Wednesday, February 5, 2003</u>

Elicitation Exercise Review

Lee Abramson reviewed the elicitation questionnaire results from Tuesday afternoon's session. Overall, the results were good and consistent with expectations. The group tended to perform better on those questions that asked for the ratio of diseases between men in different age ranges (questions 3 and 4 in exercise). The mid value tended to reasonably close to the actual 2000 census values for these questions.

Additionally, the true value was contained with the 50% interquartile region (75% - 25% percentiles) 10 out of 12 times, or 83%, which is quite good.

The average coverage interval for these questions was 71%. The coverage interval should theoretically be 90%, so that group underestimated the uncertainty. Lee Abramson indicated that typically group uncertainty is about ½ of the true value. In other words, people tend to be more confident in their responses than they should be.

The results were not quite as good when the group was asked to provide absolute disease rates for an age category (Question 2). The true value was inside the 50% interquartile range 4/6 times, or 67%. Also, the coverage interval only captured the true value 61% of the time, 10% less than for the relative questions. This performance demonstrates the supposition relative differences between conditions tend to be more accurate than absolute measures for a given condition. This will be a guiding principal in developing the elicitation framework.

LOCA Issue Development

LOCA Issue development required the group to brainstorm important LOCA issues. The group first defined a structure for categorizing issues in the form of a flowchart (Figure B.1.1). It was stressed that the LOCA frequencies in this exercise will consider only passive system failure. Active system failure will not be considered for the following reasons:

1. The panel has no specific expertise in these types of failures.
2. Failure of these components is not as rare and there is adequate data to assess their contribution to the LOCA frequencies.
3. Active components are subject to ongoing maintenance which should diminish the likelihood of future failure rate increases.

However, LOCA frequency contributions from active components will be combined with the passive component contributions to develop final LOCA estimates which can be supplied as PRA input. The estimation of these contributions will occur separately, but will be summarized for the panel at the wrap-up meeting.

The group divided the passive system LOCA sources (Figure B.1.1) into two classes: piping and non-piping. Non-piping contributions include reactor pressure vessels, steam generators, bolting flange failures, valves, pumps, etc. The distinction between piping and non-piping categorization is useful because piping has unique issues.

Piping LOCA Contributions

The group first defined piping to include vessel penetrations (e.g. control rod drive mechanism CRDM housings, instrumentation lines), piping, and safe ends. The boundary between piping and non-piping components (e.g. vessels) was defined as the nozzle (or component) side of the safe-end/piping to nozzle weld. The group then decided that piping should be categorized by the specific plant system. The piping system is important because it defines the functionality and operating history, or influence factors. The plant system is also often associated with specific piping designs and materials, or attribute functions. The relationship between the piping attribute and influence characteristics will determine its failure propensity.

For a given plant system, the variables which affect the LOCA probability fall into one of the following five categories: geometry, materials, loading history, degradation mechanisms, and mitigation or maintenance procedures. The group decided to list all the possible contributors for each variable category and then link the dependencies with a given plant system. Obviously there is a synergistic effect among these variables. The piping system requirements result in geometrical and material selection constraints. The geometrical and material choices mesh with the system functionality and operating history to determine component loading history. This specific combination dictates the degradation mechanisms that emerge. Mitigation and maintenance procedures are developed to counteract these mechanisms. The effectiveness of these strategies, however, is a function of all the other variables discussed.

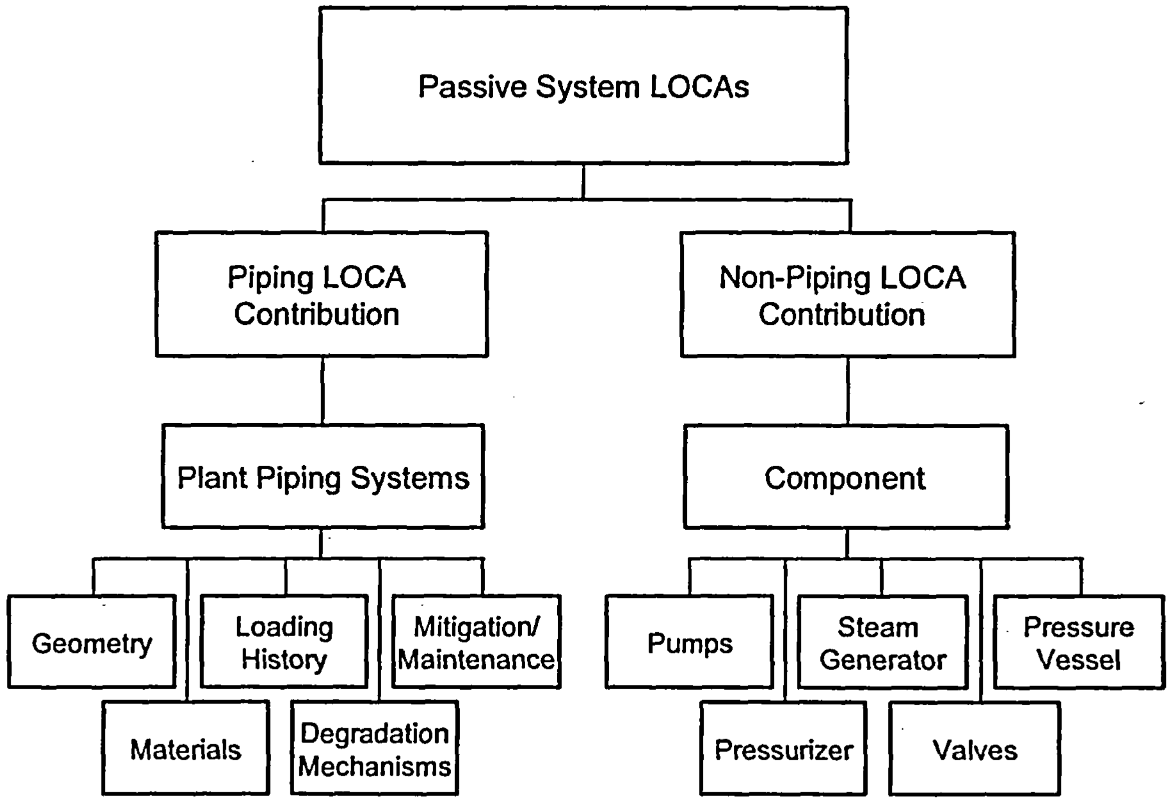

Figure B.1.1 Passive LOCA Contributions

Material Variables: It was quickly determined that the materials category should also include fabrication procedures as well. Material variables are important because they give rise to specific degradation mechanisms. Fabrication variables can lead to variations in defect formation and residual stress which can result in certain locations having a greater propensity for degradation. The important variables which affect the LOCA propensity are summarized in Table B.1.2. Table B.1.2 indicates whether the material is found in the base metal portion of the piping system or the welds. Issues that are strictly related to fabrication are separated in the table.

The circumferential versus axial welded pipe issue was raised to differentiate between seamless and non-seamless pipe. While the bulk of current piping is seamless, there is some remaining seam welded piping in service. The concern is that seam-welded piping is more susceptible to degradation and leaking than is seamless. The shop versus field welded issue was raised due to possible weld quality differences between welds made in controlled conditions during piping fabrication and those made on site during piping system assembly. Finally, it was noted that defects, residual stress irregularities, and poor material properties can be associated with repair welding. Experience has shown that field cracking and leakage is often associated with repair welds.

Table B.1.2 Piping Material and Fabrication Variables

Materials	Base metal	Weld/HAZ
304, 304L, and 304NG stainless steel	X	X
316, 316L, and 316NG stainless steel	X	X
Carbon steel	X	X
Stainless steel clad carbon	X	X
Alloy 600	X	
Alloy 82/182		X
Cast stainless steel	X	
Stainless steel bimetallic		X
Fabrication Issues		
Circumferential vs. axial welded pipe		X
Shop vs. field welds		X
Repair welds		X

Geometric Variables: Table B.1.3 lists the geometric variables which influence the LOCA frequency distributions. These variables affect piping stress, system compliance, the propensity for a given degradation mechanism, and the likelihood of leaking versus catastrophic rupture. Many of the variables are obvious and include general system information such as piping diameter and thickness (NPS and schedule), the number of welds and their location, the types and numbers of specific piping components, and the layout of supports and snubbers. The system configuration is related to the layout, but also specifically considers where active components such as pumps, valves, and flow offices are located. The variable "connections" in this table are meant to distinguish between welded connections which are more typical and flanged connections which can segregate piping from active components.

Table B.1.3 Geometric Piping Variables

NPS and Schedule (diameter and thickness)
Component Type (Elbow, tee, fittings, straight pipe, reducers, sockets, safe ends, end caps, etc.
Field fabricated vs. drawn
Crevice weld (thermal sleeves)
Number of welds & location
Configuration (active components, flow orifices, etc.)
Layout including locations and types of supports and snubbers
Connections (bolted flanges)

The final two variables in this table are a combination of geometric and material/fabrication issues: the existence of crevice welds (e.g. at thermal sleeves), and the difference between drawn piping and piping which is field-fabricated. Drawn piping is cold worked to size and is usually limited to smaller diameter piping (< xx" NPS). Field-fabricated piping may be hot forged or cold work to some extent, but the final size is often achieved by machining. This is more typical for larger diameter pipes (> yy"). Crevice or filet welds are for sleeves and other piping system attachments. Partial penetration welding can result in a greater propensity for flaws and higher residual stresses and constraint than through-wall welding.

Material Degradation Mechanisms: The next variable group describes material degradation mechanisms. As mentioned earlier, the degradation mechanisms are often associated with the piping system material. Each material is susceptible to each of the mechanisms listed, although the degree of susceptibility will greatly vary. The system loading history can also favor certain mechanisms. Table B.1.4 summarizes the mechanisms developed during the group brainstorming session. The mechanisms are segregated by the primary mechanism type. Additional sub-categories are used to identify either specific degradation mechanisms under the appropriate main category or features associated with the main category.

Table B.1.4 Material Degradation Mechanisms

Main Category	Sub-Category 1	Sub-Category 2	Sub-Category 3	Sub-Category 4
Low Cycle Thermal Fatigue	Crack Initiation	Crack Growth		
High Cycle Mechanical Fatigue	Vibration	Pressure	Temperature	
Stress Corrosion Cracking	IGSCC	TGSCC	ECSCC	PWSCC
Localized Corrosion	Pitting	Crevice Corrosion		
General Corrosion	Boric Acid (ID or OD)			
Fretting Wear				
Material Aging	Thermal	Dynamic	Radiation	Creep
Fabrication Defects and Repair				
Hydrogen Embrittlement				
Flow Sensitive	Erosion/ Cavitation	FAC		
Unanticipated (New) Mechanisms				

Fatigue degradation was separated into low cycle fatigue which is primarily driven by thermal loading fluctuations due to plant heat-up and cool-down cycles and high cycle mechanical fatigue which could result from general loading fluctuation on the piping. This loading fluctuation could be induced by vibration, pressure or temperature fluctuation (e.g. striping). High and low cyclic loading are often differentiated with respect to the loading magnitude relative to yield strength or number of cycles. For this elicitation, a rule of thumb differentiation of 1,000 cycles is sufficient to differentiate between low-cycle and high-cycle events. Both crack initiation and crack growth portions of life are important contributors to fatigue life, although crack initiation occupies a greater percentage of life in high cycle fatigue.

Stress corrosion cracking (SCC) is listed as a main category and it includes intergranular SCC (IGSCC) which was prevalent in BWRs in the late 70's, transgranular SCC (TGSCC) which affects casting components, primary water SCC (PWSCC) which has more recently surfaced in PWRs, and external chloride SCC (ECSCC). The localized corrosion category includes both general pitting and crevice corrosion which is likely in tight, stagnate areas. While general corrosion is listed as its own category, boric acid corrosion is the principal contributor. Both internal (ID) and external (OD) boric acid corrosion are included in this sub-category. External corrosion can result from leaking fluid from another component in the plant which then impacts the pipe. Fretting wear describes material erosion usually resulting from external contact with other components. This wear usually occurs with vibration loading but should be distinguished from high cycle fatigue which results in crack formation.

Material aging relates to changes in the intrinsic quasi-static material properties at the time when the component is placed into service. These properties include constitutive properties (stress/strain behavior), cyclic, crack growth resistance (da/dN versus ΔK), and resistance to crack initiation and tearing under monotonic loading (J-R behavior). Aging can occur due to thermal or radiation embrittlement; long term, or low temperature creep. Dynamic strain aging refers to the stress/strain and J-R curve resistance changes which result from dynamically applied loading. It is most evident in the ductile to brittle transition temperature shift of ferritic steels. Hydrogen embrittlement is somewhat related to the other material aging mechanisms in that strength and toughness can be affected over time. The distinction is that hydrogen embrittlement only becomes prevalent after a crack has formed. The other aging degradations do not require a preexisting defect although the system impact is certainly enhanced in their presence.

The flow sensitive category is used to capture those mechanisms which are sensitive to the flow characteristics of the piping medium. The erosion/cavitation sub-category refers to material erosion due to cavitation which occurs when vapor bubbles collapse. This is distinct from flow accelerated corrosion (FAC) where downstream turbulence results in erosion/corrosion that is not accompanied by low pressure boiling. The category of unanticipated or new mechanisms covers those mechanisms which could surface in the future which are either unknown at the present time or not deemed to be important at this time. This category is purposely vague to capture the panel's general uncertainty of the completeness of our understanding of future piping degradation mechanisms. For instance, just a few years ago, PWSCC would not likely have been considered to be an important degradation mechanism. Now however, it is of primary concern.

Fabrication defects and the repair of those defects (or lack thereof) are distinct from repair welding mentioned earlier in the material/fabrication issue table. This issue covers the likelihood of repair of fabrication defects and the possibility that these fabrication defects could lead to failure due to one of the other mechanisms listed in Table B.1.4. The repair component of this issue only considers the possibility that these defects are repaired (or not), not any new defects generated by the repair process. Defects generated by repair welds have been captured in the geometric variability section (Table B.1.2).

Loading History Variables: The next variable which contributes to the piping LOCA frequency estimates is the system loading history. The term loading history considers both the magnitude and frequency of the loading applied to the piping system over its service history. The different types of applied loading are summarized in Table B.1.4. Again, as in Table B.1.3, the loading variables are divided into main and sub categories.

The thermal loading category considers loading from differential expansion between dissimilar piping materials. This loading is potentially exacerbated at the ends of the piping if connected to a rigid component (e.g. steam generator or RPV). This loading is categorized as restraint-free expansion in Table B.1.4. Radial thermal gradients are induced in piping due to the temperature difference between the pipe ID and OD. Insulation can diminish this gradient. Thermal stratification can occur under low flow rate conditions when hotter liquid flows on top of cooler water. Boundary layer fluctuations in this interface can induce thermal cyclic loading which is referred to as thermal striping.

Table B.1.5 Loading History Variables

Main Category	Sub-Category 1	Sub-Category 2	Sub-Category 3	Sub-Category 4	Sub-Category 5
Thermal	Differential Expansion	Restraint Free Expansion	Radial Gradient	Stratification	Striping
Water Hammer	Steam Hammer				
Seismic	Inertial	Displacement			
Pressure	Normal	Transients			
Residual Stress	Design	Repair welds	Fabrication	Mitigation-Induced	
Dead Weight Loading					
SRV Loading					
Overload (Ext. and Int.)	Pipe Whip	Jet Impingement	Deflagration		
Support	Snubber malfunction	Hanger Misadjust.			
Vibration	Mechanical	Cavitation			

Water hammer, dead weight, and safety relief valve (SRV) loading are considered as separate main categories with no corresponding sub-categories. Water hammer is distinguished from other pressure transients because of its potential severity. Safety relief valve loading describes the pressure transient which occurs when SRVs are opened or closed. Dead weight loading is contributed by the weight of the pipe and any unsupported attachments. The pressure loading category includes normal operating pressure and any pressure transients other than those specifically listed in Table B.1.5 (e.g. water hammer, SRV loading, internal overloads, cavitation).

Residual stress is a prominent loading category and includes contributions from locally-induced welding process stresses related to the as-designed weld (Design sub-category in Table B.1.5) and additional contributions due to weld repair. The Design sub-category also encompasses contributions from the pipe system compliance on the weld restraint during component assembly. System compliance will obviously influence the residual stress distribution which if formed at a particular weld joint. Fabrication residual stresses include cold-springing needed to align piping during plant construction. This residual stress contribution may not be apparent from the system design which is why it is distinguished from the Design sub-category. Finally, a unique sub-category is entitled "Mitigation-Induced" to account for residual stresses which may be applied during plant operation to mitigate certain failure mechanisms. These stresses are induced by processes like weld overlay repairs (used during IGSCC mitigation) and mechanical stress improvement (applied for VC Summer). These stresses are certainly associated with the weld joint being treated, but also affect the residual stress distribution in surrounding welds and piping.

Overloads can result from external failures in other plant systems and internal accidents. These accidents can result in loading which is potentially in excess of the structural design limits of the piping. External overloads include those induced by pipe whip and jet impingement. Both of these categories require failure outside of the reactor pressure boundary as a precursor event. The loading on the reactor pressure boundary piping occurs either by pipe whip of the failed components or through water or steam jet impingement caused by the breech in the other system. A special type of internal overload sub-category is Deflagration. This describes the loading due to hydrogen combustion which occurred at the Hamaoka and Brunsbuettal plants. This category would cover not only direct failure of pressure boundary piping, but also precursor failure of secondary piping that leads to conditional failure (from shrapnel or jet impingement) of pressure boundary piping.

Another loading history category is support structure loading. This includes loading due to a malfunctioning snubber or misadjusted hanger that leads to piping system loading that is beyond the intended design limits. Vibration loading includes classical displacement loading due to nearby active component vibration (Mechanical sub-category) and loading due to cavitation which may exist within the piping system.

The final loading category is seismic which will be treated uniquely from the other loading variables listed. The seismic loading category includes both inertial and displacement loading components. Seismic loading is treated separately within PRA models and the LOCA contribution from seismic loading is also calculated separately. However, many analysts frequently calculate conditional failure probabilities due to seismic loading and this is often the principal transient of interest in most piping systems. The panel therefore seemed generally comfortable with considering the possible effects of seismic loading if the loading magnitude was specified. For these reasons, it was decided to query the panel about seismic effects separate from any other loading history contributions to LOCA. The seismic contributions can then be segregated from the final results and used to examine the effect of conditional seismic events on the LOCA frequency distributions.

Mitigation and Maintenance Issues: The final piping variable which was discussed separately is in the area of mitigation and maintenance. This is an important topic area because these procedures have been developed to ensure piping system integrity and prevent piping rupture. The effectiveness of any particular procedure is often a function of the degradation mechanism, although some issues developed are not specific to any mechanism. These procedures can sometimes result in unintended consequences which actually exacerbate the piping failure likelihood. It should be stressed that the procedures and issues in this table are not just concerned with current practice, but also future application and possible improvements. Each panel member must express his or her expectations about future LOCA performance up to the end of the license extension period.

The first four mitigation procedures (Table B.1.6) are related to piping system inspection and maintenance. In-service inspection (ISI) and risk-informed ISI (RI-ISI) considers both current application of these programs and future industry trends. Currently, more US plants are adopting RI-ISI. The effectiveness of these techniques to find and determine the extent of degradation is considered in this category. Often a technique's effectiveness is quantified by the probability of detection (POD) for a certain degradation mechanism. Leak detection considers the broad array of leak detection methods (including plant walkdowns) and their effectiveness in uncovering piping degradation. Online monitoring considers the effect that current system performance indicators (pressure, temperature, etc.) may have on preventing failures as well as future systems that could be utilized to monitor degradation in real-time.

Planned maintenance accounts for programs which monitor degradation and then replace piping segments once the degradation exceeds allowable limits. This is a popular approach for dealing with FAC in carbon steel piping. Planned maintenance also considers any component cleaning or preparation for inspections and the effect on the failure likelihood. Planned maintenance can be either beneficial or detrimental. For instance, maintenance requires closed systems to be opened which could introduce air if gas blanketing is not sufficient. Maintenance can then lead to more future problems than if the maintenance had not occurred.

Table B.1.6 Mitigation and Maintenance Variables

Mitigation/Maintenance Procedures & Issues	Degradation Mechanism Specific	Plant Specific Variable
ISI/RI-ISI	X	
Leak Detection (Plant Walkdown)	X	
Online Monitoring	X	
Planned Maintenance	X	
Water Chemistry	X	
Decontamination	X	
Internal Linings and Coatings	X	
Weld Overlay	X	
IHSI/MSI	X	
Pipe Replacement (New Materials, New Design & Layout)	X	
Improved Weld Techniques/Materials	X	
Improved Inspection Techniques	X	
Socket Weld Replacement	X	
Plant Operating Conditions	X	
Stratification Mitigation	X	
Utility Safety Culture		X
Regulatory Safety Culture		

The next group of procedures is concerned with changing either the piping medium environment or the metal/medium interface in order to impede degradation. Water chemistry is concerned with additions or changes in the basic water chemistry in order to reduce the degradation rate of a certain mechanism. For instance, hydrogenated and noble metal additions to BWR water have proven effective in impeding the rate of IGSCC. This category also considers fluctuations in water chemistry over the plant's operating cycle and the affect that this may have on failure rates. Decontamination is related to water chemistry and considers the removal of impurities in the water supply and the possible impact that this could have on the degradation rate of certain mechanisms.

The application of internal linings and coatings is used to segregate a susceptible piping material from the environment using a coating or overlay of a more resistant material. The coating or lining performs the same role as stainless steel cladding does in protecting carbon steel piping. Weld overlays, induction heating stress improvement (IHSI), and mechanical stress improvement (MSI) all attempt to change or relieve weld joint residual stress in order to impede crack growth. Normally, they attempt to create compressive residual stresses at the inside surface, or throughout the entire, piping segment. However, as mentioned earlier, they can also affect the residual stress in other sections of the piping.

The next group of mitigation and maintenance procedures is concerned with anticipated future improvements in materials, repair techniques, and inspection methods that could reduce the likelihood of future LOCAs. These improvements are captured by the categories for Improved Weld Techniques/Materials and Improved Inspection Techniques. The Pipe Replacement category considers not just the removal of possibly degraded piping, but also the replacement with new materials that are less susceptible to known, important degradation mechanisms for a certain system. This could also be coupled with new design and layout configurations to reduce residual stresses and improve accessibility for inspections. Socket weld replacement is a specific piping replacement program that is currently being considered. The effectiveness and scope of its implementation still is uncertain which is the reason that this has been included as a separate category.

The next two mitigation procedures are related to current and future plant operating performance. Thermal stratification mitigation is a specific technique employed by some plant operators in order to improve thermal mixing and reduce stratification and striping stresses which can occur in the surge line and other piping systems. General plant operating performance is a category that captures all other similarly related issues. This would include issues such as the effect of possible power upgrades, possible future changes in the heat-up and cool-down cycle, the possibility of increased time periods between outages, etc.

The final two issues (Table B.1.6) are related to utility and regulatory safety culture in general. There are many specific issues that the group lumped into these broad categories. One such issue within the utility safety culture is human error. Human error was defined by the expert panel as the likelihood that incorrect action is taken during mitigation and maintenance. This includes improper application of techniques and procedures, misinterpretation of obvious indications (beyond the POD included in ISI), and omission of a prescribed procedure. Other issues within the broad category of utility safety culture include the adoption and implementation of risk-informed management strategy which requires a detailed understanding of real-time plant risk and the objective to embrace changes that reduce overall plant risk. The impact of economic considerations is important in terms of choosing which mitigation strategies to pursue. All decisions weigh plant risk with economic considerations to hopefully arrive at the optimal mitigation strategy. However, this mitigation strategy might not lead to the lowest possible plant risk. Flow accelerated corrosion monitoring programs illustrate this concept. While the absolute lowest plant risk could be achieved for a system by replacing the pipe with FAC-resistant materials, many plants have chosen to monitor the degradation and replace only when the failure risk becomes unacceptable.

Also part of the general safety culture are the lessons-learned from past problems. This experience may decrease the response time for mitigating future problems. For instance, the industry experience with mitigating IGSCC in the early 1980's may provide some useful strategies for PWSCC issues that are currently surfacing. Response time is generally an important mitigation concept. When degradation mechanisms are identified, the failure likelihood due to these mechanisms may continue to increase with time until effective mitigation strategies are employed which reduce their propensity. Obviously, shorter response times are preferred. The industry required roughly three to four years to fully implement IGSCC cracking mitigation strategies after the issue was fully identified. A final related issue is technology transfer which is the training of and knowledge transfer to the next generation of plant operators and engineers. As the workforce continues to age and is replaced by less experienced workers, it is possible that plant risk may be affected.

The regulatory safety culture also encompasses many of the same issues discussed under utility safety culture. Certainly lessons-learned, regulatory response time, and technology transfer equally apply to the regulatory culture. The regulatory environment is also affected by the agency's interaction with the public and the changing public perception of risk. Management philosophy and the adoption of risk-informed regulations may also influence the regulatory safety culture.

The group next determined if the effectiveness of specific mitigation or maintenance procedures varied as a function of degradation mechanisms and materials being evaluated. This dependency is reflected in the second column of Table B.1.6. Plant and regulatory safety culture were considered to be general issues which do not vary significantly with the degradation mechanism. However, the utility safety culture was considered by the panel to vary from plant to plant, as indicated in Table B.1.6. Regulatory safety culture was not determined to be a plant-specific function.

BWR Piping Systems: The variables just discussed (geometry, materials, loading, degradation mechanism, and mitigation) are important for determining the overall LOCA frequencies. However, these variables are linked to each specific LOCA-sensitive piping system (Figure B.1.1). The next group task was to identify LOCA-sensitive piping and assign only pertinent variables to each system. This task spanned both the Wednesday (2/5) and Thursday (2/6) meeting days, but it is summarized here for continuity. It should also be noted that Sam Ranganath provided most of the initial input for BWRs.

The BWR piping systems that can result in a LOCA were first identified (Table B.1.7). These include the recirculation (RECIRC), feed water, steam line, high pressure (HPCS), and low pressure core spray (LPCS), residual heat removal (RHR), reactor water cleanup (RWCU), control rod drive piping (CRD), standby liquid control (SLC), instrument lines in both the reactor and in other piping systems (INST), drain lines, head spray lines, steam relief valve (SRV) lines, and the reactor core isolation cooling (RCIC) system. It should be noted that while drain lines are associated with each system, they were segregated into a separate category due to their common functionality. The materials commonly used for the piping within each system are identified (column 2 of Table B.1.7). Similarly, the safe end material (column 4) and the weld material (column 5) are also indicated. The intent of this identification was to be comprehensive and also indicate the most prevalent materials wherever possible. Table abbreviations are provided at the end of the table.

Some additional clarification is required for certain entries in this table. In the recirculation piping system, the safe end materials (stainless steel or Alloy 600) are furnace-sensitized during manufacture. The feedwater safe end is manufactured either by interference fit, butt welded, or by a triple sleeve weld overlay. The HPCS and LPCS contain both creviced and non-creviced welds between the piping and safe end. Also, the bulk of this system's piping material is carbon steel. The CRD system consists of a crevice Alloy 82/182 weld to the reactor pressure vessel head while the stub tube ("safe end" in this system) is stainless steel and alloy 600 which is welded and furnace sensitized. It should be noted that no stainless steel clad carbon steel, cast stainless steel, or bimetallic stainless steel welds were indicated in any of these systems, although they are listed in Table B.1.2.

The next variable listed in Table B.1.7 is the nominal piping size present in the system (column 3 in Table B.1.7). This is the only geometric variable (Figure B.1.1 and Table B.1.3) indicated in Table B.1.7. It was not possible to do an exhaustive listing of the possible geometric variables (as with materials) due to the complexity and plant-specific nature of variables related to layout, configuration, weld location, and component type. Each expert panel member must individually determine variability and influence of these parameters. The rationale for listing the piping size is only to provide the panel with an indication of the piping size for maximum leak rate assessment. Common piping sizes for a system are separated by commas in Table B.1.7. Size ranges are separated by a dash and the maximum piping size is given as < 4 inch for many of the smaller systems. It should be noted that the feedwater system typically consists of 10 or 12 inch diameter pipes. However, a range from 12 to 24 inch is also possible.

Significant degradation mechanisms that could be associated with piping materials in each system are included in the sixth column of Table B.1.7. Unanticipated mechanisms (UA) and fabrication defect and repair (FDR) issues are present in every system. Stress corrosion cracking (SCC) is listed for all stainless and carbon steel materials, while global corrosion (GC) is associated solely with carbon steel piping. Localized corrosion (LC) is included for carbon steel and stainless steel piping for all systems, except SLC, CRD, and instrumentation lines. Material aging (MA) was considered for the higher temperature lines that see constant use for both stainless and carbon steels. Flow sensitive (FS) degradation is present in all carbon steel piping systems with constant use. Mechanical fatigue was judged to be significant in all of the smaller piping systems (< 4 inch diameter). However, it was also considered important in the SRV and feedwater systems. Thermal fatigue (TF) was judged to be important in the feedwater, RHR, RWCU, HPCS/LPCS, and head spray piping.

Table B.1.7 BWR LOCA-Sensitive Piping Systems

System	Piping Matls.	Piping Size (in)	Safe End Matls.	Welds	Sig. Degrad. Mechs.	Sig. Loads.	Mitigation /Maint.
RECIRC	304 SS, 316 SS, 347 SS	4, 10, 12, 20, 22, 28	304 SS, 316 SS, A600*	SS, NB	UA, FDR, SCC, LC, MA	RS, P, S, T, DW, SUP, SRV, O	ISI w TSL, REM
Feed Water	CS	10, 12 (typ), 12 - 24	304 SS, 316 SS*	CS, NB	UA, FDR, MF, TF, FS, LC, GC, MA	T, TFL, WH,P, S, SRV, RS, DW, O	ISI w TSL, REM
Steam Line	CS – SW	18, 24, 28	CS	CS	UA, FDR, FS, GC, LC, MA	WH, P, S, T, RS, DW, SRV, O	ISI w TSL, REM
HPCS, LPCI	CS (bulk), 304 SS, 316 SS	10, 12	304 SS, 316 SS, A600*	CS, SS, NB	UA, FDR, SCC, TF, LC, GC, MA	RS, T, P, S, DW, TS, WH, SUP, SRV, O	ISI w TSL, REM
RHR	CS, 304 SS, 316 SS	8 - 24	CS, 304 SS, 316 SS	CS, SS, NB	UA, FDR, SCC, TF, FS, LC, GC, MA	RS, T, P, S, DW, TS, O SUP, SRV	ISI w TSL, REM
RWCU	304 SS, 316 SS, CS	8 – 24	CS, 304 SS, 316 SS	CS, SS, NB	UA, FDR, SCC, TF, FS, LC, GC, MA	RS, TS, T, P, S, DW, SUP, SRV, O	ISI w TSL, REM
CRD piping	304 SS, 316 SS (low temp)	< 4	Stub tubes – A600 and SS*	Crevice A182 to head	UA, FDR, MF, SCC	RS, T, P, S, DW, V, O, SRV	ISI w TSL, REM
SLC	304 SS, 316 SS	< 4	304 SS, 316 SS	SS, NB	UA, FDR, MF, SCC	RS, T, P, S, DW, V, O, SRV	ISI w TSL, REM
INST	304 SS, 316 SS	< 4	304 SS, 316 SS	SS, NB	UA, FDR, MF, SCC, MA	RS, T, P, S, DW, V, O, SRV	ISI w TSL, REM
Drain lines	304 SS, 316 SS, CS	< 4	304 SS, 316 SS, CS	SS, NB	UA, FDR, MF, SCC, LC, GC	RS, T, P, S, DW, V, O, SRV	ISI w TSL, REM
Head spray	304 SS, 316 SS, CS	< 4	304 SS, 316 SS, CS	SS, NB	UA, FDR, SCC, TF, LC, GC	RS, P, S, T, DW, SRV, O	ISI w TSL, REM
SRV lines	CS	6, 8, 10, 28	CS	CS	UA, FDR, MF, FS, GC, LC, MA	RS, P, S, T, DW, SRV, O	ISI w TSL, REM
RCIC	304 SS, 316 SS, CS	6, 8	304 SS, 316 SS	SS NB	UA, FDR, SCC, LC, MA	RS, P, S, T, DW, SRV, O	ISI w TSL, REM

* See note in text.
304 SS = 304 series stainless steel (Table 2)
316 SS = 316 series stainless steel (Table 2)
A600 = Alloy 600
CS = carbon steel
CS – SW = seam welded carbon steel
NB = Nickel-based weld (Alloy 82/182)
UA = unanticipated mechanisms

MA = material aging
LC = local corrosion
FDR = fabrication defect and repair
SCC = stress corrosion cracking
MF = mechanical fatigue
TF = thermal fatigue
FS = flow sensitive (inc. FAC and erosion/cavitation)
ISI w TSL = Current ISI procedures with technical specification leakage detection requirements considered.
RS = residual stress

P = pressure
S = Seismic
T = Thermal
DW = dead weight
SUP = support loading
SRV = SRV loading
WH = water (and steam) hammer
O = overload
V = vibration
TFL = thermal fatigue loading from striping
TS = thermal stratification
REM = all remaining mitigation strategies possible (eg. not unique to piping system)

Significant loading sources for BWR piping systems are also listed (column 7). All systems undergo residual stress (RS), pressure (P), thermal (T), seismic (S), safety relief valve (SRV) and dead weight (DW) loading. Water (or steam) hammer (WH) was considered to be important in the feedwater, steam line, and HPCS/LPCS systems. Support loading (SUP) was mainly considered to occur through the snubber support. This is important for the recirculation (RECIRC), RHR, RWCU, and HPCS/LPCS systems. Vibrational loading is listed for the smaller diameter piping systems. This loading is always coupled with mechanical fatigue degradation (column 6). However, vibrational loading is conspicuously absent from the feedwater and SRV lines which both have MF in the list of significant degradation mechanisms.

Overloads (O) are possible for all systems, but they are likely to be external due to pipe whip, jet impingement, or secondary system failure. However, the drain line, CRD, instrument lines, and SLC were deemed to be more likely to be susceptible to internal overloads. The thermal loading was broken down into thermal fatigue loading due to striping (TFL) in the feedwater system, and thermal fatigue loading due to stratification (TS) in the RHR, RWCU, and HPCS/LPCS systems. The head spray line, which is also judged to be TF susceptible, does not have a corresponding thermal fatigue loading source considered.

There were no mitigation and maintenance procedures that were identified by the panel as being unique for any particular BWR piping system. Standard mitigation and maintenance for all systems is in-service inspection (ISI) with credit given for technical specification leakage (TSL) detection. The technical specification leakage threshold is 1 gallon per minute. The effect of all remaining (REM) mitigation and maintenance procedures and issues (Table B.1.6) on the LOCA likelihood should be considered by the panel.

Table B.1.8 PWR LOCA-Sensitive Piping Systems

System	Piping Matls.	Piping Size (in)	Safe End Matls.	Welds	Sig. Degrad. Mechs.	Sig. Loads.	Mitigation /Maint.
RCP: Hot Leg	304 SS, 316 SS, C-SS, SSC-CS CS – SW	30 - 44	A600, 304 SS, 316 SS, CS	NB, SS, CS	TF, SCC, MA, FDR, UA	P, S, T, RS, DW, O, SUP	ISI w TSL, REM
RCP: Cold Leg/Cros sover Leg	304 SS, 316 SS, C- SS, SSC-CS, CS – SW	27 - 34	A600, 304 SS, 316 SS, CS	NB, SS, CS	TF, SCC, MA, FDR, UA	P, S, T, RS, DW, O, SUP	ISI w TSL, REM
Surge line	304 SS, 316 SS, C-SS	10 - 14	A600, 304 SS, 316 SS,	NB, SS	TF, SCC, MA, FDR, UA	P, S, T, RS, DW, O, TFL, TS	TSMIT, ISI w TSL, REM
SIS: ACCUM	304 SS, 316 SS, C-SS	2 - 12	A600, 304 SS, 316 SS,	NB, SS	TF, SCC, MA, FS, FDR, UA (FAC)	P, S, T, RS, DW, O	ISI w TSL, REM
SIS: DVI	304 SS, 316 SS	2 - 6	A600, 304 SS, 316 SS,	NB, SS	TF, SCC, MA, FS, FDR, UA (FAC)	P, S, T, RS, DW, O	ISI w TSL, REM
Drain line	304 SS, 316 SS, CS	< 2"			MF, TF, GC, LC, FDR, UA	P, S, T, RS, DW, O, V, TFL	ISI w TSL, REM
CVCS	304 SS, 316 SS	2 – 8	A600 (B&W and CE)	NB	SCC, TF, MF, FDR, UA	P, S, T, RS, DW, O, V	ISI w TSL, REM
RHR	304 SS, 316 SS	6 – 12			SCC, TF, MA, FDR, UA	P, S, T, RS, DW, O, TFL, TS	ISI w TSL, REM
SRV lines	304 SS, 316 SS	1 – 6			TF, SCC, MF, FDR, UA	P, S, T, RS, DW, O, SRV	ISI w TSL, REM
PSL	304 SS, 316 SS	3 – 6		NB	TF, SCC, MA, FDR, UA	P, S, T, RS, DW, O, WH, TS	ISI w TSL, REM
RH	304 SS, 316 SS	< 2	A600		MF, SCC, TF, FDR, UA	P, S, T, RS, DW, O, V, TS	ISI w TSL, REM
INST	304 SS, 316 SS	< 2	A600		MF, SCC, TF, FDR, UA	P, S, T, RS, DW, O, V	ISI w TSL, REM

C-SS = cast stainless steel
SSC-CS = stainless steel clad carbon steel
FW = fretting wear
TSMIT = thermal stratification mitigation
HREPL = vessel head replacement

PWR Piping Systems: LOCA-sensitive PWR piping systems were also determined (Table B.1.8) along with associated piping safe end, and weld materials; pipe size; significant degradation mechanisms; significant loading sources; and system-dependent mitigation and maintenance procedures. The format of this table is identical to the BWR summary table (Table B.1.7) and the abbreviations have been retained. Unique abbreviations are defined at the bottom of Table B.1.8. It should be noted that the group did not discuss broad differences between Westinghouse (W), Babcox and Wilcox (B&W), and Combustion Engineering (CE) designs. In fact the only place it is noted (Table B.1.8) is in the use of A600 safe ends in the CVCS system in CE and B&W plants. However, any plant design distinctions may be important for certain LOCA classes and should be considered by each expert during analysis.

The LOCA-sensitive PWR systems listed include reactor coolant piping (RCP) hot leg, RCP cold leg, and RCP crossover legs, the surge line, safety injection system (SIS) accumulator line (ACCUM) and SIS direct volume injection (DVI) line, drain lines, chemical and volume control system (CVCS), RHR, SRV lines, pressurizer spray lines (PSL), control rod drive mechanism (CRDM) lines, reactor head (RH), in-core instrumentation (ICI), and instrumentation (INST) lines. The hot leg was segregated from the other RCP components by the group due to its higher operating temperature. The SIS system was divided into the ACCUM and DVI components to account for the piping size, material, and functionality differences. The RH group was intended to capture all the non-CRDM lines that penetrate the upper reactor vessel head. This grouping is distinct from the ICI system. The INST line grouping here considers mainly lines within piping systems and not the reactor. It is worth noting that this grouping is different from the grouping for BWRs where INST lines capture both piping and reactor lines.

The materials utilized in PWR piping systems are similar to those in counterpart BWR systems. One difference is the inclusion of cast stainless steel (C-SS) in RCP, surge line, and ACCUM PWR piping. Also, stainless steel clad carbon steel (SSC-CS) is prominent in certain plant designs within the RCP. There is also less use in general of carbon steel in PWRs. It should be noted that the only material listed in Table B.1.2 which is not explicitly listed in either BWR (Table B.1.7) or PWR (Table B.1.8) piping systems is bimetallic stainless steel welds.

The degradation mechanisms are again tied to the material and functional considerations of the piping system. The FDR and UA categories are included for all systems, as is thermal fatigue. Stress corrosion cracking (SCC) was affiliated primarily with stainless steel piping, but also for carbon steel. Material aging was listed for several higher-temperature, constant-service piping systems (PSL, RHR, RCP, SIS) and mechanical fatigue was deemed important for smaller diameter piping, including the CRDM. Flow sensitive degradation, specifically FAC, was determined to be important in only the SIS system piping, while fretting wear (FW) is listed only for the ICI system.

Significant loading sources are consistent with the BWR piping sources. Pressure (P), seismic (S), thermal (T), residual stress (RS), dead weight (DW), and overload (O) loading histories are sources for all systems. Smaller lines again are again considered to be susceptible to vibration (V) loading and this loading is linked to the MF degradation mechanism. The RCP system is considered to have additional support loading contributions, mainly due to snubber malfunction. Both thermal stratification and thermal fatigue loading due to striping and heat-up/cool-down were listed as significant for the surge line.

Thermal fatigue loading is also important for the RHR and drain lines according to the group, while the reactor head and pressurizer surge lines are influenced by thermal stratification. The PSL also must consider water hammer. Only the SRV lines need to consider SRV transients which is quite different that the BWR classification. All of the major loading variables (Table B.1.5) were considered in either BWR or PWR systems. However, hanger misadjustment and cavitation loading were not specifically mentioned. They would certainly fall under the broader loading categories listed in Table B.1.5, but may need to be considered individually by each expert during the elicitation.

As with BWR piping, in-service inspection with credit for leak detection is existent for all piping systems. All remaining mitigation and maintenance issues should also be considered for their effect on the LOCA frequencies. However, some specific mitigation procedures have been highlighted. This includes thermal stratification mitigation (TSMIT) which some operators practice to limit surge line loads. Also, reactor vessel head replacement (HREPL) is a solution being considered to alleviate CRDM cracking concerns. The group will need to consider the extent and effectiveness (now and in the future) of each of these specific procedures.

Base Case Development

During the elicitation, each expert will determine the biggest contributors for each LOCA frequency threshold category (Table B.1.1) separately for BWR and PWR plants. The information summarized in Tables B.1.2 – B.1.8, and discussed in the previous sections, is simply intended to identify those issues and variables which contribute to the LOCA frequency distributions. Each expert will determine the magnitude and likelihood of each variable separately, but more importantly will determine the importance of the interrelationships among the variables.

Each panel member will not be asked to provide absolute LOCA frequencies. All questions will be structured so that relative differences with a specific base case will be queried. The base cases will be associated with absolute frequencies and quantitative LOCA estimates will be derived from these values and the relative relationships provided during the elicitation. The group spent quite a bit of time and effort both understanding the role of the base cases in the elicitation and assessing their importance. Ideally the base cases are chosen to represent significant contributing conditions to the total LOCA frequency estimates in order to minimize the extrapolation required during questioning. Representative, significant base cases should therefore theoretically improve the elicitation accuracy. Once this concept was understood, the group settled on several base case conditions for PWR and BWR systems. Pete Riccardella provided the original suggestions which were largely adopted by the panel after the analysis framework was clarified. The base case discussion evolved over the Wednesday and Thursday meeting days. All of the discussion will be summarized in this section for consistency.

Two base cases were developed for BWR piping systems (Table B.1.9). The first case will examine the recirculation system piping. All the various piping sizes will be considered, and original 304 stainless steel material will be assumed that has not been replaced during plant operation. The safe end is non-creviced Alloy 600 which is connected to the piping and vessel by Alloy 82/182 weld material. Only the IGSCC (subcategory of SCC, Table B.1.2) will be considered as the degradation mechanism. The loading will consist of pressure, residual stress, and dead weight nominal components. Transients to be considered include SRV loads and seismic. The base case will assume that normal water chemistry (NWC) is used in the system.

The next BWR base case will examine the feed water system. A 12 inch diameter carbon steel pipe will be analyzed. The safe end and weld materials were not specified and will need to be defined by the base case analysis team (to be discussed subsequently). The degradation mechanisms for this base are flow accelerated corrosion (FAC) and thermal fatigue (TF). Loading sources include pressure, thermal, residual stresses, and dead weight nominal components. Thermal fatigue loading (TFL) from stratification and possibly striping will provide alternating loads, and water flow velocities will also be included to assess fluid loading. Transients for this base case will include water hammer and seismic. Once again NWC will be assumed.

Three base cases were constructed for PWR systems (Table B.1.9). The first will examine the hot leg in the reactor coolant piping system. A 30 inch diameter Type 304 stainless steel pipe will be considered with Alloy 600 safe ends and Alloy 82/182 bimetallic welds. This base case will examine thermal fatigue and PWSCC. Loading will again include pressure, thermal, residual stress, and dead weight nominal loads and thermal fatigue alternating loads. Transients will include seismic loading and a pressure pulse transient, the magnitude and duration of which is still to be determined.

Table B.1.9 Base Case Analyses

Plant Type	System	Piping Size (in)	Piping Material	Safe End Material	Weld Material	Degradation Mechanism	Loading	Mitigation/ Maint.
BWR	RECIRC	12 – 28	Original 304 SS	Non creviced A600	A82	IGSCC	P, S, RS, DW, SRV	NWC, leak detection, ISI
	Feed water	12	CS			FAC, TF	P, S, T, RS, DW, WH, Flow velocities	NWC, leak detection, ISI
PWR	RCP – Hot Leg	30	304 SS	A600	A82	TF, PWSCC	P, S, T, RS, DW, pressure pulse	ISI, leak detection
	Surge Line	10	304 SS		A82 at Pressurizer	TF, PWSCC	P, S, T, RS, DW, pressure pulse	ISI, leak detection
	SIS: DVI HPI/makeup	4	SS/CS			TF	P, S, T, RS, DW, pressure pulse	ISI, leak detection

The next PWR base case is a 10 inch diameter surge line. The surge line material is Type 304 SS and an Alloy 82/182 bimetallic weld will be included at the pressurizer. No safe end materials will exist. Once again, thermal fatigue and PWSCC will be considered. Loading will include pressure, thermal, residual stress, and dead weight nominal loads and thermal fatigue alternating loads. Transients will include seismic loading and a pressure pulse transient, the magnitude and duration of which is still to be determined.

The final PWR base case was the most ill-defined case because it was added after initial group discussions. It was added to provide a base case for a smaller diameter piping system. This base case will need to be defined more completely prior to analysis. A 4 inch diameter high pressure injection/makeup line will be examined for thermal fatigue degradation. The piping material, welding, and safe end materials still need to be specified. Nominal loading is once again provided by pressure, thermal, residual stress, and dead weight loads. Thermal alternating loads will be defined and seismic and pressure pulse transients will be considered.

Absolute LOCA frequencies will be developed for each base case and for each threshold leak rate category defined in Table B.1.1. There are six leak rate categories and five base cases; therefore at least thirty separate calculations will be required to fully define the base case frequencies. The base cases will include analysis of many welds and other piping components. The LOCA frequencies for the system will obviously be the summation of the contributions from all system components. The frequencies will also be determined as a function of time. Three time periods will be evaluated: 25 years after plant startup (current day), 40 years after start-up (original design life), and 60 years after start-up (end-of-life extension).

The panel decided that seismic transients would be handled as part of a sensitivity study. As mentioned previously, seismic-induced LOCAs will not be determined as part of this elicitation for several reasons: the PRA models that these estimates will be used for do not consider seismic loading; there has been significant work in developing seismic LOCA estimates; and the group has no specific expertise in seismic analysis. However, many panel members are experienced in conducting analysis when the seismic loading history is provided. Many panel members also seem comfortable with comparing other loading histories to seismic events. Therefore, it was decided that the elicitation would ask for comparisons with conditional seismic loading (probability of occurrence of 1) separately in order to conduct a sensitivity analysis of the effect of seismic loading.

Originally, the group determined that the seismic loading magnitude will be 0.3 g for the sensitivity study. Pete Riccardella suggested changing the 0.3 g criteria for the seismic condition to an ASME Code faulted-stress condition. By doing this soil conditions, damping characteristics, etc. do not have to be considered. While there was some discussion on the merits of this suggestion, the issue was not finalized.

The base cases will also perform a sensitivity analysis to determine the effect of the frequencies both with and without in-service inspection. Standard ISI techniques will be considered in the analysis. Credit will also be given in this analysis for leak detection. The leak detection threshold for the base case analysis will be leak rates which are commensurate with the service history database as defined in the SKI-pipe database. The specific leak rate threshold associated with the database must be defined.

A base case team was established to develop four separate LOCA frequency estimates for each of the five base cases. The base case team will consist of **Vic Chapman, Bengt Lydell, David Harris,** and **Bill Galyean.** The team will model a specific piping system and define plant operating characteristics for each base case. Then, the team will develop input for each of the five LOCA variables (material, geometry, loading, degradation mechanism, and mitigation/maintenance) within the parameter constraints identified in Table B.1.9. The team will share information to ensure that each analysis is considering the same nominal conditions for each base case.

Each base case team member will develop their own LOCA estimates for each system using whatever methodology they choose. The likely general approaches for each team member are summarized in Table B.1.10. Each specific methodology will require additional assumptions. It is incumbent that each team member catalog required assumptions and document the methodology used to arrive at their base case LOCA estimates. This information will then be rigorously, yet concisely, presented to the remaining panel members. This should allow each panel member to completely understand the assumptions, methodology, and results generated by each base case team member. It needs to be stressed that once the general conditions are developed, the base case members should independently develop their estimates without further consultation. This step is necessary to retain realistic sample uncertainty in the calculated results.

Table B.1.10 Base Case Approaches

Base Case Team Member	Analysis Approach
Vic Chapman	PFM using PRODIGAL code
Bill Galyean	Direct analysis of service history data
David Harris	PFM using PRAISE code
Bengt Lydell	Direct analysis of service history data

The base case team would collaborate to ensure that the PFM analyses accurately capture that leaking pipe service history. This is the one aspect of the exercise that contains plant operating experience data. Initial PFM calculations will be conducted based on best-estimate assumptions and the current leak rate frequency predictions will be compared with the service history. At this point, PFM input assumptions may be changed in order to match the operating history. Each PFM model should accurately document the input variables, any model changes, and results both before and after benchmarking. This benchmarking exercise will help the remaining panel members gauge uncertainty in the calculations.

The panel will supply background information to the base case calculation team as required. All requests for background information will be coordinated by Rob Tregoning to ensure proper cataloging and dissemination to the group. Some volunteers already offered certain background information. Bruce

Bishop has run his own PFM models for seven Westinghouse plants that could ultimately be used to help verify the base case calculations.

It was also stressed by the panel that it is important to make the PFM modeling conditions as close as possible to the postulated service history conditions so that the various base case approaches can be directly compared to assess uncertainty and possible inaccuracies. For instance, many existing PFM models assume that all repairs are perfect (no defects). However, many repairs introduce new defects and most large flaws are associated with repairs. This fact (and other similar issues) is naturally captured within the service history database, and needs to be considered within the PFM modeling if possible.

Confidentiality

A discussion was held on the confidentiality of participant's responses during the exercise. Rob Tregoning indicated that all information provided as part of this exercise will remain confidential and will not be distributed to anyone not specifically involved in the exercise. The kick-off meeting has been videotaped, but this will not be distributed outside of the group. Elicitation sessions will be taped for accuracy, but this information will also not be made public. There will be public reporting of the assumptions, methodology, elicitation results, and calculated LOCA frequencies that stem from this exercise. However, the summary reporting will only identify the names, affiliations, and possibly credentials of the expert elicitation panel and the facilitation team early in the report. No reference to individual opinions will be documented.

Day 3 – Thursday, February 6, 2003

Use of Base Case in Elicitation

The base cases LOCA frequencies will provide absolute LOCA estimates that each panel member will use to anchor the relative likelihood of LOCAs in other (non-base case) piping systems. Each panel member will also determine how well the base cases depict expected current and future LOCA performance in the piping systems that they model. It is therefore *not* important that a panel member agree with the modeling assumptions, approach, and results provided by the base case team. However, it is imperative that the base case development is completely understood by each panel member. Each panel member will be able to correct perceived deficiencies in the calculated base case frequencies during the elicitation. Each panel member will also determine, relative to the base case results, the LOCA contributions of other (non base case) piping systems, and the contributions and uncertainty induced by the primary piping system variables.

Reference Case Development

In order to decompose the elicitation topics further, the group determined that it would be useful to further decompose non base case piping systems and variables. This was accomplished by defining a set of reference conditions for each LOCA-sensitive piping system identified in Tables B.1.7 and B.1.8. The reference conditions are similar to the base cases in that they define a unique set of conditions (materials, geometric variables, mitigation and maintenance procedures, and degradation mechanisms) that can be analyzed. They are different from the base cases in that absolute LOCA frequencies will not be developed for the reference cases by the base case team. The reference cases for various systems will be compared to determine the *relative* LOCA-severity among piping systems. LOCA-severity variability within any specific system can then be gauged with respect to the reference case for that system. It will be up to each panel member to determine the method for developing these relative comparisons.

The BWR reference cases (Table B.1.11) represent specific combinations of the possible BWR piping as previously listed in Table B.1.7. In general, one material and degradation mechanism has been chosen is ideally representative of each piping system. The effect of fabrication defects and repair (discussed earlier) should be considered for its effect on the other degradation mechanism in all cases. Nominal pressure (P), thermal (T), residual stress (RS), and dead weight (DW) loading should be considered for all cases. One loading transient was identified for each system. All the transients should be fairly well identified based on past discussion, but the overload transient for the CRD piping needs to be better defined. In all cases, the snubber is considered to be functional.

Table B.1.11 BWR Reference Case Conditions

System	Piping Material	Piping Sizes (in)	Safe end	Welds	Degradation Mechanisms	Loading	Mitigation and Maintenance
RECIRC	304 SS	10, 12, 20, 22, 28	304 SS	SS	SCC, FDR	P, T, RS, DW, SRV	NWC, ISI w. TSL, 88-01 (AI), 182
Feed Water	CS	10, 12, 12 - 24	304 SS	CS	FAC, FDR	P, T, RS, DW, WH, TFL	NWC, ISI w. TSL, 88
Steam Line	CS – SW	18, 24, 28	CS	CS	FAC, FDR	P, T, RS, DW, SRV	NWC, ISI w. TSL, 88
HPCS, LPCS	CS	10, 12	304 SS	CS	TF, FDR	P, T, RS, DW, TS, SRV	NWC, ISI w. TSL, 88
RHR	304 SS	8 – 24	304 SS	SS	SCC, FDR	P, T, RS, DW, TS, SRV	NWC, ISI w. TSL, 88
RWCU	304 SS	8 – 12	304 SS	SS	SCC, FDR	P, T, RS, DW, TS, SRV	NWC, ISI w. TSL, 88
CRD piping	304 SS	< 4	A600 and SS	Creviced NB welds	SCC, FDR	P, T, RS, DW, O	NWC, ISI w. TSL, 88
SLC	304 SS	< 4	304 SS	SS	SCC, FDR	P, T, RS, DW, SRV	NWC, ISI w. TSL, 88
INST	304 SS,	< 4	304 SS	SS	MF, FDR	P, T, RS, DW, V, SRV	NWC, ISI w. TSL, 88
Drain lines	304 SS	< 4	304 SS	SS	SCC, FDR	P, T, RS, DW, SRV	NWC, ISI w. TSL, 88
Head spray	304 SS,	< 4	304 SS	SS	TF, FDR	P, T, RS, DW, SRV	NWC, ISI w. TSL, 88
SRV lines	CS	6, 8, 10, 28	CS		MF, FDR	P, T, RS, DW, SRV	NWC, ISI w. TSL, 88
RCIC	304 SS	6, 8	304 SS	SS	SCC, FDR	P, T, RS, DW, SRV	NWC, ISI w. TSL, 88

Specific piping, safe end, and weld materials were not determined by the group. Table B.1.11 represents an initial attempt to select these materials based on the general discussion. However, the group did decide to consider uncrevised Type 304 where proper in the reference cases. The mitigation and maintenance for all systems should assume normal water chemistry (NWC). Standard ISI with technical specification leakage detection should be considered along with augmented inspection as defined in generic letter 88-01. The mitigation and maintenance also has listed alloy 182 and stress improved, but these concepts need to be better defined and summarized.

The PWR reference case conditions are provided in Table B.1.12. The reference cases were again distilled from the LOCA-sensitive PWR piping systems (Table B.1.8). The philosophy behind this table was consistent with the BWR reference case development (Table B.1.11) with a few notable exceptions. The effects of fabrication defects and repair on the other listed degradation mechanisms should again be considered in all piping systems. However, several PWR reference cases list multiple other degradation mechanisms which is a departure from the BWR approach (Table B.1.11). The PWR piping reference cases also account for nominal loading supplied by pressure, thermal, residual stress, and dead weight loading to each system. However, very few PWR systems have associated loading transients while all BWR systems do. It may be necessary to add associated PWR transients for consistency. The PWR piping, safe end and weld materials were also not specified. Some initial choices have been made in Table B.1.12, but feedback from the group is required in order to finalize selection. The mitigation and maintenance to be considered for each reference case consists of ISI with TSL. No other special mitigation procedures were identified.

Table B.1.12 PWR Reference Case Conditions

System	Piping Material	Piping Sizes (in)	Safe end	Welds	Degradation Mechanisms	Loading	Mitigation and Maintenance
RCP: Hot Leg	304 SS	30 - 44	A600	NB	TF, SCC,FDR	P, T, RS, DW	ISI w TSL
RCP: Cold/ Crossover Legs	304 SS	22 - 34	A600	NB	TF, FDR	P, T, RS, DW	ISI w TSL
Surge line	304 SS	10 - 14	A600	NB	TF, FDR	P, T, RS, DW, TFL, TS	ISI w TSL
SIS: ACCUM	304 SS	10 - 12	304 SS	SS	TF, FDR	P, T, RS, DW	ISI w TSL
SIS: DVI	304 SS	2 - 6	304 SS	SS	TF, FDR	P, T, RS, DW	ISI w TSL
Drain line	304 SS	< 2"		SS	MF, TF, FDR	P, T, RS, DW, V	ISI w TSL
CVCS	304 SS	2 – 8		SS	TF, MF, FDR	P, T, RS, DW, V	ISI w TSL
RHR	304 SS	6 – 12			TF, FDR	P, T, RS, DW, TS	ISI w TSL
SRV lines	304 SS	1 – 6			TF, FDR	P, T, RS, DW, SRV	ISI w TSL
PSL	304 SS	3 – 6		NB	TF, FDR	P, T, RS, DW, WH	ISI w TSL
RH	304 SS	< 2	A600		TF, FDR	P, T, RS, DW, TS	ISI w TSL
INST	304 SS	< 2			MF, TF, FDR	P, T, RS, DW, V	ISI w TSL

It is important that the baseline and reference case conditions be clearly defined prior to the start of the elicitation so that each panel member understands the general attributes of each of these cases. As mentioned previously, the elicitation questions will be structured to query the variability and uncertainty associated with each piping system with respect to the reference cases. Each panel member will compare the reference and base cases to assess the relative importance of each piping system to the total LOCA frequencies. Every effort will be made to accommodate all requests and information will be shared among the group. Additionally, any areas or issues which are not clear to a panel member should be raised to Rob Tregoning as soon as it arises.

Non-Piping LOCA Contributions

The final portion of the meeting concentrated on developing issues associated with non-piping contributions (passive failures only) to the LOCA frequencies. Active components will be analyzed separately during this program from service history data. Because active components have maintenance plans, the group in general expects that the failure rate of these components will not increase in the future. Service history should therefore adequately represent active component failure rates. This rationale is the basis for considering only passive component failures within this elicitation.

The non-piping contributions will be combined with the piping component to determine the total LOCA frequency (Figure B.1.1). The group decided to break these issues down by component functionality. This is an analogous approach used to tackle the piping contribution which initially segregated piping systems by functionality. Five main components were determined as candidates for passive failures: the pressurizer, the reactor pressure vessel (RPV), valves, pumps, and steam generators/steam systems. The valve component category encompasses both pressure isolation valves at Class 1 to Class 2 piping boundaries and also loop-stop valves. The pumps category only considers pumps in the reactor coolant or recirculating water system.

For each component category, the panel developed sub-categories which represent specific possible failure modes (e.g. what portions of the component could fail passively). Each failure mode is governed by the same variables that are important for piping systems (material, geometry, loading, degradation mechanisms, and mitigation/maintenance). Unfortunately the group did not have sufficient time or resources to fully develop comprehensive variable lists in the same manner as for piping systems.

Table B.1.13 illustrates the failure modes developed for pressurizer failures. Please note that all table abbreviations for this and all subsequent tables in this section are as previously defined unless indicated. Bold items in the failure mode sub-category of Table B.1.13 (and all following tables) indicates that operational data exists which captures that component failure. For instance, in Table B.1.13, the group thinks that data is available on heater failures.

Table B.1.13 Pressurizer Failure Scenarios

Component	Geometry	Material	Degradation Mechanisms	Loading	Mitigation/ Maintenance	Comment
Shell		A600C-LAS, SSC-LAS	GC, SCC, MF, FDR, UA			Boric acid wastage from OD
Manway		NB-LAS, SSC-LAS, LAS, HS-LAS (Bolts)	GC, SCC, MF, SR, FDR, UA			Bolt failures
Heater Sleeves	Small diam. (3/4 to 1 in)	A600, SS	TF, MF, SCC, FDR, UA			Req. multiple failures
Bolted relief valves		C-SS	MA, FDR, UA			
Nozzles		SSC-LAS C-SS	CD, TF, SCC, MA, FDR, UA, GC			Same as surge line

NB-LAS = nickel-base clad low alloy steel
SR = Stress Relaxation and loss of preload

The panel identified failures in the pressurizer shell, manway, heater sleeves and nozzles as passive LOCA candidates. Also, the pressurizer bolted relief valves could fail. The group generally did not have information on component geometries and loading and mitigation/maintenance were not discussed by the panel due to lack of time. However, some specific issues were discussed for each of these failure modes. The shell failure envisioned would most likely occur by boric acid wastage from the outer diameter of the shell. Manway failures would result by multiple bolt failures. Heater sleeves fail due to PWSCC, but as a result of their size, multiple failures are required in order to result in a LOCA. Bolted relief valves could fail due to steam cutting or localized bolt corrosion resulting from boric acid leaks.

The reactor pressure vessel (RPV) failure modes (Table B.1.14) focused on vessel head bolt failure, failure of CRDM connections, nozzle failure, RPV wastage, and RPV corrosion fatigue. Upper head vessel head bolt failure is most likely due to human error during removal at each refueling cycle. Human error could occur as a result of improper installation procedures. Problems, however, could be identified during prestart-up inspection. The lower head bolts are not removed during refueling and they could be susceptible to common cause failure resulting from local bolt corrosion leading to several simultaneous bolt failures. A certain percentage of these bolts are inspected at each outage and the assumption is that inspection would not be effective in identifying the degradation prior to failure. These requirements may uncover the likelihood of common cause errors leading to some latent failure that is not immediately evident and shed light on other possible failure mechanisms. An example of a common cause failure is a torque wrench/tensioner which is out of calibration so that all bolts are improperly installed and then can possibly fail during operation.

CRDM connections far outside of the reactor could be welded, bolted, or threaded and seam welded. The degradation mechanism would be a function of the specific connection. For instance, welded connections would be susceptible to the mechanisms and loading discussed previously for CRDM components. Bolted CRDM connections would be subject to steam cutting, boric acid corrosion, aging and other degradation mechanisms that are unique to bolts. It must be stressed that the CRDM connections in this table refers to the CRDM which connects to the drive mechanism. Inboard connections are considered to be part of the "CRDM piping system" discussed earlier. For bolted connections, this demarcation line is the flange joint. There group identified service data for CRDM leakage from bolted flanged connections.

Table B.1.14 Reactor Pressure Vessel (RPV) Failure Scenarios

Component	Geometry	Material	Degradation Mechanisms	Loading	Mitigation/ Maintenance	Comment
Vessel Head Bolts		high strength steel	GC, FDR, UA		Human error	Removal leading to human error (common cause failure) during refueling
RPV wastage		SSC-LAS LAS	GC, FDR, UA, MA			LAS = some BWR upper head, Boric acid wastage (upper & lower head, shell)
CRDM connections		SS	FDR, UA			welded, bolted, threaded + seal weld
CRDM	4-6	A600 base nozzle, SS, C-SS, and NB-LAS housing with NB weld	SCC, TF, MF, LC, GC, FDR, UA	P, S, T, RS, DW, O	HREPL, ISI w TSL, REM	Nozzles and piping up to connection
Nozzles		LAS, SSC-LAS,	TF, MF, LC, GC, SCC, FDR, UA			LAS = BWR only
ICI	< 2"	304 SS, 316 SS	MF, SCC, TF, FW, FDR, UA	P, S, T, RS, DW, O, V	ISI w TSL, REM	
RPV Corrosion Fatigue		SSC-LAS LAS	LC, MF, MA FDR, UA			LAS = some BWR upper head, Initiate at cladding cracks (upper & lower head, shell)
BWR penetrations		SS	SCC, LC, FDR, UA			Stub tubes, drain line, SLC, instrumentation, etc.
PWR penetration		SS, A600	SCC, FDR, UA, LC, MF, TF			

NB-LAS = nickel-base clad low alloy steel
SR = Stress Relaxation and loss of preload

There are two RPV degradation mechanisms which were specifically discussed. The first was degradation of the shell, upper head, or lower head due to boric acid corrosion. The second mechanism was corrosion fatigue developed at through-thickness cladding cracks in the shell, upper head, or lower head. The nozzle category is subject to similar degradation mechanisms as in the attached piping. It should be stressed that the nozzle category only considers the flared portion of the nozzle up to the reactor shelf. The nozzle safe end was earlier defined as part of the piping system. The group identified that some data on nozzle issues exists.

Valve failure modes are summarized in Table B.1.15. The cast stainless steel valve bodies are susceptible to an array of potential degradation mechanisms. These include cavitation (CAV), thermal fatigue (TF), and material aging (MA). Casting defects (CD) are another particular concern. Failure due to the other mechanisms listed could initiate at either the defects, or repairs of those defects. The main steam isolation valve (MSIV) body is associated with similar failure modes. Specific failure modes for valve bonnets, and valve bonnet bolts were not discussed. Presumably, bonnet bolt failures would be susceptible to the same failure mechanism of other bolts: aging, boric acid corrosion, steam cutting, etc. The hot leg/cold leg loop isolation valve failure modes were also not discussed. However, failures in these valves could be described in terms of the bonnet, body, or bonnet bolt failure sub-categories listed earlier. It should also be noted that valve sizes are generally consistent with the piping system where they are located.

Table B.1.15 Valve Failure Scenarios

Component	Geometry	Material	Degradation Mechanisms	Loading	Mitigation/ Maintenance	Comment
Valve Body		CS, SS C-SS	FAC, CAV, LC, TF, MA, GC, CD, SCC, FDR, UA			CS, SS = BWR only
Valve Bonnet		CS, SS C-SS	FAC, LC, GC, SCC, MA, CD, FDR, UA			CS, SS = BWR only
Bonnet Bolts		HS-LAS	GC, SCC, FDR, UA SR			
Hot Leg/Cold leg loop isolation valves			FDR, UA			
MSIV Body			CAV, TF, MA, CD			

HS-LAS = High Strength Low Alloy steel (SA540 GrB23, SA193 GrB7)
CAV = Cavitation Damage
SR = Stress Relaxation and loss of preload

Steam generator tube rupture (Table B.1.16) can occur from a variety of different mechanisms including thermal fatigue, mechanical fatigue, stress corrosion cracking, and general corrosion. The tubes can also be degraded by mechanical deformation (MECDEF), or denting, during installation, inspection, or cleaning. Steam generator tubes are too small to lead to a LOCA due to a single tube failure. Therefore, multiple tube rupture needs to also be considered in order to achieve a certain size LOCA. There is data which exists for steam generator tube rupture.

Steam generator failure can also occur at the manway (specifically bolt failure), the steam generator shell, or the nozzles. These various failure modes were also not sufficiently discussed so little information has been defined in Table B.1.16. However, the nozzle failure issues will likely be similar to the associated piping system, while manway bolt failure would be caused by the same types of mechanisms as for other bolt failures.

The pump failure modes (Table B.1.17) are similar to many of the failure modes already discussed for other components. The cast pump bodies are potentially subject to the same degradation mechanisms (CAV, TF, CD, MA) as other cast components. The recirculation (RECIRC) bonnet bolts and RCP nozzle are also susceptible to mechanisms discussed earlier. The only unique mode considers an incipient failure of a pump flywheel which could initiate collateral damage in other components or in other piping systems. There was no appropriate passive pump failure data that was identified by the group.

Table B.1.16 Steam Generator/Steam System Failure Scenarios

Component	Geometry	Material	Degradation Mechanisms	Loading	Mitigation/ Maintenance	Comment
Tube Rupture	5/8 to 3/4" diam.	A600	TF, MF, SCC, GC, LC, FRET, MECHDEF, FDR, UA			single and multiple tube rupture
Manway Bolts		CS, LAS	SCC, GC, LC, SR, FDR, UA			
Shell		CS, LAS,	GC, LC, MF, TF, FDR, UA			
Nozzles to safe end		SSC-LAS CS, LAS SSC-CS	FAC, SCC, FDR, UA			
Tube Sheet Failure		NB-LAS A600	SCC, GC, FRET, MF, FDR, UA			

FRET = fretting or mechanical wear

Table B.1.17 Pump Failure Scenarios

Component	Geometry	Material	Degradation Mechanisms	Loading	Mitigation/ Maintenance	Comment
Pump Body		C-SS, SSC-CS	CAV., TF, CD, MA, SCC, fatigue			
RECIRC Bonnet Bolts		HS-LAS	SCC, GC, SR			
RCP nozzle						
Flywheel failure						initiating collateral damage – secondary pipe failure

HS-LAS = High Strength Low Alloy steel (SA540 GrB23, SA193 GrB7)
SR = Stress Relaxation and loss of preload

It is obvious that the non-piping passive LOCA sources have not been nearly as well-defined as the piping system sources, and they must be better defined prior the elicitations. However, due to the number and complexity of the components, the panel realized that it may not be possible to fully define all the variables listed in the tables above. At a minimum, the group decided that it would need isometric drawings for as many of these components as possible.

The manner for arriving at the LOCA contributions of these other components will be similar to the approach followed for the piping contribution. Reference cases will be developed and absolute LOCA estimates will be assigned to those numbers. However, these reference cases will be based strictly on data. The bolded items in Tables B.1.13 – B.1.16 are component failures that are supported by service history data. This data will first need to be accumulated and analyzed. Karen Gott and Bill Galyean are possible sources for some of this data. There is an EPRI database called PM-BASIS which consists of mainly active components, but there may be some data on bonnets and packings. Also, Spence Bush may have some data for these components that might be useful to the group. Finally, the group discussed that there may be data available for feed water nozzles.

Once the data is developed, it will be made available to the group. This data will make up the base case information for the non-piping components. The group will also be asked how representative the base case data is for future (end-of-life-extension) LOCA estimates. The LOCA propensity (for each leak threshold rate) for the components without data will also be queried relative to these base cases. This approach is identical to the development of the piping LOCA contributions discussed earlier.

MEETING MINUTES FOR SECOND ELICITATION MEETING FOUR POINTS SHERATON, BETHESDA MD

Day 1 – June 4, 2003 - Base Case Review

Dr. Rob Tregoning (USNRC) welcomed everyone to the Second Elicitation meeting and reviewed the agenda for the two days. Everyone in attendance introduced themselves. A package of the Day 1 presentations was provided to everyone. Rob warned the group that there was a lot of material to cover in each presentation. Furthermore, he indicated that we should not treat what is presented at this meeting as final, but more of a snapshot of where we are presently. Next, Rob reviewed the objectives for the first day of the meeting. The objectives for the first day were:

1. Review base case conditions
2. Understand assumptions, methodologies, and results calculated by each base case member
3. Understand important factors and variables that lead to differences among results
4. Determine what additional calculations are required to complete the base case analysis

Rob also indicated that he was not asking everyone to agree with the results to be presented, but simply to understand what was done. The panel members can state differences of opinion during their individual elicitations.

Next, the second day agenda (Elicitation Coordination) was discussed. The second day agenda has been adjusted slightly to ensure adequate time for the topic of non-piping LOCA frequency determination since it received less attention at the last meeting

Rob then reviewed the meeting objectives for Day 2. The Day 2 objectives were:

1. Finalize elicitation question sets and provide consistent understanding of each question.
2. Determine methodology for evaluating non-piping LOCAs and identification of non-piping base case data.
3. Determine methodology for evaluating conditional seismic loading including determination of seismic loading magnitude.
4. Determine what information experts will require prior to their elicitations and assign action items for providing information.
5. Develop final schedule and time-frame for upcoming elicitations.

Bruce Bishop (Westinghouse) asked if the panel would get a status report on the new Probabilistic Fracture Mechanics (PFM) code being developed as part of the USNRC program. Rob indicated that time was not available at this meeting, but that sometime in the fall or winter an initial meeting will be set up where this new code being developed by Battelle and Engineering Mechanics Corporation of Columbus (Emc^2) could be presented.

Presentation 1 – Base Case Review and Summary Results by Rob Tregoning of the USNRC

The purpose of this presentation was to review the conditions analyzed by the base case team and summarize the calculated results to date. This talk served as a prelude for the next four presentations by the base case team members.

The base case results will be used to anchor elicitation responses by the elicitation panel members as part of their individual elicitations. The elicitation members can use one or all of the base case results directly for their anchoring, or provide their own base case analysis if they choose. There were a total of five (5) base cases defined at the first elicitation meeting in February, see Table B.2.1.

Table B.2.1 Base Case Conditions

Base Case Identification	Piping System	Pipe Diameter, inches	Piping Materials	Degradation Mechanisms	Loading	Mitigation
BWR-1	Recirculation	12 to 28	Type 304 stainless (originally), non-crevised A600 safe ends, nickel based (NB) welds	IGSCC	Pressure (p), residual stress (RS), dead weight (DW), safety relief valve transient (SRV)	Normal water chemistry (NWC), leak detection (LD), in-service inspection (ISI), augmented inspection per Generic Letter 88-01 (88-01)
BWR-2	Feedwater	12	Carbon steel	Flow-assisted corrosion (FAC), thermal fatigue (TF)	P, RS, DW, thermal (T), water hammer (WH),	NWC, LD, ISI, 88-01
PWR-1	Hot leg	30	Type 304 stainless, A600 safe ends, NB welds	Primary water stress corrosion cracking (PWSCC), TF	P, RS, DW, T, pressure pulse (PP)	LD, ISI
PWR-2	Surge line	10	Type 304 stainless, A600 safe ends, NB welds at pressurizer	PWSCC, TF	P, RS, DW, T, PP	LD, ISI
PWR-3	High pressure injection makeup nozzle (HPI/MU) (B&W)	4	Stainless and carbon steel	TF	P, RS, DW, T, PP	LD, ISI

As part of the base case effort, the base case members were to evaluate the LOCA frequencies at 25 years (current day), 40 years (end of life), and 60 years (end of plant life extension). These results will then be used by the individual elicitation panel members to anchor their respective responses so that they can estimate the various LOCA frequencies at these same time periods.

The goal for the base case members is to calculate results for the set of conditions listed in Table B.2.1. The base case members also shared their results and presentations prior to this meeting so that there was a common format for the presentations. At this time the base case results comparison charts in the handouts should be viewed as works in progress. Furthermore, some results from Vic Chapman (OJV Consultancy) are still forthcoming. Vic and Chris Bell need to provide additional information to the panel members on how they conducted their base case analyses.

The current base case results are summarized in slides 16 through 20 of this presentation. For David Harris's calculations, the frequency results at 60 years were average over the 20 year time period from 40 to 60 years while the 25 year estimates were averaged over the first 25 years of operation. It is important that results are consistent (with consistent assumptions and conditions) among each base case team member.

The hot leg results (PWR-1 on page 18 of the handout) indicates large initial uncertainty. Bruce Bishop questioned if the results are for individual welds or the overall system. The response was that the intent was that these results should reflect the frequencies for the overall system. It was pointed out that the hot leg results should reflect the LOCA frequencies for the hot leg only. The results should not consider all the other lines associated with the reactor coolant system (RCS), i.e., the cold leg, cross over, surge line, etc.

The slides 21 and 22 (Remaining Work and Differences Among Methodologies) of Rob's presentation (Base Case Conditions and Summary Results) were not included in the handout and were to be filled out later by the team members. These updates will be posted on the ftp site once available. It was indicated that the LOCA frequencies for the lower leak rate categories (> 380 lpm [100 gpm]) include all the incidences of LOCAs in the higher leak rate categories, e.g., the 380,.000 lpm (100,000) gpm bin should include all incidences of LOCAs in the 1,900,000 lpm (500,000 gpm) bin.

Presentation 2 – Report No. 1 by Base Case Team to Expert Panel on LOCA Frequency *Distributions*
By Bill Galyean, INEEL

Bill employed a "top down" approach in his analysis of the past service experience historical data. His database represents approximately 2,600 LWR-years of operating experience. The resultant average age of a plant is 23 years. In that 2,600 years of LWR operating experience there have been no passive system LOCAs with a resultant leak rate greater than 380 lpm (100 gpm) (Category 1 LOCAs).

As part of this analysis, Bill assumed that cracks and leak events are indicators of LOCA frequencies. They indicate system susceptibility. In order to get a LOCA, Bill's analysis assumed that a piping system must first have a leak or a crack. In the first 2,647 years of US LWR experience represented in Bill's database there have been approximately 1,100 crack and leak events, but *no* LOCAs. Note, at the first elicitation meeting the demarcation between leaks and breaks was set at a 380 lpm (100 gpm).

A comment was made that small pipes are more susceptible to LOCAs than large pipes, i.e., large pipes are less likely to fail catastrophically. A question was raised as to why limit the analysis to US operating experience only. Bill did not categorically know whether there have been any 380 lpm (100 gpm) LOCAs worldwide. Another reason to limit his analysis to US operating experience is that there are some fundamental design differences between US and other overseas plants. Pete Riccardella (Structural Integrity Associates) thought that if Bill had included foreign experience that the number of years of operating experience would have about doubled so Bill's LWR LOCA frequency number of 1.9E-04/year be would reduced by a factor of two to approximately 1.0E-04/year. It is important to understand the basis for this 1.9E-04/year number since everything else is referenced to this number. This number is the total number of LOCAs of all sizes.

It was noted that this is a different approach than followed in NUREG/CR-5750. This analysis was not an attempt to update NUREG/CR-5750. Pete Riccardella asked if this database included all of the small diameter socket weld cracks that occur due to vibration fatigue. Bill indicated that this was the case, even though these small diameter lines could not result in a 380 lpm (100 gpm) leak. It was pointed out that there was no distinction between cracks and leaks in Bill's analysis. Any crack deeper than 10 percent of the wall thickness was included in the analysis.

Gery Wilkowski (Engineering Mechanics Corporation of Columbus) asked if the analysis of the feedwater system (BWR-2) included flow accelerated corrosion (FAC) as a failure mechanism. Gery noted that on the secondary side there have been large breaks in some piping systems due to FAC. Bill indicated that he limited his database search to those systems that affected reactor coolant pressure boundary integrity.

Karen Gott was surprised at the low number of incidences for the feedwater system. Most of the problems seen to date with the feedwater systems have been outside the primary portion of system. Furthermore, cracks in nozzles are associated with the RPV and not piping. For the PWR systems there has been a lot of feedwater cracking, but those cracks have been on the secondary side. It was also pointed out to the panel that everyone has access to the SLAP database that Bill used for his analysis. It is now on the ftp site. Rob encouraged everyone to use it as part of their elicitation exercises. The SLAP database is current up to the end of 1998.

A question was raised about the validity of the single IGSCC failure reported in the carbon steel feedwater system. Bill indicated that this was the reported database value. Karen Gott said they've seen such cracking in Sweden as well.

The BWR recirculation system provided a unique problem for the base case analysis. From a materials standpoint, the base case was the old system. Bill segregated the data by old pipe (Type 304 stainless) versus new pipe (Type 316 nuclear grade [NG]). For the old pipe (Type 304 stainless) there were 127 events in 550 years of operating experience versus 3 events in 410 years of operating experience for the new pipe (Type 316NG).

The resultant leak/crack frequency for the old pipe (127 events/550 years = 0.231 events per year) is about a factor of 2 greater than the overall leak/crack frequency for the overall recirculation system history (old plus new), i.e., 130 events/960 years = 0.135 events per year. It was pointed out though that this improvement may be more due to other factors than pipe replacement only. The improvement could also be due to changes in water chemistry, or the installation of weld overlay repairs. Hence, it may be more appropriate to refer to the pipe systems as mitigated (new) or unmitigated (old) pipe systems.

It was stated that the base case is unrealistic in that it is for the old pipe case (Type 304 stainless) and no one uses that material anymore. Also, the base case does not account for the incorporation of water chemistry improvements which all plants have already implemented.

As part of his analysis, Bill made an assumption that the LOCA categories/sizes (e.g., 380, 5,700, 19,000, lpm, etc. [100, 1500, 5000 gpm, etc.]) are related on a logarithmic sense (1, 0.3, 0.1, 0.03, 0.01, 0.003). Half likelihood on logarithmic sense realizing that larger LOCAs are a subset of Category 1 (380 lpm [100 gpm]) LOCAs.

The 40 welds for the PWR-1 case (hot leg) include the cold leg and cross over leg welds. This is inconsistent with the assumption stated above that the PWR-1 case only considers the hot leg (not the cold leg or cross over leg). (Note, there are typically 3 loops in a PWR plant and there can be 5 to 7 welds per loop, but the loading is not the same for all these welds.)

Rob Tregoning indicated that the correlation between pipe size and LOCA size (gpm) that were originally supplied are subject to change.

Bill's aging correction factor is for thermal fatigue and should not be used for the other failure mechanisms.

The non-pipe LOCAs that Bill included are for passive systems only (bolted flanges, etc.). He did not include active system contributions, e.g., stuck open valves, to the non-pipe break frequencies.

Bill's results for the "Current Estimate" are significantly smaller than the LOCA frequencies reported by others (WASH-1400, NUREG-1150, NUREG/CR-5750) due to the larger database (more years of service experience without a LOCA). The best agreement is with the NUREG/CR-5750 results. Bill employed a Bayesian approach as part of his LOCA frequency analysis by assuming a half of failure for these very low occurrence events. This is a very common data analysis practice. Since Bill's analysis is based on the analysis of past service history data, the data implicitly include ISI and other mitigation experience implemented by industry.

Each plant has a PRA which includes LOCA frequency estimates for the plant. These LOCA frequency estimates are often based on WASH-1400 or NUREG-1150 and the ranges shown on pages 36 and 37 of Bill's presentation are the ranges for the Individual Plant Evaluations (IPE). Thus, the IPE range, WASH-1400 and NUREG-1150 frequencies shown by Bill on pages 36 and 37 of his handout are closely related/interlinked. The uncertainty in Bill's analysis stands for error factor (EF in slides 36 and 37) is based on an error factor of 10. This is somewhat crude and somewhat arbitrary, but the scope of the base case analysis did not ask for uncertainty, just a best estimate.

Bill indicated that he has no strong technical basis for extending his "Current Estimate" analysis out to 40 or 60 years. With Bill's approach, the LOCA frequencies will go down as more years of operating experience are accumulated, unless a LOCA event occurs in the future. Bill presented information predicting 8 years to double the frequencies for thermal-fatigue events. This estimate conservatively assumes no industry-wide mitigation programs. Rob Tregoning indicated that he did not include results for Bill in the summary table for 40 and 60 years of operation because they are most appropriate for current estimates.

Bruce Bishop was very uncomfortable predicting the future out to 40 or 60 years. He felt that the uncertainties are going to increase dramatically. Lee Abramson (USNRC) responded that this is a natural experience and likely shared by others on the panel. However, it's incumbent that each member to attempt these predictions. Rob and Lee stressed again that the panel will not be forced to answer any questions that they are very uncomfortable with. Rob further stressed that the March 2003 SRM (staff requirements memorandum) specified that the NRC staff needs to revisit the LOCA frequency estimates every 10 years and the effort will be most concerned with the next 10 years. However, it is still important to gain longer-term insights.

A question was raised as to whether or not to have Bill extend his analysis out to 40 and 60 years? Gery Wilkowski said, yes, he wanted to see Bill's assumptions. Fred Simonen (PNNL) would like to see more partitioning by pipe size as part of Bill's analysis. Dave Harris (Engineering Mechanics Technology) concurred. Sam Ranganath (formerly of General Electric) specifically indicated that it would be helpful for the BWR recirculation system because they replaced the smaller diameter pipes, but not the 28-inch diameter pipes. Bengt Lydell is to check on the statistics in his database to see if any of the 28-inch diameter BWR recirculation lines had leaks and provide this information in his final base case report. Pete Riccardella and Sam Ranganath were unaware of any General Electric large diameter recirculation pipes that leaked.

Report No. 2 by Base Case Team to the Expert Panel on LOCA Frequency Distributions by Bengt Lydell (Erin Engineering and Research)

Bengt, like Bill Galyean, used service history data in his analysis, but in a much different manner. Whereas Bill Galyean followed a "Top Down" approach, Bengt followed a "Bottoms Up" approach.

Bengt's presentation assumed that all BWR welds were category D & E welds. For BWRs, Bengt indicated that Category D & E welds specify inspection criteria based on Generic Letter 88-01. Category D welds have been subjected to weld overlay repairs and Category E welds are subject to IGSCC.

Bengt's used a different database than Bill Galyean did is his analysis. Bengt's database is proprietary (PIPEex) and the panel will not have access to this during the elicitation. The SLAP database that Bill used is available to the panel members on the website. The cut-off date for PIPEex events is the end of 2002 while the cut-off date for the SLAP database is the end of 1998. PIPEex has about twice the number of data entries as does SLAP and includes international experience.

Bengt only looked at welds in his analysis. He did not consider base metals. His database did not show any occurrences of base metal indications. However, he invoked a wide definition for what was encompassed by the term "weld". He included the heat-affected-zone (HAZ) and counter bore region into his definition of what a weld was. Bengt also did not consider degradation of non-piping passive components in his analysis.

For PWR systems, he assumed that the V.C. Summer and Ringhals cracks were circumferentially oriented cracks, not axially oriented. Whereas Bill did a "top down" analysis, Bengt did a "Bottoms up" analysis. The failure rates were derived for individual welds, and then an integration system level model was formed by combining the contributions from each individual weld failure to an overall pipe system failure frequency.

The term "prior" has very specific meaning in this analysis. It means "before ***mitigation*/remedial action** in response to a significant pipe failure". Hence, failure rates that are input to LOCA frequency calculations explicitly account for reliability improvements (mitigation methods) made in response to past pipe degradation histories. Failure in Bengt's analysis is defined as a "through-wall flaw resulting in leakage". Bengt did not include surface cracks found during ISI in his data reduction process.

Karen Gott has a report in Swedish that may be valuable in this effort. This report gives the number of leaks and the number of ISI detected surface cracks. Gery Wilkowski thought that was important information since leaking through-wall cracks are readily detected, but the surface cracks that would grow to be long in length are more of a LOCA threat. The number of records shown on slide 10 from Bengt's presentation includes both leaks and cracks.

There was much discussion among the group in an effort to understand slides 11 through 14. In slide 13, Bengt did not eliminate welds if mitigation was performed prior to 15 years of operation in developing his prior distribution. There were no leaks in BWRs after 15 years. From years 10 to 15, it is possible that there may have been some plants that used mitigation, but those mitigated plant weld numbers were still included in the weld failure rate analysis, i.e., that may account for why the weld failure rate was not accelerating as the number of years increase. Bengt used a Monte Carlo simulation to create this plot, where he needed to estimate the number of welds for the number of plants that were at a certain age. Results in slide 14 include results from US, Spanish, Swedish, and Japanese plants. The results were adjusted by the number of welds and type of welds.

These preliminary results are used to determine the "prior" LOCA frequencies (weld failure rate). The next step is to determine the "posterior" frequencies based on "prior" distributions. Bengt accounts for uncertainty in the knowledge base, which is fundamental difference between his analysis and Bill's. Dave Harris thought the "posterior" frequencies should be equal to the Prior LOCA frequencies times the Likelihood Function. Lee Abramson explained that the likelihood function was built in.

It is noted in slide 19 that the Bayesian update strategies are different for each base case. It was also noted again that the weld failures represent leaks only and not cracks. Weld failure was defined as a through-wall flaw with leakage less than or equal to the tech spec limit for undefined leakage.

Gery Wilkowski noted that in January 2003, PWSCC was found in a surge line bimetallic weld at the pressurizer in a Belgium plant. It was noted that some transient event is needed to cause a tech spec limit flaw to propagate to a higher category (well beyond 100 gpm) LOCA. Slide 23 shows that the conditional probability of failure ($P_{L/F}$) is a function of the nominal pipe diameter (DN), i.e., $P_{L/F} = a \times DN^b$, much like what is in NUREG/CR-5750 (Beliczey and Schulz, i.e., $P_{R/TWC} = 2.5/DN$). While the exact formulations are different, the conditional failure probability in all cases is an inverse function of pipe size.

The form of the conditional leak probability given a failure is inconsistent with the original development of the Beliczey and Schulz correlation which was developed to relate leaks to breaks as a function of pipe size. This use (see slide 24) may not be physically realistic because it assumes that larger diameter pipes are less likely to result in a category 0 leak. The implication is that larger diameter pipes are less likely to reach a Category 0 leak, and then progress to higher leak rates. Slide 26 presents information on the aspect ratios of IGSCC cracks. It was asked how the deep, full circumference cracks ($a/t = 0.5$, $\theta/\pi = 1.0$) formed. The expectation is that these are likely crevice cracks. It was suggested that it would be nice to break down this data by pipe size in order to assess the relevance.

Bengt assumes that a through-wall crack can only propagate into a large leak if there is a large transient event. Slide 27 documents the loading categories assumed to drive the crack among various LOCA categories. Category 0 to Category 1 LOCA progression can occur assuming moderate loading, while to go from a Category 0 or Category 1 LOCA to a Category 6 LOCA, would require an extreme loading transient. The general consensus of the panel members was that this was a very subjective approach. It was noted that there were about 400 water hammer events reported, but Bruce Bishop and Guy DeBoo said that if there were this many water hammers, then the plant piping system was probably redesigned.

The extrapolation of results from the "Current Estimate" to 40 and 60 years is based on posterior analysis of prior results assuming no additional failures. If one assumes no additional failures, the failure rates will go down with time. Bengt will examine possibly extrapolating his base case results out to 60 years using another assumption.

Report No. 3 by Base Case Team to the Expert Panel on LOCA Frequency Distributions by Dave Harris (Engineering Mechanics Technology)

Dave prefaced his comments with the thought that he thinks that the base case results are surprisingly close considering the differences in the approaches. Dave indicated that an important input to any PFM analysis is the stress history. This requirement is contrary to the service history approaches where the stress history is reflected in the operating experience. Dave's analysis is performed on individual pipe locations which are then integrated to determine the overall system frequency.

Some key points from the crack initiation and crack growth portion of Dave's presentation are that piping failures occur due to the initiation and growth of cracks. Cracks initiate due to stress corrosion or fatigue. Growth is controlled by fracture mechanics (other than early SCC). The question was asked as to what assumption Dave used as to the size of the crack once it initiates. Dave's PRAISE code assumes that fatigue cracks are 7.6 mm (0.3 inch) deep (per a criteria proposed by Argonne National Laboratories [ANL]). This is based on an assumed 25 percent load drop definition of crack initiation from an S-N specimen test. In addition, in PRAISE the SCC rules differentiate between early SCC growth and fracture mechanics growth since the early growth is faster than calculated by fracture mechanics analysis. The

SCC rules in PRAISE assume a 0.025 mm (0.001 inch) deep surface crack with some distribution function on length. The latest version of PRAISE (2002) includes updates to the S-N curves that incorporate environmental effects. Pete Riccardella noted that Art Deardorff of SAI was doing an update of some of the environmental S-N results from EPRI. For crack growth, the focus of PRAISE is semi-elliptical part-through surface cracks. PRAISE considers crack growth in both the depth and length directions (K at both the maximum depth and at the ends of crack.)

Dave pointed out that fatigue failure of welds is dominated by growth from pre-existing crack-like fabrication defects. Thus, the flaw distribution of initial fabrication defects is an important parameter to define. PRAISE stipulates that the final failure is controlled by tearing instability. PRAISE treats the stresses at maximum load as load-controlled stresses from stability analysis perspective.

The leak rate in PRAISE is computed based on the length of the leaking through-wall crack on the inside pipe surface using the SQUIRT leak-rate code. The SQUIRT code has recently been updated with numerous technical enhancements as part of the USNRC Large Break LOCA program. PRAISE includes some of the mechanistic dependent crack morphology parameters from some of the earlier versions of SQUIRT, but not the new COD-dependent roughness, number of turns, and flow path length to thickness ratio parameters. In addition SQUIRT has been made more user friendly by incorporating a graphical user interface. Note, WinPRAISE is PC-PRAISE with a Windows pre-processor for entering input parameters.

Dave used a stratified sampling technique that allows for the evaluation of extremely small probabilities events such as the case of fatigue crack growth from pre-existing defects (10E-17 frequencies). The approach also assumes that all cracks with leak rates greater than 19 lpm (5 gpm) are discovered and subsequently removed from service which implies that getting a higher Category LOCA (e.g., a 5,700 lpm (1,500 gpm) LOCA) would most likely result from the growth of a long surface crack that pops through the wall thickness and immediately becomes a TWC with a length equal to length of the surface crack on inside pipe surface. (The only other means of achieving a higher Category LOCA would be through some sort of transient event.)

Dave postulates that inaccuracies in leak-rate calculations will not significantly impact final LOCA frequencies. Even assuming the leakage detection capability equals zero should not have a large effect. Sam Ranganath felt that a surface crack grows 3 or 4 times faster in the length direction than it does in depth. He wasn't sure if that was due to multiple initiation sites, or the surface growth rate being higher. The analysis in PRAISE considers crack growth in both the depth and length directions (K at both depth and ends of the crack) with an RMS value of K in each direction.

The detection probabilities shown in slide 14 are based on depth only, not length. It is likely that current technology has better performance. It was also noted the fatigue crack growth parameter used could be improved based on newer results. Dave Harris indicated that sensitivity studies show that using improved crack growth parameters in PRAISE (e.g., including environmental effects) will result in changes in LOCA frequencies on the order of a factor of 2. Slide 16 presents an example S-N initiation curve for a low alloy steel. Vic Chapman raised the concern that there may not be a plateau or fatigue limit with the higher number of cycles. Gery Wilkowski commented that the default flow stress values (slides 17 and 18) are very tight from a standard deviation perspective, especially in light of what was seen in NUREG/CR-6004 from an analysis of PIFRAC data.

The NUREG-6674 stresses have been downgraded from the design basis stresses by Jack Ware of INEEL to make them more realistic with fewer transients. The stresses may have been elastically calculated values, which can go well above yield and still be allowed for secondary stresses by the Code. PRAISE uses the most realistic values available. The final results (frequencies) from Dave's analyses are very dependent on stress input values. Dave commented that one only needs stresses at the high stress locations. These high stress locations dominate the final frequency answers.

Slide 21 shows the surge line stresses. These stresses are probably for the flank of the elbow, not the weld. These values can't be used for the girth weld location (i.e., they are too high). The stresses shown are stress amplitudes (the stress ranges will be twice these values). In addition to stresses and number of occurrences, one also needs some input as to the spatial distribution for these stresses. One of the short comings of PRAISE is that PRAISE doesn't have a model for FAC for the feedwater lines. The fatigue initiation models are only in latest versions of PC-PRAISE, i.e., from NUREG/CR-6674 published in June 2000.

An ad hoc procedure was used with pc-PRAISE in order to obtain results for larger leak rates (stratified sampling for fatigue crack growth is not available for fatigue crack initiation). One of the handouts provided shows this ad hoc procedure.

A question arose with slide 26 concerns the fact that Dave's analysis shows that the cumulative failure probabilities continue to increase after 20 years after a weld overlay repair is applied at 20 years. Past experience at Battelle as part of the Degraded Piping Program showed that weld overlays are very effective. They have much higher strength than the base metal. Another aspect of their application is that they apply a very high compressive stress at the crack plane. These high compressive stresses should restrict any further crack growth of the surface crack. In addition, it was noted that these high compressive stresses may preclude the environment from getting to the crack tip. Dave noted that he put in a linear approximation of the stresses through the thickness, the increased thickness of the overlay, and the crack growth equations for Type 316 NG into his analysis. WIN-PRAISE uses an adjusted weld residual stress pattern (linear gradient) for the case of post-weld overlay residual stresses (see slide 38). Dave will also present the failure probabilities without the weld overlay so that an assessment of its effect can made.

Bruce Bishop asked how much was Dave's results affected by inspection. Dave didn't think that the final frequencies would be affected that much. Dave's results showed that there was a minimal change in LOCA frequencies for the hot leg as a result of the application of a 5SSE earthquake. This was not surprising to Dave since he has found similar behavior in a previous study. Lee Abramson pointed out that one means of seeing the effect of the earthquake is to compare conditions for equal probabilities. For example, for the 40 year time period analysis, no earthquake results in 1.3E-18 LOCA frequency for the no leak case while for the same 40 year time period analysis, a 5SSE earthquake results in 1.3E-18 frequency for a double-ended guillotine break (DEGB).

Dave, generally found that the LOCA frequencies were not highly dependent on J_{Ic} and dJ/da. Gery Wilkowski thought that if the toughness was low enough that one was operating in the EPFM regime then the LOCA frequencies may be more dependent on toughness. Gery indicated that he thought that the toughness values used in Dave's base cases were too high for weld crack locations, or aged cast stainless steel pipe and fittings.

PRAISE can't account for time dependent material properties. Thus, to account for aging, one would need to input aged properties in at time equal to zero.

The very low LOCA frequencies for the hot leg in slide 31 may be an artifact of the failure mechanism (fatigue) chosen for analysis. Higher frequencies may be seen for some other mechanism, such as PWSCC. This is a case that we may want to analyze in future analyses.

Report No. 4 by Base Case Team to the Expert Panel on LOCA Frequency Distributions by Vic Chapman and Chris Bell

The first part of the presentation was presented by Chris Bell (Rolls Royce) with the final few slides presented by Vic Chapman. Chris presented the general assumptions and methodology of the PRODIGAL code. Note that PRODIGAL actually has several modules. One of them is to determine weld defect size from welding information. Another is to determine failure probabilities for navy nuclear power plants. All results presented are for a per weld basis.

In PRODIGAL, surface imperfections with a depth of 0.1 mm (0.004 inch) are assumed, but there are no SCC initiation/growth models in PRODIGAL. Past study has shown that there is little sensitivity to the assumed depth and sizes much less than 0.1 mm (0.004 inch) have little effect. For surge line case, there is sensitivity to the existence of 1 mm ((0.04 inch) defects, see slide 19.

Based on some work of Ritchie, Pete Riccardella felt that the 0.1 mm (0.004 inch) initial defect size was near the lower bound of the region where fatigue crack growth (da/dN) models were valid. It was noted that Omesh Chopra from Argonne thinks that this lower limit is closer to 0.5 mm (0.02 inch). It was suggested that this limit is dependent on the grain size of the material. Note, cast stainless steels can have very large grain sizes.

Vic and Chris felt that they could not do a generic analysis for seismic considerations since the effect of seismic has been found to be highly dependent on plant layout. Therefore they didn't consider seismic stresses in their analysis. In addition, they only considered the three PWR base cases since they had little experience with BWRs.

They used the same stress data as Dave Harris did for consistency purposes. As Dave did, they used a second order distribution of stresses thru the thickness per NUREG/CR-5505 criteria that was done by PNNL in 1998.

The failure criterion for their instability analysis is based on the FAD approach in R6 using K_{IC}, i.e., crack initiation would equal failure. The crack initiation used for stainless steel was closer to wrought base metal rather than weld metal which would be an order of magnitude lower, or aged cast stainless steel which could be another factor of 3 lower than the weld metal toughness.

Bruce Bishop asked what the mean temperature in the analysis is used for. It is used for material property considerations, such as flow stress, but not for subcritical crack growth. Slide 16 from Vic and Chris' presentation shows the cumulative probability of a TWC (probability of a leak occurring). The implication from this slide is that if you are going to have a leak, it will occur in the first 25 years of operations. It was noted that slide 16 is conditional on having a crack (probability of having a crack is 1).

Rob Tregoning indicated that he wants each of the panel members in the next week to make list of what they want to see (e.g. data they used in doing their analysis) from either individual participants or from the group as a whole.

It was noted that the dominant hot leg cycles are those due to heat up and cool down. There are only on the order of 5 of these cycles per year.

Pete Riccardella thought that there were a lot of cycles on the PWR HPI/Make up nozzle each year. Dave Harris countered that he thought that there were only about 40 of these cycles in 40 years of operations. Bengt Lydell agreed with Pete and thought there were a lot of thermal cycles. Pete thought that something was missing here. Bengt said there was a nice ASME paper on the cyclic stress history of these nozzles that we could use in the analysis.

Chris indicated that although they typically keep the aspect ratio constant in their PRODIGAL runs, they have the ability to grow cracks in length, and often get very irregular crack shapes.

In slide 24, the "separation" referred to is the crack-opening displacement. In this slide, at the surge line elbow, Vic speculates that the crack starts to act as a hinge so that crack opening becomes very large for the longer crack lengths. This assertion is the basis for slide 25 illustration of the crack frequency versus angle distribution that was assumed in the analysis. This relationship was created by Vic with guidance from metallurgist to rectify predicted and experimental crack lengths.

It was commented that for the same input, we are seeing similar results from PRAISE (Dave Harris' results) and PRODIGAL (Vic Chapman and Chris Bell's results). It was noted that as far as this exercise is concerned, PRODIGAL's strength is the defect distribution analysis. PRODIGAL only looks at thermal fatigue while PRAISE can look at other failure mechanisms. PRAISE can also look at crack initiation while PRODIGAL cannot. However, neither accounts for FAC or PWSCC at this time. PRAISE can also look at crack initiation whereas PRODIGAL assumes crack growth from weld defects or surface imperfections.

Vic pointed out that embedded cracks can straddle the compressive zone of the residual stress field through the thickness so that once they break thru to inside surface they are already through the compressive zone. This is in contrast to case where a crack is growing through the wall thickness from the inside pipe surface and the crack gets trapped in the compressive zone near mid thickness. There was also the question of whether embedded cracks are affected by the environment.

The next topic for discussion was to plan the next step for the base case calculations.

Bruce Bishop would like to bench mark the PFM results against the 25 service history data developed by Bill Galyean and Bengt Lydell, and then use the PFM models (PRAISE and PRODIGAL) to predict the 40 and 60 year results. Rob Tregoning thought that this was an excellent idea. Rob also indicated that we could do this for some cases, but not all cases, e.g., FAC in the feedwater system or PWSCC.

Sam Ranganath would like to know when we compare results where do we get good agreement and where not. Rob indicated that we haven't made comparisons on a consistent basis as of this date but that this would be rectified.

The next issue focused on additional stresses to consider in the PFM results. The surge line and HPI/MU cases were mentioned. Dave Harris has already done some additional analysis for the surge line, based on refined stresses developed by Art Deardorff. Gery Wilkowski noted that he had surge line stresses from a Westinghouse Owner's Group report used for LBB analyses. Pete agreed to verify that the previous stresses provided by Art are appropriate. Pete also agreed to provide more accurate HPI/MU stresses.

Sam Ranganath would like to lower the stresses to 70 MPa (10 ksi) for the recirculation line (BWR-1) and to lower the feedwater line stresses (BWR-2) by 20 percent.

The next area where the panel thought we may want to focus is some sensitivity analysis using different material properties. Gery Wilkowski volunteered to supply some distributions of material properties (mainly toughness values for welds and aged cast stainless steels) developed as part of NUREG/CR-6004.

Gery Wilkowski also wanted to see the ratio of surface crack to through-wall cracks removed from service, and the distribution of the lengths and depths of those service removed surface cracks. He then wanted to see a comparison of the PFM probability of leaks for IGSCCs from PRAISE and the distribution of surface cracks that might exist up to the time that piping might have been replaced/repaired (15 service years?). Karen suggested looking at the more recent results on a yearly basis because ISI wasn't sensitive enough to find surface defects in early years. Cracks were only discovered once they became a leaking crack. Also, in Sweden, even if whole pipe sections were removed, all the welds were inspected, whereas in the US if the pipe system was replaced they did not spend the effort to inspect welds that were being removed from service. Hence, the database of cracks removed from service should separate Swedish and US plants. The important aspect of this comparison is to see the population of the ISI remove surface crack lengths compared to the surface flaw sizes calculated by PFM. If the service removed crack lengths from ISI are much longer than calculated by the PFM analyses, then the PFM analysis should underestimate the future failure probabilities. Gery noted several times that failure probabilities should be controlled by development of long surface cracks, not the growth of leaking through-wall cracks. Bengt Lydell has some papers relating ISI-detected surface-crack geometries to through-wall crack leaks that he can provide on the ftp site and can provide to Rob Tregoning.

It was suggested to include PWSCC in the hot leg base case analysis. Dave Harris has initially done this using IGSCC relationships for preexisting flaws. Initiation is not accounted for. Gery Wilkowski noted that the crack growth rate through the weld metal (along the dendritic grains) is much faster than IGSCC. Gery will work with Karen Gott and Bill Cullen to see how Dave can adjust the PRAISE model to get initiation and growth for PWSCC. Gery will provide the IGSCC initiation and growth equations that PRAISE uses to Bill Cullen so that appropriate constants can be provided for PWSCC in PRAISE for base case calculations. Gery and Karen will provide information to Dave Harris on PWSCC crack initiation and growth models that he can use to evaluate the impact of PWSCC on the appropriate base case calculations.

The panel thought it was important to address Dave Harris' strange results for weld overlay repairs. What is leading to high growth rates after the overlay is applied. Can a comparison be shown with what would happen if no weld overlay had been applied? Dave Harris is to address this concern of the unexpectedly high growth rates after the weld overlay repair is applied.

When the summary comparison tables are completed, the PFM subgroup needs to clearly identify where the PFM conditions do not agree with service history. Rob Tregoning indicated that the base case subgroup needs to finish up the base case calculations by the end of the month. We cannot delay individual elicitations any longer.

Day 2 (June 5) – Elicitation Coordination

Rob Tregoning started the morning by reviewing the agenda for the second day. There were six (6) basic items to cover. These were:

- Reviewing the elicitation questions
- Reviewing the leak rate versus pipe break size evaluation
- Addressing the non-piping LOCA evaluations
- Reviewing the conditional seismic evaluation
- Addressing the additional information required prior to the individual elicitations for piping and non-piping evaluations
- Elicitation scheduling

The specific objectives for the second day included:

- Finalize the elicitation question sets and to provide a consistent understanding of each question.
- Determine the methodology for evaluating non-piping LOCAs and the identification of non-piping base case data.
- Determine the methodology for evaluating conditional seismic loading including determination of the seismic loading magnitude.
- Determine what information the experts will require prior to their elicitations and assign action items for providing information.
- Develop the final schedule and time-frame for the upcoming elicitations.

Presentation 1 (Day 2) – Elicitation Questions: Structure and Review by Rob Tregoning (USNRC)

Rob stressed that prior to their individual elicitations, each panel member needs to do their homework and answer as many of the elicitation questions as possible. Panel members can change their answers at any time during this exercise, including during the actual elicitation and afterwards.

It was indicated that the term "LOCA frequencies" should really be "LOCA probabilities" in this presentation during the analysis of the conditional emergency loading. Rob agreed to make this change to the presentation.

Pete Riccardella questioned whether we should expand the conditional seismic loads to conditional emergency and faulted loads which include transients like water hammer as well as seismic. Fred Simonen thought that we needed to talk with the PRA people. Alan Kuritsky (USNRC) indicated that these other potential LOCA causing events were currently not included in the PRAs. Rob suggested tabling this discussion until later in the agenda. Bruce Bishop thought that if we continued with the traditional seismic approach then we need to consider the fact that the snubbers may not work probably. Bruce indicated that there is a significant probability that the snubbers may not work as advertised.

Rob next addressed the top down elicitation structure. Bill Galyean used a top down approach where he assigned an overall piping LOCA contribution, and then looked at the breakdown in the contribution due to piping system, geometry, load history, mitigation, materials, and degradation mechanisms. Vic Chapman questioned what the top down approach gave us (we start with the final answer that we are looking for). Lee Abramson indicated that at the end of the day we will get numbers in each of the blocks on slide 3 so that we can decompose the problem into the small pieces. The panel members can initially choose either a top down or bottoms up approach.

Bruce Bishop asked how soon the panel members are going to have the final base case results prior to the first elicitation. Rob indicated that he would be working with the base case members the week of June 9 to finalize their answers. He hoped to have the final results by the end of June. However, the base case conditions are well-known

The panel members need to supply their answers (on a pre-established form) and the facilitation team will work with the individual panel members individually. The important point of the pre-elicitation exercise is to quantify the median value results, provide some qualitative uncertainty and also rationale. During the elicitations the panel members can change answers as they interact with the facilitation team and points are clarified.

Bruce Bishop felt that the "utility safety cultural" should be "utility operations cultural" in that safety and economic drivers both feed into the operations cultural. Karen Gott indicated that the IAEA definition of safety cultural (and how we defined safety cultural at the kick off meeting) includes both safety and economic aspects. With regards to the ratios on safety cultural issues, ratios greater than 1.0 indicate that things are getting worse; less than 1.0 means things are getting better.

Slide 6 shows a flow chart in which the question is which variables are independent. Variables in this context are geometry, load history, mitigation, materials, and degradation mechanisms. For these variables, each expert may need to estimate the future impact of that variable, e.g., what new materials, or what new degradation mechanisms, should be expected in the future. We may want to look at the past to estimate what might happen in the future (e.g., Gery Wilkowski's plot of new failure mechanisms with time which shows a new mechanism approximately every 7 years).

There ensued a long discussion on the comparisons of reference cases (defined at the kick off meeting) with baseline cases. We defined a reference case for each piping system (e.g., diameter, material, degradation mechanism, etc) whereas we only defined a few baseline cases to which the reference bases are to be compared (i.e., anchored). For example, the hot leg (base case) may be a natural comparison for the reference case for the cold leg.

For Questions 3A.1 and 3A.2 (slide 7) and all related questions, Rob will change the "surge line" to "cold leg" example so questions don't refer to a system that is both a base case and a reference case, realizing that we will end up asking same question for a surge line as well. For those systems which have both a base case and a reference case, these comparisons may be more natural and easier than inter-system comparisons.

It was decided to eliminate the assessment of which variables are independent or dependent. All variables will be considered to be dependent as originally defined during the kick-off meeting. Thus, the original Question 3A.2 will be eliminated. Other References to correlated or independent variables in other questions should be eliminated in the final presentation version. For original Question 3A.3, the requirement to list at least 80 percent gets the most significant contributions, but not all of them.

There was considerable discussion on what the panel members would need to provide for Question 3A.3 and 3A.4. We need to know what variables have a major impact on LOCA frequencies to quantitative that impact as best as possible. Lee Abramson tried to make the point that things would become clearer once individual panel members got into their elicitations and tried to put numbers to the answers to the questions.

A question was asked if any attempt is going to be made to look at plants and determine how many plants have a certain combination of variables (V1, V2, V3, etc.), and how many plants have another set of variables (V2, V4, V5, etc.), and how many have another set. It was noted that some variables will be important at certain plants and other variables will be important at other plants. Rob indicated that it would be nice to have such information, but it was not practical to get such information in the time frame we have. This could possibly be a follow-on effort. However, it should be stressed that the plant design information will not likely result in a significant change in the analysis. If certain designs do not contribute to the LOCA frequencies then they are not significant contributors and the experts can focus on designs that do as long as they exist in several plants. If the population of the significant contributors is in error by 2 to 3, it will likely not matter.

The only issue to avoid during quantification is if you believe that only a few (1 – 2) plants of a certain design, operating experience, etc. significantly contribute to the generic LOCA frequencies. These plants should not be explicitly considered in these generic estimates. However, but possibly applicability of these generic results to those design conditions should be discussed during the elicitation.

Elicitation Question (EQ) 3A.1 compares a base case to a reference case, then EQ 3A.4 will use the impact of the important variables to compare other similar piping systems to the reference cases. Reference cases are the link back to the base case for which we will have actual LOCA frequency estimates. The panel members don't have to do the mapping back to the base cases, the facilitation team will do that. Then the facilitation team will filter the results up (bottoms up approach) to get an overall LOCA frequency.

There was some discussion about how the facilitation team would integrate these results and how the experts could account for these individual contributions.

Again, it was emphasized that if panel members are not comfortable in answering specific questions, then they need to say so. If the panel members need to make a crude assumption, then do so, but indicate that during the individual elicitations so the facilitation team can help estimate the level of uncertainty. Dave Harris thought that it won't be clear to him how this all fits together until he starts the process. Then he is sure that he will have a lot of questions. Rob Tregoning and Lee Abramson told him, and the rest of the panel members, to call them for any clarification of questions. Rob also indicated that he will be contacting each expert prior to their elicitation to discuss issues.

The flow chart on slide 9 is for a "top down" approach, much in the motif of what Bill Galyean discussed on Day 1. As part of this approach, one only needs to tie one system (of those identified as being important contributing systems to LOCA frequencies) to the base case since previously we had identified individual piping system contributions. One can make this connection through a reference case if not a base case system, or can tie directly to the base case if the system is a base case system. This approach may be more straightforward for people who need to integrate variables in mind, which might lead to more uncertainty.

Panel members can chose which approach to follow (top down or bottoms up), and they can switch back and forth depending on different systems. As part of their homework prior to their elicitation, they only need to do one approach, but the facilitation team may ask about each approach during the elicitation. There will be different tables to fill out with each approach. The bottoms up approach may be more rigorous with less subjectivity, but the top down approach may be easier to understand. One can get top down answers from the bottoms up approach, but can't do reverse. By doing both approaches for certain systems, experts can search for consistency in their analysis.

Rob discussed the elicitation questions related to non-piping components starting at slide 12. At the kick-off meeting in February we didn't spend as much effort developing the base case and reference cases for non-piping components as we did for the piping systems. Thus, these gaps needed to be filled during the remainder of this meeting.

Slide 12 illustrates the flow chart for the "bottoms up" approach for the non-piping components. At the kick-off meeting, we had identified five (5) non-piping components to consider (pressurizer, valves, pumps, reactor pressure vessels, and steam generators). In order to estimate the frequencies for these non-piping components, the panel members need to pick either a piping or non-piping base case for comparison. The facilitation team will integrate the results in a manner similar to the bottoms up approach for piping systems.

There was a question about the nature of the non-piping base case conditions, especially in light of the thorough discussion about the piping base cases during the previous day. We had originally planned to have precursor data for the non-piping base cases for comparison. However, we do not have data identified yet. We may have to drop the idea of using non-piping base cases and only have piping base cases to compare to. We will come back to this issue later in the afternoon.

Bruce Bishop felt that tying non-piping components back to piping base cases would be difficult. He foresaw lots of dissimilarities between non-piping and piping in failure mechanisms, etc. He suggested that we try our best to come up with some non-piping base cases. Even if we can't come up with base cases for all 5 of the components, if we could come up with base cases for a few, that would be better than nothing.

Pete Riccardella asked if we have defined the failure mode for these non-piping components. Rob felt that we don't have a clear definition at this time. Rob felt that we had to take more of a mechanistic viewpoint. There was also general confusion about the definition of "failure mode" for the non-piping issues. Rob indicated that this means the failure mechanism. It was agreed that "failure mechanism" is a preferable term and Rob will change the phrasing for the elicitation questions from "failure mode" to "failure mechanism" to avoid any confusion.

It was again emphasized that the panel members will not be asked to provide absolute LOCA frequencies. However, if a panel member prefers to think in terms of frequencies, they should feel free to do so. Elicitation questions will ask for relative comparisons with the base cases and other conditions. The facilitation team will then make the calculations to get the absolute LOCA frequencies. The panel members should make the best comparisons possible and are not compelled to answer questions in areas where they have no expertise.

Rob showed an example of a table that he may provide the panel members for them to fill out for elicitation questions (EQ) 3B.1 and 3B.2 for the top down approach, see Table B.2.2.

The complete set of tables to be filled out for the elicitation will be provided electronically by Rob. There will be a space for comments in each table row to initiate discussion during the elicitation process. The tables will be provided in Excel format. If a panel member wants to change the Excel spreadsheet format they should feel free to do so as long as the cell references for each answer remains unchanged. The final calculations will be done in Excel. Therefore, the elicitation results should be provided to Rob in the excel spreadsheets if at all possible. Hardcopies or MS Word versions of the tables can provide upon request.

Rob will provide the panel members with a copy of the spreadsheet that he will use to calculate LOCA frequencies sometime during the elicitation process. Rob will attempt to complete this spreadsheet to the individual elicitations so that the panel members can see how their responses reflect their calculated LOCA frequencies. However, this will be a lower priority than coordinating the information exchange among the expert panel and finishing the base case calculations.

Table B.2.2 Elicitation Questions 3B.1 & 3B.2
BWR Piping Systems: Important System Contributions to LOCAs

	25 Years of Plant Operation				40 Years of Plant Operation				60 Years of Plant Operation			
LOCA Cat.	Systems	System Cont.	5% LB	5% UB	Systems	System Cont.	5% LB	5% UB	Systems	System Cont.	5% LB	5% UB
1												
	Total				Total				Total			
2												
	Total				Total				Total			
3												
	Total				Total				Total			
4												
	Total				Total				Total			
5												
	Total				Total				Total			
6												
	Total				Total				Total			

Presentation on Non-Piping LOCA Evaluation: Base Case Data and Remaining Issues by Rob Tregoning (USNRC)

Prior to this presentation, Rob noted that this presentation was not included in the handout, but will be put on the ftp site. Rob also asked that the base case team provide him in an electronic format with any references that they used so that the references can be put on the ftp site.

It was first noted that we have not been successful in locating additional failure data for several of the components where we were lacking data. Fred Simonen had spoken with Spencer Bush and was not able to locate additional data. Rob and others were also somewhat unsuccessful.

Based on the leak rate versus opening area from one of the prior presentations, Pete Riccardella questioned if we needed multiple failures for the heater sleeves as shown in slide 3 in Rob's presentation. Bruce Bishop indicated that thermal fatigue needed to be added to the degradation mechanisms for vessel head bolts (slide 4). Rob indicated that these tables are not filled in completely at the present time. These tables are neither comprehensive nor as complete as the piping component table. We focused more on the piping issues at the kick off meeting than we did on the non-piping issues.

For all of the major groups, we initially listed at least one component (**bolded item** in the original non-piping tables in presentation and kick-off meeting notes document) where the panel thought that failure (leaks or cracks) data is available. It was thought that these **bolded items** represented potential non-piping base cases for anchoring. If these non-piping base cases can't be develop, then we will have to use a piping base case for anchoring. As Bruce Bishop indicated earlier that would be an unnatural comparison, making it somewhat difficult. The question was asked how we develop these non-piping base cases. It was thought that areas where we could come up with service history data, e.g., steam generator tubes, would be logical first choices. It was thought that data on steam generator tube failures should be easy developed.

Bruce Bishop indicated that there was an INEEL NUREG report by Vic Shah that has piping and non-piping failure data that could possibly be used for base cases. Obviously, this would be a good place to start. It was noted though that this INEEL report is for PWRs only, and the steam generator tube service history data will be included in this report. Bill Galyean and Rob Tregoning are to locate and distribute copies of this report to the panel members.

Pete Riccardella volunteered to run a base case analysis for feedwater nozzles and the belt line region for the RPV due to low temperature over-pressurization. Pete thought he could have some analysis results by the end of the month. These would be for BWRs and will be done using predetermined flaw distributions. Note, to date there have only been cracks, there have been no leaks to date.

Another potential non-piping base case for anchoring is the pressurized thermal shock (PTS) study for the belt line region of the RPV. This would be for PWRs. Rob Tregoning will extract this data. Gery Wilkowski noted that we should use caution in using the PTS results and should only use the contribution from non-pipe break transients to ensure that the comparison is consistent. In a related action, Bruce Bishop volunteered to provide a Westinghouse nozzle study for PWRs and will also provide PWR vessel failure probability for areas outside the beltline region covered by the PTS study.

Pete Riccardella suggested classifying the alloy 600 penetrations as non-piping failures for consistency with other nozzles. A suggestion was made to move the in-cores and CRDMs to the non-piping category. Fred Simonen suggested putting them all under a separate category called vessel penetrations, with a separate bin for RPVs and pressurizers. Karen Gott and Pete Riccardella are to create a base case for penetrations using the CRDM data based on a prior MRP study. Rob Tregoning will move all the vessel penetrations from piping to the vessel bin and supply updated tables.

Bengt Lydell has a non-piping data base that he will query by end of the month. He will also query the IRS data base (Incident Reporting System by INEA) by end of the month. Note the IRS database only includes data countries chose to include. It was also noted that MITI and NUPEC (both in Japan) have a data base for non-piping components. The NRC supposedly has a copy of this database. Rob will check into relevancy and availability. Bill Galyean volunteered to examine his database to see any relevant non-piping events, i.e., stream generator tube rupture statistics, incidents of bolting connections, etc., by the end of the month. Rob will be the conduit for getting the results from the various individuals searching the databases out to the rest of the group. Results will be posted to the ftp site and more important items will be bulk emailed to the panel.

The question was asked if the panel members could take home the modified tables for the pressurizer, RPV, pumps, valves, and steam generators and fill them out and return them back to Rob within two weeks. Rob will modify these non-piping tables (Tables B.1.7 and B.1.8 from the First Elicitation

Minutes) with the additions made during the second meeting discussions and send them out to the panel members. Each panel member is to modify these non-piping tables and get them back to Rob Tregoning.

Fred Simonen has an electronic version of the GALL report (Generic Aging Lessons Learned) which identifies the key degradation mechanisms that could be used to help fill out these tables. Fred will extract the relevant tables to be used in identifying the key degradation mechanisms.

Presentation on Conditional Seismic Evaluation
By Rob Tregoning (USNRC)

Each base and reference case includes at least one transient since transients are needed to initiate a LOCA event. At this time the transients are poorly defined.

Fred Simonen has seen reports that show seismic stresses, but he has no idea of the magnitude of the non-seismic transients (e.g., water hammer, safety relief valve transients). He and Gery Wilkowski would like help in establishing a rough order of magnitude for these types of transients. This information could be best expressed as a percentage of the Service Level stresses. Pete Riccardella indicated that safety relief valve (SRV) transients could be on the order of a small earthquake, just with a higher frequency. Bruce Bishop agreed to provide some water-hammer transient stresses for the pressurizer. Gery Wilkowski volunteered to provide some summary information from past probabilistic LBB analyses

Bengt Lydell indicated that the water hammer frequency is about 5E-3. There are more water hammer events on the secondary side, but there are design basis events that can cause water hammer on the primary side. It as noted again that without a transient a large LOCA is highly unlikely. The cracks will just leak until they are detected, and then will be repaired. Long surface cracks that don't leak are drivers for large LOCAs.

Guy Deboo (Consolidated Edison) volunteered to provide stresses and frequencies for transients (e.g., feedwater line water hammer, SRV, and seismic from the LaSalle plant). Gery Wilkowski to provide some tables from NUREG/CR-6004 showing the N+SSE stresses for about 30 piping systems. This data was originally developed for the ASME Section XI Working Group on Pipe Flaw Evaluation. Additionally, Gery Wilkowski and Guy Deboo will provide stresses (not frequency) for some large faulted loads that are not really expected to occur over the life of plant. Gery is to examine the N+SSE stresses from the USNRC leak-before-break (LBB) submittal database. Guy noted that a 1SSE amplitude earthquake, based on seismic hazard curves, is expected to occur once over 40 years (design basis). The frequency (not amplitudes) of the seismic hazard curves are generally considered to be conservative. Gery noted that the design basis apparently is conservative since he is sure that an SSE event has ever occurred at any US (or other) plant. Hence, the seismic event frequency could perhaps be down graded to 0.5/2,600 events/year rather than 1/40 events per year.

For the smaller transients, Dave Harris will extract stresses from a NUREG report by Fred Simonen. Guy Deboo, Pete Riccardella, and Sam Ranganath will provide some data on normal operating transients. Pete Riccardella will get some data showing comparison of design versus actual transients based on some thermal fatigue analysis from a Sandia report.

Bruce Bishop has some plant specific ISI data that he could provide which provides transient information, but he won't be able to provide it expeditiously due to other commitments. In fact, it is unlikely he will be able to provide this during the timeframe of this effort. Pete Riccardella concluded that the actual transients were not as severe as the design basis transients, but there are typically more of them.

It was also agreed that the group should have isometric drawings for the LOCA-sensitive piping. Bengt Lydell will inventory his electronic drawing database. The purpose is to look at generic systems to get an idea of how many welds are involved, pipe sizes, etc. Guy DeBoo will also evaluate the ISO drawings in his archive. Bengt will coordinate with Guy Deboo on this effort. For the time being it was decided that we would not seek additional isometric drawings until we determine if we are missing any of the major piping systems. Guy Deboo will then help obtain drawings (e.g. ISI drawings) for missing systems that at least indicate the number of welds. In general, multiple isometrics of similar systems are useful since each plant design is somewhat unique.

Dave Harris thought we only needed census data, i.e., number of welds as a function of pipe sizes, etc. Dave will provide a census that he has developed of number of welds for the base case piping systems. Dave and Bengt Lydell will coordinate on this action. Gery Wilkowski indicated that there is a MRP report with locations and numbers of bimetal welds that may be useful to review. Gery will provide a table of Inconel weld locations in different piping systems from the MRP-44 report.

Rob Tregoning asked if the panel should consider redefining the base and reference cases. The biggest concern is that the loadings and the mitigation/maintenance may need to more accurately reflect the operating experience. The panel could redefine these variables in a way that is more consistent with the quantitative analyses that was presented for the base cases earlier. The approach is to define these base and reference cases load and mitigation variables to reflect historical plant operating experience for the first 25 years of plant life (e.g. the current LOCA estimate). The consensus opinion agreed with this proposal. Guy Deboo added that he would like to see the base cases run out to 40 and 60 years, with a seismic event included. It was again noted that the objective of the "current estimate" analyses (i.e., out to 25 years of plant life) was to provide a benchmark against historical data.

Rob also asked if the panel wanted to consider more than one degradation mechanism for each reference case since the operating experience contains contributions from all applicable degradation mechanisms. Rob Tregoning wants to make sure each panel member is making the same relative comparisons with the reference cases and that these relative comparisons are natural. However, the group consensus was that it is easier to use the reference cases for anchoring when only considering one degradation mechanism and that no other changes in the reference cases should be adopted.

Vic Chapman argued that if we run the probabilistic fracture models for the first 25 years for benchmarking purposes, and we see a failure, then we should exclude that result since in reality we have not seen any failures to date in the service history. Lee Abramson agreed with this thought. The base case team members who analyzed the service history (Bill Galyean and Bengt Lydell) already have inherently benchmarked using the service history experience. The probabilistic fracture models (David Harris and Vic Chapman) still need to benchmark their data.

Bruce Bishop would like Karen Gott, Bill Galyean, and Bengt Lydell to extract the failure mechanisms as a function of piping system from their databases. Bill will have one of his colleagues do this. Bengt and Karen will query their databases to get a list of degradation mechanisms as a function of piping system. Bruce Bishop would like someone to publish a list of plants by design type. Rob Tregoning indicated that he would provide this information.

Several people wanted Dave and Vic to run their models using refined stress histories. Pete Riccardella agreed to redefine the loads for the HPI/MU nozzle. Gery Wilkowski volunteered to get some more realistic surge line stresses for the surge line elbow case. It should actually be for a crack in the girth weld at the elbow, not a crack in the body of the elbow.

Bill Galyean and Bengt Lydell will determine the frequency of IGSCC leaks and surface cracks as a function of time and pipe size using their respective databases. The surface crack data should be characterized by the length and depth of the flaws if possible. The Swedish data should be separately characterized from the US plant data, since the US plants may not have characterized the flaw shapes by ISI if the piping was being removed from service. In Sweden, flaws from removed pipe systems have been characterized. A further requirement of this query to examine the recirculation system piping studied in the base case would also be helpful. Bengt will provide all this information in an Excel spreadsheet.

Presentation on Leak Rate versus Pipe Break Size Evaluation
By Rob Tregoning (USNRC)

Rob started by reviewing the history behind this analysis. Rob was not able to find a reference for the basis of PWR correlations developed in NUREG-1150. The BWR correlations were extracted from GE NEDO studies performed in the early 1980's. New correlations have been based on several closed-form solutions appropriate for BWR steam and liquid lines and PWRs.

Rob still needs to add a column correlating the pipe diameter to the leak rate/area in slides 8 of 9 of this presentation. It was agreed that the pipe diameter should assume a single ended guillotine break. Rob also agreed to change the word "axial" to "transverse" on slide 9 of the original presentation which relates the pipe fracture area to transverse piping displacement. Gery Wilkowski noted that once a full double-ended-guillotine break occurs that the two ends of the pipe are jets that move away from each other so that he would be hesitant to use the analysis on slide 9 which assumes a transverse displacement. The base case team members will update their results using the new pipe break size to leak rate correlations developed for this presentation.

The final topic on the agenda was a discussion on the schedule for the elicitations.

Rob Tregoning and Lee Abramson want to do 2 elicitations early, possibly the week of July 14th. Then take a month off and restart the elicitations in mid August with the goal of completing all of the elicitations by the end of September. That would leave about a month to analyze the results. Rob is looking for volunteers to be the first two individuals to go through the elicitation process. All panel members should give their schedule to Rob Tregoning so that the individual elicitations can be scheduled.

Once the elicitations are complete and the results analyzed, a wrap-up meeting will be held in the October/November timeframe. As part of this meeting the calculated LOCA frequencies will be presented and any interesting and surprising results from individual questions will also be presented. The panel will also be solicited for feedback on the process. We will try to identify strengths and weaknesses with the process. Any necessary follow-on work will be defined during this meeting. We are also open to suggestions for conducting a reanalysis of these results in ten years. It is anticipated that the wrap-up meeting will take two days.

Again, it was noted that the panel members can change the results after their individual elicitations and after the wrap-up meeting. However, the final report must be submitted by the end of December. It is too premature to speculate on the form of the final report. However, confidentiality of the individual elicitation opinions will be maintained.

MEETING NOTES FROM US NRC LOCA ELICITATION WRAP-UP MEETING
DOUBLETREE HOTEL, ROCKVILLE, MD
FEBRUARY 10-12, 2004

Day 1 – February 10, 2004

Rob Tregoning from the USNRC welcomed everyone and explained logistics for the meeting. Rob had everyone introduce themselves. Next, Rob reviewed the agenda for the three-day meeting. Day 1 would focus mainly on piping; Day 2 on non-piping, and Day 3 on emergency and faulted loads plus soliciting feedback on the process. There are no results to present on the topic of emergency and faulted loads. Only the basic approach will be shown.

Presentation #1 – Elicitation Project Plan, Schedule, and Milestones
By Rob Tregoning

The NRC has initiated some ongoing work looking at active mechanisms, e.g., stuck open valves. Bill Galyean is doing this.

There is a SECY paper due to Commissioners on March 31, 2004 with LOCA frequencies for normal operating loads.

Rob will distribute a draft NUREG documenting expert elicitation results so the panel can provide feedback on the NUREG. Rob experts that the experts will only a have short time (~2 weeks) to provide feedback.

For the April-June public meetings, Bruce Bishop from Westinghouse suggested meeting with the Westinghouse Owners Group (WOG) risk-based group.

Most critical future milestone is finalizing individual expert responses for normal operating loading frequencies by February 25th.

Presentation #2 – Elicitation Results (Box and Whisker Plots
By Rob Tregoning

Rob made a presentation on the details of the box and whisker plots that will be shown over the next 3 days. Many different methods of calculating percentiles; we used Standard method; fundamental message is that doesn't make much difference in final analysis.

Presentation #3 – Safety Culture
By Rob Tregoning

Bruce commented on VG4 with respect to poor US safety culture that didn't see a problem until starting seeing circumferential cracks; Pete commented that it was an economic issue since US utilities charge 7 to 10 cents per KW-hr while overseas may charge 40 cents per KW-hr

Karen Gott pointed out that her experience was that the first time a plant experiences a problem it is a big problem so she would agree with second major bullet on VG#4; on second sub-bullet she thought a better word is experience instead of sensitive.

On VG#5 it was pointed out that while the NRC may have only one vote on code changes, it has the ultimate veto vote.

Bruce felt day of single plant utility is numbered. Helmut Schulz and Karen disagreed with the last sub-bullet on VG6; utilities are willing to invest in older plants since they are already paid for (less capital investment); may be a difference in international experience and US practice.

Helmut commented with regards to VG9 on decommissioning that they have not seen any increase in LER events in the last few years before decommissioning for those plants that were decommissioned.

With regard to VG10 and negative bullets related to risk-informed regulation. Bill Galyean commented that utilities have limited resources and risk informed process helps prioritize; Vic Chapman cautioned against turning crank and getting an answer without thinking of why.

Bruce commented that end of life may be 80 years not 60; based on comment made at NRC recent meeting.

Helmut pointed out that boric acid corrosion of manway bolts of 15 to 20 years ago was more serious from a LOCA perspective that Davis Besse head problem of today.

Bottom line is no effect of safety culture on LOCA frequencies. There will be no adjustments to frequencies; no major discussion on part of panel with regards to this bottom line conclusion.

Presentation #4 – Piping Base Case Evaluation I
By Bengt Lydell

Bengt used a bottoms up evaluation based on service history experience.

Markov is standard approach common to any advanced reliability approach; this was the technical basis used by Bengt to develop time dependency.

Bengt's model allows for imperfect repairs or inspections.

Can go from S (unflawed condition) to R (rupture condition) if have some extraordinary event such as gas accumulation at Hamaoka in Japan.

VG13 results are per "weld year"; some of earlier plots are "per reactor year".

Presentation #5 – Piping Base Case Evalution II
By Dave Harris

Vugraph #3 is a summary of revised results since July 2003.

Could use VC Summer Hot Leg/RPV nozzle weld crack as benchmark, but need to be careful to consider all aspects such as differences in weld residual stresses due to repairs.

VG12 shows an order magnitude difference in LOCA frequency for a difference of 14 MPa (2 ksi) in normal operating stress; this was thought to be pretty sensitive result.

Dave assumed a linear residual stress field through the thickness due to the weld overlay repair.

The solid line in VG13 is PRAISE result while prior and post symbols come from Bengt's results.

Presentation #6 – Elicitation Question I: Base Case Evaluation
By Rob Tregoning

There was quite a bit of disagreement on first bullet on with regards to the perceived disadvantages of the various approaches.

Bruce argued that PFM approaches have been benchmarked against service history and shown to agree well.

Helmut and Bruce stressed that we are not in a position to review various approaches and we should not provide such a review in the NUREG report.

Need to stress in NUREG that general comments are not a group consensus, but individual responses.

Presentation #7 – Elicitation Calculation Framework
By Lee Abramson

Split distributions necessary if upper bound and lower bound are not symmetric with respect to mid value.

Used log normal distribution since results provided by participants fit log-normal distribution; also log-normal distribution easily to manipulate; also tradition is that log-normal is used in risk based approaches; bottom line is that due to variability in responses should not make that much difference as to what distribution chosen.

Analysis will yield medians of mid values and bounds as well.

Presentation #8 – PWR Piping
By Rob Tregoning and Paul Scott

The second main bullet on VG7 should be decrease with "decreasing" piping size.

Dave Harris disagreed with comment that PFM models had problems modeling mitigation.

There was a problem with VG16 with interpreting results for Expert L.

A number of the experts were surprised with VG17 that surge line results for Cat 5 are comparable to that of cold leg.

There is a discrepancy in maximums for Category 6 LOCAs at 25 years between VG19 and 20. VG20 shows participant L having the maximum value while VG19 shows participant B as having the maximum value.

Participant J showed the most impact of age on the LOCA frequencies; really obvious in VG21.

Bruce pointed out that for higher Category LOCAs that there was less uncertainty; may be an artifact that there are less systems than can contribute (only large diameter); also these larger systems are better inspected, i.e., better controlled.

Every plot shown is for 25 years unless specifically stated.

Presentation #9 - BWR Piping
By Rob Tregoning and Paul Scott

Similar degradation mechanisms as with PWR mechanisms: thermal fatigue, mechanical fatigue,

Sam commented with regards to Slide #4 that analysis of BWR feedwater says it should crack, but don't find these cracks in service.

For Slide 8 it was suggested to change "risk" to "high" in last bullet.

Maximum diameter of reactor water cleanup (RWCU) system is 6 inch; not 24 inch as shown in VG17.

Day 2: February 11, 2003

Rob started the second day at 0810 by reviewing agenda for the next two days.

Presentation #10 – Non-Piping Database Development
By Bill Galyean

LER database at ORNL will no longer be available after 2/29/04; the database will be moving to INEEL but in a different format.

Failures defined as leaks or cracks.

Presentation #11 – Non-Piping Base Case Development: CRDM and LTOP LOCAs
By Pete Ricaradella

Pete used the VIPER program to predict beltline failure frequencies (per vessel year) for typical BWRs.

VG3 and 4 are frequency plots and not probability plots.

For VG7 for large Category LOCAs, see big impact with time between 40 and 60 years; attributed to effect of radiation embrittlement; for smaller LOCAs don't see much effect of time.

EDY stands for Effective Degradation Years; used to normalizes degradation to a reference of a 600F operating temperature.

For CRDM nozzle ejection probability the assumption in VG11 that immediately have circumferential TWC of 30 degrees is highly conservative according to Bruce in that most are axially oriented.

Of 30 plants, there were 11 nozzles that had circumferential cracks; all of these plants were at about 20 EDY so Pete could take time factor out; total number of nozzles in 30 plants was 881

POD curve for NDE; cracks were EDM notches that were compressed to make them tight; eventually will get some real cracks from the North Anna head that can be used for calibration/validation

VG15 shows the probability of leak in one of 98 nozzles in this plant per vessel year; shows effect of NDE on probability of leakage

VG15 and 16 show effectiveness of NDE and how PFM models can account for inspection in their analyses.

VG17 shows decreasing frequency with time which reflects benefit of inspection.

To get 19,000 lpm (5,000 gpm) leakage, need ejection of 2 nozzles; most likely scenario is for collateral damage as one ejects and causes damage to adjacent nozzles.

Presentation #12 – Steam Generator Tube Rupture Frequencies
By Rob Tregoning

Almost everyone agreed that for PWRs the dominant failure scenario for Category 1 LOCAs was steam generator tube failures.

Used non-piping database which was augmented back to 1987 to capture 2 major events in '87 and '89. There have been 15 leaks since 1990 with 4 events over 380 lpm (100 gpm) since 1987.

On VG2 the reference to Nine Mile Point should be to Indian Point.

Fred Simonen stated that usual scenario assumed for higher category LOCAs is common cause failures such as losing pressure on secondary side causing pressure differential across tube and failure of multiple already degraded tubes.

Rob's analysis of independence (ignoring common cause) was viewed with a great deal of skepticism; Bill Galyean commented that if look at LERs always see multiple tube degradation but only see single tube rupture in the LERs.

This analysis is for 25 years, panel members left to their own devices for later years.

Presentation #13 – Overview of PTS Re-Evaluation Project
By Rob Tregoning

For this analysis all of the crack growth is from a PTS event; not fatigue.

For plants with multiple pass cladding there is a very low probability of flaw penetrating multiple passes; exception was Oconne that was single pass cladding.

Big driver were flaws between plate and axial flaw region.

For those that want to anchor against PTS, Rob will use updated results base on some average values for Oconee, Beaver Valley, and Palasdies

Presentation #14 – Elicitation Question VI: PWR Non-Piping
By Rob Tregoning

Analyses tend to get easier as go up on LOCA sizes in that less systems to worry about.

As a group the panel was uncomfortable with comments on Effect of Operating Time for Category 4 LOCAs.

Bill Galyean suggested that for presentation to public, may want to consider some other means of reporting extremely low frequencies (~1e-15); Rob wasn't sure he could define cut off value; also when look at median values, these low numbers don't impact final answer; Helmut supported this approach.

Helmut suggested that we explicitly state that a major assumption was that everything was fabricated in accordance with Code standards; no counterfeit bolts, etc.

Participant J only has fatigue in his analysis, once he includes other mechanisms, then his frequencies will go up by 2 or 3 orders of magnitude, but will still be low and probably will not alter the final median results.

Presentation #15 – Elicitation Question VI: BWR Non-Piping
By Rob Tregoning

Two of the six respondents who responded to this question only provided frequencies out to Category 5 LOCAs even though BWR non-piping could cause a Category 6 (Vessels) LOCA; thus Rob only provides results for Category 5 LOCAs here.

Concern with thermal aging of cast stainless steel is when it is present in concert with some other degradation mechanism.

CRDM refers to stud tube housing on bottom head of BWRs.

Expert F didn't consider RPV for Category 1 LOCAs; question was whether they didn't think important or did they not have a means of making an evaluation.

Lots less spread in results for RPV than valves and pumps; spent more time and more work in past on RPV than pumps and valves.

Expert C sees a decrease in freq with time but may be an artifact of the fact that he anchored against BWR recirculation line that shows a decrease in LOCA frequency with time; he provides no rationale for why he would expect non-piping LOCA frequency to decrease with time.

Presentation #16 Piping and Non-piping combined Results
By Rob Tregoning

Ratios of non-piping to piping for various category LOCAs are for 25 years only.

Pete made the point that most of the plots that Rob has shown are for 25 years, while he thought 40 and 60 years more important since 25 years is in past and the associated problems have been addressed while 40 and 60 years are for future; Rob responded that not that much difference between 25 and 40 years with some effect for certain category LOCAs at 60 years.

Sam commented that he was somewhat surprised that non-piping contribution less than piping contribution for BWRs in that piping has some active mechanisms that have been successfully mitigated in past; Karen responded that non-piping components more robust.

Bill Galyean warned about combining group distributions and panel distributions that may introduce a bias in that various members of group defined boundaries of system differently.

Much discussion on whether to chose group median or individual medians; Rob and Lee haven't done panel distributions yet.

Comparison of BWR and PWR – effect of mitigation encompassed in results for BWRs, but not PWRs – BWRs have been doing mitigation for 15 to 20 years whereas PWRs are just starting with mitigation for PWSCC.

Inclusion of S/G tube rupture for Category 1 LOCAs will be problematic for some people in that PRA people aren't used to accounting S/G tube failures in with rest of data; typically S/G tube rupture data is presented separately; in future Rob will present data both with S/G tube rupture data and without.

Categories 1, 2, and 3 are historically same as small, medium, and large break LOCAs respectively.

For PWR MB LOCAs, major contributor is CRDM, not S/G tube ruptures.

Discussion of whether MB LOCA was Category 2 or Category 3; some thought that MB LOCA was more in line with Category 3 LOCA.

Bengt felt we are comparing apples and oranges as we try to compare our results with historical results; NUREG/CR-5750 didn't look at non-piping per se whereas we did, although Bill Galyean indicated that if there had been indications of TWC in non-piping components then he would have included that data in his analysis in 5750; some thought that due to apples and oranges nature of our approach with 5750 that we shouldn't present these comparisons but Rob argued that if we don't present these comparisons then others will; Some argued that we should present frequencies for multiple LOCA categories when we compare with small, medium, and large break LOCAs for 5750

Lee reviewed the feedback questionnaire.

Day 3: February 12, 2004

Presentation 17: Emergency and Faulted Loading: Elicitation Approach and Responses

Water hammer type loadings should be in normal operating loading history.

What we are asking panel members to estimate is only the conditional failure probability given a stress with magnitude i ($P_{L/Si}$)

Ken commented that the seismic anchor motion (SAM) stress which is a secondary stress may be a bigger contributor than some primary stresses such as inertial stresses.

On VG entitled Elicitation Requirements, we are asking panel to do first bullet, we will do 2nd bullet, and plants would do 3rd and 4th bullets.

Asking them for a given system and degradation mechanism for their estimate of L50, P50, Ppl, L_{pl}, L_{tsl}, and P_{tsl} and then we will interpolate to get entire curve.

Some people argued during their elicitations that non-piping and large piping are non contributors to LOCA frequency due to seismic; we will look at results from piping and then decide what and if we will do anything for non-piping considerations.

Bruce argued that can get some very high loads due to malfunctions of snubbers.

P_{bc} and L_{bc} are probabilities for base case and likelihood of base case

Presentation 18: Remaining Work
By Rob Tregoning

Rob discouraged panel to make changes to bring their results more in line with others, would encourage panel to make changes if they heard something technically that made them rethink their answers.

Everyone will be involved in reviewing and critiquing NUREG reports; everyone wanted to be involved with the process.

The question of how: possibly another meeting, VTC, circulate vugraphs for review and feedback (electronically); possibly couple with some other meeting (ASME, PVP, etc).

Karen and Dave would want to meet before the NUREG was finalized; others seemed to agree with this.

Ideally we would circulate draft, we would then get comments back, we would then synthesize comments and then feed them back to the group and then meet; all before finalizing NUREG.

Helmut and Bruce supported idea of VTC (maybe limit to a few sites).

Other option is provide slides; review slides on computer and then have a conference call to review; limit to a few hours at a time (bite off small chunks).

Presentation #19 – Remaining work on Active Systems
By Bill Galyean

PORV stands for pilot operated relief valve.

Difficult to correlate stuck open valves categories with leak rate sizes/categories; size of valves will vary between plants.

Presentation #20 – Emergency and Faulted Loading Base Case Development
By Gery Wilkowski

No uncertainties applied to loads in Gery's analysis.

Base case assumes idealized TWC geometry.

Did not do any subcritical crack growth; thus did not consider residual solutions.

LBB.ENG2 is in form of closed form solutions.

For predicting large crack growth in a pipe tests it is better to use J-M than J-D.

Duane Arnold crack would have failed if subjected to Level B type loading.

Global secondary stresses act as primary stresses if crack large enough such that failure stress is below yield strength.

Presentation #21 – Lee reviewed feedback

It would have helped to have had Gery's presentation earlier, before the panel tried to answer seismic question.

Would have been nice to have a video of plants showing various systems as one tours plants with video camera.

Amount of information available was overwhelming; try to do a division of labor so one or two people review something and provide a tutorial to others so everyone is working from same basis; otherwise everyone is inventing the wheel themselves; maybe have a meeting to review these tutorials.

Need a roadmap of where information can be found.

Periodic/weekly update of changes made to ftp site; alternatively an alert message when something added to site; maybe a readme file when something added and what was added and when.

NRC management must make sure that staff are available to panel members during the process; Rob getting pulled off for Davis Besse was a problem; delayed things and then panel members only had a few weeks to respond at the end.

Bruce would like time at meetings to do actual work on elicitations because once they get back home they will get pulled off on to other things and won't be able to get back to answering questions for a long time.

APPENDIX C

ELICITATION TRAINING EXERCISE RESULTS

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ELICITATION TRAINING EXERCISE RESULTS

As part of the panel member kick-off meeting in February 2003, elicitation training was provided for the elicitation panel. The training involved the panel members answering a series of almanac-type questions for which numerical answers were available. The panel members provided both their best estimate of the answer as well as relative ratios with respect to other quantitative responses. In this way the panel members got an appreciation of the benefits of the anchoring process used throughout the elicitation process.

C.1 Training Questions

The following questions were used in the training exercise.

Q1. According to the 2000 census, how many men 65 or over were there in the U.S.?

Q2. In 1995, how many American men age 65 or older suffered from the chronic conditions listed?

Q3. What is the ratio of the rate for men 45- 64 years old to the rate for men 65 and older for each of the conditions listed?

Q4. What is the ratio of the rate for men under 45 years old to the rate for men 45 - 64 years old for each of the conditions listed?

The answer to Q1 is 14.4 million. The chronic conditions referred to in Q2, Q3, and Q4 and the corresponding answers are listed in Table C.1.

Table C.1 Correct Value (CV) Results to Elicitation Training Questions

	Q2	Q3	Q4
Condition	Rate per 1000	(Age 45-64 rate) / (Age 65+ rate)	(Under 45 rate) / (Age 45-64 rate)
Arthritis	404.7	0.44	0.13
Cataracts	125.1	0.13	0.11
Diabetes	123.6	0.50	0.10
Hearing Loss	366.8	0.56	0.20
Heart Disease	362.4	0.40	0.17
Prostate Disease	118.0	0.30	0.054

C.2 Elicitation Training Responses

As described in Section C.3.2, the experts were asked to supply three numbers for each question: a mid-value (MV), a low value (LB) and a high value (UB). The MV has a nominal 50/50 chance of falling above or below the correct value. The interval (LB, UB) has a nominal 90% chance of covering the correct value.

The following tables summarize the responses made in the training exercise. There were between 15 and 17 sets of responses to each question. (Although there were only 12 experts on the panel, members of the facilitation team were also invited to participate.) The number of respondents is indicated following each question. The table columns summarize the responses relative to the correct value (CV). The first column indicates the number of respondents where CV < LB, i.e., where the coverage interval fell above the CV; the third column indicates the number of respondents where CV > UB, i.e., where the coverage interval fell below the CV. Thus, the total of the first and third columns is the number of respondents whose coverage intervals did not cover the CV. The second column lists three numbers that summarize the set of mid-values provided for each row of the table. These are the lower quartile (LQ), median and upper quartile (UQ), respectively. About one quarter of the MVs are less than the LQ and about one quarter of the MVs are greater than the UQ. Hence the interquartile interval (LQ, UQ), denoted by IQI, contains about one half of the MVs. (These three summary statistics are used to construct box and whisker plots, as described in Appendix L.) For ease of reference, the rounded correct values are listed following the conditions for the Q2 - Q4 tables.

Q1. According to the 2000 census, how many men 65 or over were there in the U.S.? (N = 17) (CV = 14.4 million)

Table C.2 Summary of Respondent Results for Question Q1

CV < Coverage Interval	LQ, Median, UQ	CV > Coverage Interval
N = 3	16, 20, 28	N = 0

Respondents tended to overestimate the CV. Since LQ = 16, about three quarters of the MVs were larger than the CV. However, percent coverage at 82% was near the nominal 90%, with 3 (18%) lying above the CV and none lying below.

Q2. How many American men age 65 or older suffered from the following chronic conditions in 1995? (N = 15)

Table C.3 Summary of Respondent Results for Question Q2

		Rate per 1000	
Condition	CV < Coverage Int.	LQ, Median, UQ	CV > Coverage Int.
Arthritis (405)	N = 1	135, 200, 400	N = 9
Cataracts (125)	N = 2	50, 150, 200	N = 2
Diabetes (124)	N = 0	90, 150, 250	N = 3
Hearing Loss (367)	N = 1	200, 300, 500	N = 5
Heart Disease (362)	N = 1	150, 200, 375	N = 6
Prostate Disease (118)	N = 3	125, 200, 375	N = 2

Four of the six IQIs covered the CV, and the two which did not almost did. Three of the medians were above the CV and three were below. Thus, the MVs for the six conditions as a whole exhibited no systematic bias in estimating the CVs. However, the coverage intervals tended to underestimate the CVs. Of the 90 coverage intervals, 27 (30%) lay below the CV and 8 (9%) lay above. The average percent coverage of all 90 intervals was 61%. Over the six conditions, the percent coverage ranged from a low of 33% to a high of 80%.

Q3. What is the ratio of the rate for men 45- 64 years old to the rate for men 65 and older for each of the conditions listed? (N = 16)

Table C.4 Summary of Respondent Results for Question Q3

		(Rate for ages 45-64) / (Rate for age 65+)	
Condition	CV< Coverage Int.	LQ, Median, UQ	CV > Coverage Int.
Arthritis (0.44)	N = 1	0.20, 0.30, 0.50	N = 5
Cataracts (0.13)	N = 2	0.10, 0.20, 0.30	N = 3
Diabetes (0.50)	N = 0	0.25, 0.40, 0.50	N = 3
Hearing Loss (0.56)	N = 0	0.25, 0.30, 0.30	N = 6
Heart Disease (0.40)	N = 0	0.30, 0.30, 0.50	N = 2
Prostate Disease (0.30)	N = 2	0.20, 0.20, 0.40	N = 3

Five of the six IQIs covered the CV, but respondents tended to underestimate the CV. Five of the six medians were below the CV. Of the 96 coverage intervals, 22 (23%) lay below the CV and 5 (5%) lay above. The average percent coverage of all 96 intervals was 72%. Over the six conditions, the percent coverage ranged between 62% and 88%.

Q4. What is the ratio of the rate for men under 45 years old to the rate for men 45 - 64 years old for each of the conditions listed? (N = 16)

Table C.5 Summary of Respondent Results for Question Q4
(Rate for under 45) / (Rate for ages 45-64)

Condition	CV < Coverage Int.	LQ, Median, UQ	CV > Coverage Int.
Arthritis (0.13)	N = 2	0.10, 0.20, 0.30	N = 2
Cataracts (0.11)	N = 1	0.05, 0.10, 0.20	N = 1
Diabetes (0.10)	N = 8	0.20, 0.30, 0.50	N = 0
Hearing Loss (0.20)	N = 1	0.10,.0.20, 0.30	N = 2
Heart Disease (0.17)	N = 2	0.12, 0.20, 0.30	N = 2
Prostate Disease (0.054)	N = 6	0.05, 0.10, 0.20	N = 1

Five of the six IQIs covered the CV, but respondents tended to overestimate the CV. Four of the six medians were above the CV, and two were equal or almost equal to the CV. Of the 96 coverage intervals, 20 (21%) lay above the CV and 8 (8%) lay below. The average percent coverage of all 96 intervals was 71%. Over the six conditions, the percent coverage ranged between 50% and 88%.

C.3 Discussion

The results of the training exercise were consistent with several of the basic premises underlying the elicitation structure and methodology. First, apart from Q1, the responses to the other three questions as a whole did not exhibit any systematic over- or under- estimation bias. Q2 had no systematic bias, Q3 tended to underestimate, and Q4 tended to overestimate the CVs. This result is consistent with the basic premise of the elicitation process, which is that the panel responses as a whole have no systematic bias (see Section C.3).

Second, the percent coverage of the (LB, UB) intervals were less than the nominal 90% for all four questions. Q1 had the highest percent coverage at 82%, perhaps because the question dealt with demographic data with which the respondents were relatively more familiar. Q3 and Q4 had the next highest percent coverage at about 71% each and Q2 had the lowest percent coverage at 61%. This result is consistent with the rationale for the overconfidence adjustments made to the panelists' uncertainty intervals (see Section E.6.2).

Third, the two questions (Q3 and Q4) that asked about ratios of rates had higher percent coverage than the question (Q2) that asked about absolute rates. This result is consistent with the rationale for the basic structure of the elicitation questions, which ask about relative rather than absolute LOCA frequencies (see Section C.8).

APPENDIX D

PIPING BASE CASE RESULTS OF BENGT LYDELL

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An Application of the Parametric Attribute-Influence Methodology to Determine Loss of Coolant Accident (LOCA) Frequency Distributions

Report No. 2 to the NRC Expert Panel on LOCA Frequency Distributions

Prepared for

U.S. Nuclear Regulatory Commission
Washington (DC)

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Mr. Karl N. Fleming (Technology Insights, Inc., San Diego, CA) provided constructive review comments on a draft of this report. Mr. Fleming also provided the Markov model solutions supporting the calculation of time-dependent LOCA frequencies.

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ABBREVIATIONS

ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
CL	Cold Leg (of a PWR RCS)
CV	Chemical and Volume Control
DN	Nominal Pipe Size, metric [mm]
DPD	Discrete Probability Distribution
EPRI	Electric Power Research Institute
FW	Feedwater
HAZ	Heat Affected Zone (of weld)
HL	Hot Leg (of a PWR RCS)
HPI	High Pressure Injection
HWC	Hydrogen Water Chemistry
IGSCC	Intergranular Stress Corrosion Cracking
IHSI	Induction Heat Stress Improvement
ISI	Inservice Inspection
LOCA	Loss of Coolant Accident
MSI	Mechanical Stress Improvement
MV	Mid Value (50% percentile)
NDE	Nondestructive Examination
NMU	Normal Makeup
NWC	Normal Water Chemistry
NPS	Nominal Pipe Size, US [inch]
PBF	Pressure Boundary Failure
PWSCC	Primary Water Stress Corrosion Cracking
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RR	Reactor Recirculation
SC	Sensitivity Case
SI	Safety Injection
SS	Stainless Steel
TS	Technical Specifications[1]
UT	Ultrasonic Testing

Notation

A	Attribute
a/t	Ratio of crack depth to pipe wall thickness
C	Conditional failure probability given a flawed weld and an unusual or severe loading condition
F	Failure (= through-wall flaw)
L	Large leak
P	Probability
S	Susceptibility (to degradation)
W	Weld count
Ø	Pipe diameter
ϕ	Crack occurrence rate
λ	Failure rate (frequency of PBF resulting in leak rate $\leq$ TS limit for unidentified leakage)
ρ	Rate of large leak event
ν	Leak/spill rate (gpm)
σ_{NO}	Normal operating weld stress (ksi)
ω	Repair rate

[1] For TS leak rate limits see for example NUREG-1431 (Vol 2, Rev. 2, June 2001): Standard Technical Specifications Westinghouse Plants – Bases, Section B 3.4.13, RCS Operational Leakage. For unidentified leakage the Limiting Condition for Operation (LCO) is 3.8 lpm (1 gpm), and for identified leakage the LCO is 38 lpm (10 gpm). See also NUREG-1433 (Vol. 2, Rev. 2, June 2001): Standard Technical Specifications General Electric Plants, BWR/4 – Bases, Section B 3.4.4, RCS Operational Leakage. For unidentified leakage the LCO is 19 lpm (5 gpm). Further, if an unidentified (BWR) leakage has been identified and quantified, it may be reclassified and considered as identified leakage; however, the total leakage limit would remain unchanged.

D.1 Background

Limited to consideration of Code Class 1 piping failures, Base Case Report Number 2 documents an assessment of BWR and PWR loss of coolant accident (LOCA) frequency distributions. The assessment is a demonstration of the role of statistical analysis of service experience data and Markov modeling in a "bottom-up" approach to piping system reliability analysis.

D.1.1 Objectives

Using primary coolant piping design information for three reference plants (one BWR plant and two PWR plants), the overall objective is to determine LOCA frequency distributions that are representative of currently operating U.S. nuclear power plants, including current in-service inspection (ISI) practices and degradation mitigation strategies. This determination is done analytically using a parametric model of piping reliability. The LOCA frequency distributions are determined for three time periods. To address today's piping reliability state-of-knowledge the LOCA frequency is determined at T = 25 years. Next the LOCA frequency is extrapolated to T = 40 years to represent the primary system piping reliability status at the end of a 40-year operating license. Finally an extrapolation is made to T = 60 years to account for a possible license renewal. Analytically, this extrapolation is concerned with the potential impact on the structural integrity of the piping by material aging as well as by reliability improvement efforts.

As implied by the report title, the objective is to develop LOCA frequency distributions. The report addresses two aspects of LOCA frequency distributions. It develops LOCA frequencies associated with a distribution of flow rate threshold values ranging from 380 lpm (100 gpm) at the low end to beyond 380,000 lpm (100,000 gpm) at the high end. Additionally the study develops statistical uncertainty distributions for each set of LOCA frequencies to account for the uncertainty in the input parameters to this piping reliability analysis.

D.1.2 Base Case Definition

During a meeting in Rockville (MD) in February 2003 [D.1], the Expert Elicitation Panel members defined five Base Cases that are denoted as BWR-1, BWR-2, PWR-1, PWR-2 and PWR-3, respectively. The five Base Cases are:

<u>BWR Base Case (Plant 'B')</u>

- <u>BWR-1</u>; Reactor Recirculation (RR) System. This reference case includes one-of-two RR System loops. Each loop consists of one NPS28 recirculation pump loop with a NPS22 manifold with five NPS12 risers; NPS is nominal pipe size in inch. The reference case excludes any small-diameter piping or tubing attached to the main RR piping. With a few exceptions, the selected piping system layout is representative of a BWR/4 reference plant as described in NUREG/CR-6224 [D.2]. The Base Case RR System does not include the NPS4 bypass line, however. The RR piping is fabricated from austenitic Cr-Ni stainless steel of Type A-304 (≥ 0.035% carbon).
- <u>BWR-2</u>; Feedwater (FW) System. As defined by isometric drawings, this reference case includes Loop B of the Class 1 portion of the FW System (i.e., the part of the FW System that is located in the drywell containment structure). This system of two loops includes NPS12, NPS14 and NPS20 piping. The FW piping is fabricated from carbon steel of Type A-333 Gr. 6.
- Section D.1.3 includes additional information on the BWR Base Case system definitions.

<u>PWR Base Case (Plant 'A.a/b')</u>

- <u>PWR-1</u>; Reactor Coolant (RC) System. As defined by an isometric drawing, this reference case includes one of the NPS30 hot leg (HL) in the RCS.

- PWR-2; Pressurizer Surge Line. As defined by an isometric drawing, this reference case includes the NPS14 piping, which connects the pressurizer to the cold leg (CL).
- PWR-3; High Pressure Injection/Normal Makeup (HPI/NMU) System. As defined by an isometric drawing, this reference case includes the 2-½ inch schedule 160 line between the containment isolation valve and the RCS cold leg (CL).
- The PWR base cases associated with the RC hot leg and pressurizer surge line are typical of a 3-loop Westinghouse PWR (Plant A.a). The PWR base case associated with the HPI/NMU line is typical of a Babcock & Wilcox PWR (Plant A.b).
- Section D.1.4 includes additional information on the PWR Base Case system definitions.

D.1.3 BWR Base Case System Descriptions

Plant B is a BWR/4 assumed to have been in commercial operation for at least 25 years. Similar to many other operating BWR/4 plants in the USA, Plant B is also assumed to be operating with a combination of IGSCC Category D and E welds, according to the nomenclature of U.S. NRC Generic Letter 88-01 [D.3, D.4]. In other words, the plant has experienced some IGSCC and the affected welds have been reinforced by weld overlays. It is further assumed that none of the IGSCC susceptible welds have been subjected to any stress improvement (SI) process such as induction heat stress improvement (IHSI) or mechanical stress improvement (MSI). It is also assumed that the weld overlay repairs (WOR) were all performed in the 1982-1988 timeframe. Finally, Plant B is assumed to have been operating with normal water chemistry (NWC) at all time.

The system descriptions in this section are extracted from design information supplied by members of the Expert Elicitation Panel. The BWR-specific system information is included in the following documents and drawings:

- Document No. EPRI-156-310: Degradation Mechanisms Evaluation for Class 1 Piping Welds at Plant B [D.5].
- Excel-file entitled "PlantBWelds." This Excel-file includes weld lists with locations for the RR and FW ASME Section XI Code Class 1 piping. The lists are organized by weld identification numbers (as they appear on the isometric drawings identified below) nominal pipe size and pipe schedule. The Excel file forms the basis for the LOCA frequency model used to derive the LOCA frequency distributions.
- Isometric drawing numbers 6M721-5358-5 (RR System Loop B Ring Header), 6M721-5359-5 (RR Loop B Suction & Discharge Piping), 6M721-2336-1 (FW System Inside Drywell), and 6M721-3537-5 (FW System Inside Drywell).

D.1.3.1 Reactor Recirculation (RR) System - The RR System evaluated in this study consists of two recirculation pump loops external to the reactor pressure vessel (RPV). These loops provide the piping path for the driving flow of water to the RPV jet pumps. Each loop contains a variable speed recirculation pump and two motor operated isolation valves (one on each side of each pump). The recirculation loops are part of the nuclear system process barrier and are located inside the drywell containment structure. The pipe segments that are subject to evaluation in this study consist of:

Loop A: The Class 1 portion starts at the RPV nozzle N1A and is reconnected to the RPV at nozzles N2F, N2G, N2H, N2J, and N2K. Class 1 lines for the Residual Heat Removal (RHR) and Reactor Water Cleanup (RWCU) Systems are connected to this loop. These particular Class 1 lines are excluded from the study scope, however. Loop A is excluded from the BWR Base Case.

Loop B: The Class 1 portion starts at RPV nozzle N1B and is reconnected to the RPV at nozzles N2A, N2B, N2C, N2D, and N2E. Part the original design, a NPS4 bypass line at valve F031B has been removed from the system. Class 1 lines for the RHR and RWCU Systems are connected to this loop. These particular Class 1 lines are excluded from the study scope, however.

D.1.3.2 Feedwater (FW) System - The FW System provides feedwater to maintain a pre-established water level in the RPV during normal plant operation. The Condensate and the FW Systems take water from the main condenser and deliver it to the RPV after passing it through the feedwater heaters and demineralizer system. The Class 1 portion of the FW System consists of two loops:

Loop A: Loop A starts at valve F076A and a connection to the High Pressure Coolant Injection (HPCI) discharge line (at valve F006), and connects to the RPV at nozzles N4A, N4B, and N4C. The HPCI discharge line is excluded from the study scope. Loop A is excluded from the BWR Base Case.

Loop B: Loop B starts at valve F076B, connection to the Reactor Core Isolation Cooling (RCIC) discharge line at valve F013, and a discharge from the RWCU System (at valve F220), and connects to the RPV at nozzles N4D, N4E, and N4F. The RCIC and RWCU discharge lines are excluded from the study scope.

D.1.4 PWR Base Case System Descriptions

The system descriptions in this section are extracted from design information supplied by members of the Expert Elicitation Panel. The PWR-specific system information is included in the following documents and drawings:

- Document No. EPRI-156-330: Degradation Mechanism Evaluation for Class 1 Piping Welds at Plant A.a [D.6]. This document summarizes the degradation mechanisms applicable to a Westinghouse 3-loop PWR.
- References [D.7-D.9] include design information as well as degradation mechanism information applicable to the HPI/NMU system of Plant A.b.
- Excel file entitled "PlantAWelds." This Excel file includes weld lists for the RC system of Plant A.a. The lists are organized by weld identification numbers (as they appear on the isometric drawings identified below), nominal pipe size and pipe schedule. This Excel-file forms one of the bases for the PWR LOCA frequency model used to derive the LOCA frequency distributions.
- Isometric drawing numbers 1MS-22-2262 and CGE-1-4100A (RC Hot Leg), C-314-601 and CGE-1-4500A (pressurizer surge line), and 17-MU-23 (HPI/NMU piping).

D.1.4.1 Reactor Coolant (RC) System (Plant A.a) - The RC System evaluated in this study consists of three similar heat transfer loops connected in parallel to the RPV. Each loop contains a reactor coolant pump (RCP), steam generator, and associated piping and valves. In addition, the system includes a pressurizer, a pressurizer relief tank, interconnecting piping, and instrumentation necessary for operational control. The analysis in this report is concerned with a portion of one of the three RC loops; the portion from the RCP to the RPV (this is one of the hot legs). The pressurizer surge line connects the pressurizer to the RC cold leg Loop A. In summary, the piping sections that are subject to evaluation in this study consist of.

RC-HL: The analysis is concerned with 1-of-3 hot legs. The Loop A HL starts at the RPV, includes an RCP and connects to the 'A' steam generator (S/G). The HL piping is fabricated from stainless steel piping. The section of the HL from the RPV to the RCP is of 31 inch inside diameter, while the section from the RCP to S/G is of 27.5 inch inside diameter piping.

Surge Line: The single surge line is fabricated form NPS14 stainless steel piping and connects to the 29-inch RC cold leg.

D.1.4.2 High Pressure Injection (HPI)/Normal Makeup (NMU) Line (Plant A.b) - In Plant A.b, each of the four RCS cold legs is equipped with high-pressure injection piping. Two of these 2 ½ inch (ID, or approximately NPS3-¾) stainless steel piping lines provide the normal makeup flow to the RCS and they connect to the cold leg via nozzle assemblies. Each of the nozzles is comprised of a base nozzle and a safe-end. To prevent thermal cycling of the base metal each nozzle is equipped with a 1.5-inch thermal sleeve. The analysis is concerned with one of the two HPI/NMU lines.

D.1.5 Summary of Scope Limitations

As outlined above, this LOCA frequency assessment is limited to specific portions of BWR and PWR Code Class 1 piping. The BWR base cases include contributions from potential pipe breaks in Loop B of the respective RR and FW System. Pipe break frequency contributions from normally pressurized sections of HPCI, RCIS, RHR or RWCU piping are not considered in this study. Piping system design information beyond that itemized above is not accounted for in this study. The PWR base cases include contributions from 3-of-3 RC hot legs and 2-of-2 HPI/NMU lines, respectively.

Excluded from the analysis are LOCA frequency contributions due to degradation and failure of cast stainless steel components such as valve bodies. While there is some documented evidence of degradation of such components, (e.g., [D.10]) the frequency of a through-wall defect in valve bodies and pump casings is viewed as being considerably lower than for welds in Class 1 systems.

D.1.6 Technical Approach to LOCA Frequency Estimation

Existing service experience with piping systems shows a strong correlation between failures and presence of an active degradation mechanism in combination with service conditions and transient loading conditions. It is therefore possible to estimate piping reliability parameters through statistical analysis of service experience data. Such analysis includes data processing whereby the appropriate reliability attributes are correlated with influence factors as described in SKI Report 97:26 [D.11].

In this Base Case Report the technical approach to LOCA frequency estimation builds on statistical analysis of service data associated with ASME XI Class 1 piping in the BWR and PWR operating environments. The study accounts for two kinds of uncertainties in piping reliability analysis, namely data uncertainty and state-of-knowledge uncertainty. The pipe failure database on which this study is based is called PIPExp [D.12], which is the extended version of the OPDE pipe failure database [D.13]. A description of PIPExp is included in Appendix A. The uncertainty analysis is performed by using a Monte Carlo merge technique to develop the LOCA frequency distributions. A commercial software package called Crystal Ball (Version 2000.2.2), which is an add-on for Microsoft Excel, is used to perform this Monte Carlo merge operation. Time-dependent LOCA frequencies are developed using a Markov modeling approach [D.14].

The BWR Base Case analysis is based on the degradation mechanism analysis as documented in Reference [D.5], and it builds on insights from an earlier BWR LOCA frequency pilot study [D.15-D.16]. The PWR Base Case analysis is based on the degradation mechanism analysis as documented in References [D.6, D.8], and builds on insights and results from an earlier sensitivity analysis performed in support of a risk informed inservice inspection (RI-ISI) evaluation [D.12]. That sensitivity analysis addressed the impact of using a different pipe failure database on the RI-ISI weld selection.

D.1.7 Study Conventions

Throughout this report, pipe sizes are referenced by nominal pipe size (NPS), which indicates standard pipe size without an inch symbol. The smallest pipe size considered in this study is NPS3-¾ (Plant A.b). All references to specific material types are made according to designations by the American Society for Testing and Materials (ASTM). The term "weld failure" is used to indicate a rejectable (non-through-wall or through-wall) flaw.

During the NRC LOCA Elicitation Kick-off Meeting [D.1], a LOCA was defined as "a breach of the reactor coolant pressure boundary which results in a leak rate greater than 380 lpm (100 gpm)." Instead of using the traditional (or historical) LOCA size classes (small – medium – large) that are based on break size, this study uses LOCA sizes that are based on leak rate threshold values as indicated in Table D.1 (adapted from [D.1]) and Table D.2.

Table D.1 LOCA Size Classification Threshold Values

LOCA Category	Flow Rate (ν) Thresholds [gpm]	Comment
0	ν > 10	Cat0 corresponds to a pressure boundary failure (breach) resulting in a leakage exceeding the T.S. limit for identified leakage.
1	ν > 100	Breach in piping of up to 1.8-inch diameter (BWR), and 1.7-inch diameter (PWR); see Table D.3.
2	ν > 1500	Breach in piping of up to 3.3-inch diameter (BWR), and 3-inch diameter (PWR)
3	ν > 5000	Breach in piping of up to 7.3-inch diameter (BWR), and 6.8-inch diameter (PWR)
4	ν > 25,000	Breach in piping of up to 18.4-inch diameter (BWR), and 14-inch diameter (PWR)
5	ν > 100,000	Breach in NPS28 RR piping (BWR) yields on the order of 230,000 gpm. Breach in RCS hot leg piping of up to 31-inch diameter.
6	ν > 500,000	Applies to PWR RCS-HL base case only, and only for a relatively short time following a postulated DEGB

Table D.2 Estimated Flow Rates from Restrained Double-Ended Guillotine Break (DEGB)[2]

Pipe Size [NPS]	Restrained DEGB (Plant A – PWR)			Restrained DEGB (Plant B – BWR)		
	Break Size [sq.in.]	Press. [psig]	Max. Flow Rate [gpm]	Break Size [sq.in.]	Press. [psig]	Max. Flow Rate [gpm]
1	.41	2250	540	.41	1250	467
2	1.65	2250	2158	1.65	1250	1869
4	6.60	2250	8633	6.60	1250	7476
6	14.84	2250	19424	14.84	1250	16823
8	26.39	2250	32280	26.39	1250	29908
12	59.37	2250	72495	59.37	1250	42411
14	80.81	2250	98624	80.81	1250	57698
22	199.54	2250	243542	199.54	1250	142478
28	323.22	2250	394497	323.22	1250	230790
30	371.05	2250	452867	N/A	--	--

[2] Technical basis for leak rate calculation is documented in an attachment to Minutes of Meeting (2nd Elicitation Meeting), Bethesda (MD), June 4-5, 2003.

The estimation of weld failure rates uses Bayesian reliability analysis methodology, and involves the development of prior and posterior failure rate distribution. In this study the term 'prior' refers to piping reliability characteristics before the implementation of industry programs to mitigate or eliminate susceptibilities to certain degradation mechanisms. The term 'posterior' refers to observed or expected reliability characteristics after reliability improvement actions have been implemented.

D.1.8 Report Organization

This report consists of eight sections and four appendices. Section D.2 is an overview of the analysis steps. Section D.3 summarizes the service experience applicable to the BWR and PWR Base Cases, respectively. Using the PIPExp database, Section D.4 includes a summary of the data interpretation and data processing steps necessary to derive piping reliability parameters that apply to the base case definitions. Section D.5 documents the results of the pipe failure rate estimation while Section D.6 is a documentation of the models used for estimating LOCA frequency, while Section D.7 is a summary of results. Section D.8 is a list of references. Note that the Base Case results used in Table E.1 in the main body can be obtained from Tables 16, 17, and 20 in this report.

Appendix A summarizes the PIPExp database structure. Appendix B includes the Excel spreadsheets that are used as the basis for the LOCA frequency models, and Appendix C includes the Excel spreadsheets for the calculation of time-dependent LOCA frequencies. Finally, Appendix D is a summary of selected, significant Code Class 1 and 2 pipe failures in commercial nuclear power plants worldwide.

D.2 Technical Approach

Base Case Report 2 develops BWR and PWR LOCA frequency distributions using a 'bottom-up approach.' Statistical analysis of relevant service experience data is used to quantify the weld failure rate and rupture frequency of individual welds. Next the failure rate and rupture frequency (= LOCA frequency) for an entire system is calculated by concatenating the individual weld failure rates and rupture frequencies. Markov model theory is used to evaluate the influence of alternate strategies for in-service inspection and leak detection on the frequency of leaks and ruptures.

D.2.1 Overview of Analysis Steps

Different approaches have been applied to estimating pipe failure rates and rupture frequencies; from probabilistic fracture mechanics, via direct statistical estimation to expert judgment. The most straightforward approach is to obtain statistical estimates of piping component failure rates based on data collected from field experience. A variation of this approach is to augment statistical estimates of pipe failure parameters with simple correlations that express the problem in terms of a failure rate and a conditional probability for each failure mode of interest such as the approach used in NUREG/CR-5750 [D.17].

A limitation of the statistical analysis approach is that attempts to segregate the service data to isolate the impact of key design parameters and properties of various degradation and damage mechanisms often leads to subdividing a database into very sparse data sets. If not optimized properly, this approach may introduce large uncertainties in the failure rate estimates. In addition, historical data may reflect the influence of no longer relevant inspection programs. If changes to these programs have been implemented, such changes may render the failure rate estimates no longer relevant. In risk-informed applications, the failure data and analysis methods need to provide future predictions of piping system reliability that can account for changes in the inspection strategy or improvement in the NDE technology.

An objective of the work documented in this report is to demonstrate the utility of a pipe failure data collection. Time-dependent LOCA frequencies are calculated by making full use of the PIPExp database in combination with Markov model theory [D.14]. The LOCA frequency calculation in this report is structured to support the Expert Elicitation and consists of four steps; each step is addressed in a separate report section:

- Section D.3. The service experience that is applicable to the five bases cases is summarized in this section. The data summaries correspond to queries in the PIPExp database.
- Section D.4. The approach to calculating time-dependent LOCA frequencies is presented. A Bayesian update process is used to derive failure parameters that reflect the attributes of respective base case definition. The results of this analysis step are in the form of generic weld failure rate distributions. These distributions represent the industry-wide service experience prior to the implementation of the specific pipe failure mitigation programs that are currently in place.
- Section D.5. In this section current state-of-knowledge (or base case specific) weld failure rate distributions are develop. The chosen estimation approach includes a formal uncertainty analysis that accounts for uncertainty in the failure data and exposure data. Engineering judgment and insights from the review of service data are used to address the conditional probability of pipe failure given presence of through-wall flaws.
- Section D.6. An Excel spreadsheet format is used to develop LOCA frequency models corresponding to each of the five base cases. These models generate LOCA frequency distributions at T = 25 years. A Markov model is used to investigate the time-dependency of LOCA frequencies. The output of this model consists of LOCA frequencies at T = 40 years and T = 60 years.

D.2.2 Sensitivity Analyses

Two types of sensitivity analysis are included in this report. The first type addresses the impact on results by an assumed incompleteness of the failure data collection. The second type relates to the sensitivity of the time-dependent LOCA frequencies to different assumptions about leak detection and in-service inspection. The sensitivity analysis results are included in Section D.6.

D.3 Service Experience Data Application to the Base Case Study

The PIPExp database documents service experience with Code Class 1, 2 and 3 and non-safety related (or Class 4) piping in commercial nuclear power plants worldwide. For the time period 1970-2002, this database was queried for service experience data specific to the Base Case piping systems. The results of the database queries are summarized here, and they form the input to the data processing and failure parameter estimation in Sections D.4 and D.5.

D.3.1 PIPExp Database, Revision 2003.1

The pipe failure database utilized in the Base Case Study is called PIPExp. It is an ACCESS database and an extension of the OPDE database [D.12-D.13]. Since the conclusion of the original work in 1998 [D.11, D.17], the pipe failure database has been significantly expanded both in terms of the absolute number of event records and the depth of the database structure (Appendix A provides additional details). Lessons learned through database applications have been used to enhance the structure. In this study of HPI/NMU-, FW-, RC- and RR-piping reliability the statistical analysis is based on service data as recorded in PIPExp and with cutoff date of December 31, 2002. The analysis is inclusive of applicable worldwide BWR- and PWR-specific service experience with Code Class 1 piping. As of 12-31-2002 the database accounted for approximately 1,992 and 3,621 critical reactor-years of operating experience with commercial BWR and PWR plants, respectively.

The database is actively maintained and periodically updated. The effort involved in populating the database while at the same time assuring data quality is not trivial. As an example, changing regulatory reporting thresholds imply that an ever increasing volume of raw data reside in restricted and proprietary database systems rather than in the public domain. For an event to be considered for inclusion in the database it undergoes screening for eligibility. For example:

- The equipment failure must be positively identified as a piping component failure external to the reactor pressure vessel (RPV). A failure involves a pressure boundary degradation, which can be non-through-wall (crack with a/t-ratio ≥ 10%, where a = crack depth and t = wall thickness) or a through-wall leak.
- There must exist documented evidence in the form of a hard copy (e.g., USNRC Inspection Report, Licensee Event Report, ISI Summary Report, Problem Identification Form, Condition Report, ASME Code Repair Relief Request, etc.) from which a sufficiently detailed case history is developed. The documented evidence of pipe degradation/failure must contain information on its location within a piping system (e.g., with reference to an isometric drawing and/or P&ID), metallurgy, operating conditions, impact on operation, method of discovery, failure history, etc. so that a data classification may be independently verified.
- Where the documented evidence is deemed incomplete, additional information is solicited through direct contact with plant personnel or by accessing supplemental data.
- There must be sufficient technical information available to fully address the complex relationships between piping reliability attributes (or design parameters) and influence factors (e.g., fabrication/welding techniques, environmental conditions such as water chemistry, flow conditions) on the one hand and degradation/failure mechanisms on the other.
- Differentiation between UT indications versus confirmed crack indications. Only the latter are included in the database given an a/t-ratio ≥ 10%.

Following on the initial data screening, each event selected for inclusion in the database is subjected to a classification so that the unique reliability attributes and influence factors are identified. Including memo fields, text fields, numerical fields and data filters, up to 114 database fields describe each record of the database.

D.3.2 Review of BWR-Specific Piping Service Experience

Limited to the BWR Base Case systems, this section summarizes the worldwide service experience with Code Class 1 piping. The results of this review are input to the pipe failure rate estimation.

D.3.2.1 RR Piping Service Experience - The original piping material in BWR plants commissioned prior to mid-1980 is austenitic stainless steels that contain more than 0.03% carbon. During welding these steels are susceptible to sensitization that results in a loss of corrosion resistance. Intergranular stress corrosion cracking (IGSCC) occurs when the sensitized steel is subjected to stresses and corrosive environment. Sensitization can be avoided by controlling the carbon content to below 0.03%. Another approach to controlling sensitization is to add strong carbide formers such as titanium or niobium to the steel. Stainless steels with additions of titanium or niobium are called "stabilized." It is noted that low-carbon content unstabilized stainless steel or stabilized stainless steels are not completely immune to IGSCC, however [D.18].

For Plant B, IGSCC is the predominant degradation mechanism acting on the RR piping welds, including heat-affected zones. During early plant life some weld reinforcements were performed where the inservice inspection revealed presence of surface penetrating, and subsurface cracking due to IGSCC. Since the analysis of LOCA frequency distributions is based on a degradation mechanism evaluation, the PIPExp database is queried for service data including IGSCC. The database queries are summarized in a set of charts and tables below. The database currently includes a total of about 1000 records on IGSCC in BWR piping. Figure D. 1 shows the number of weld failures due to IGSCC by calendar year and Figure D. 2 shows the number of weld failures by year of operation. Here a weld heat affected zone with $a/t \geq 10\%$ is characterized as a "weld failure."

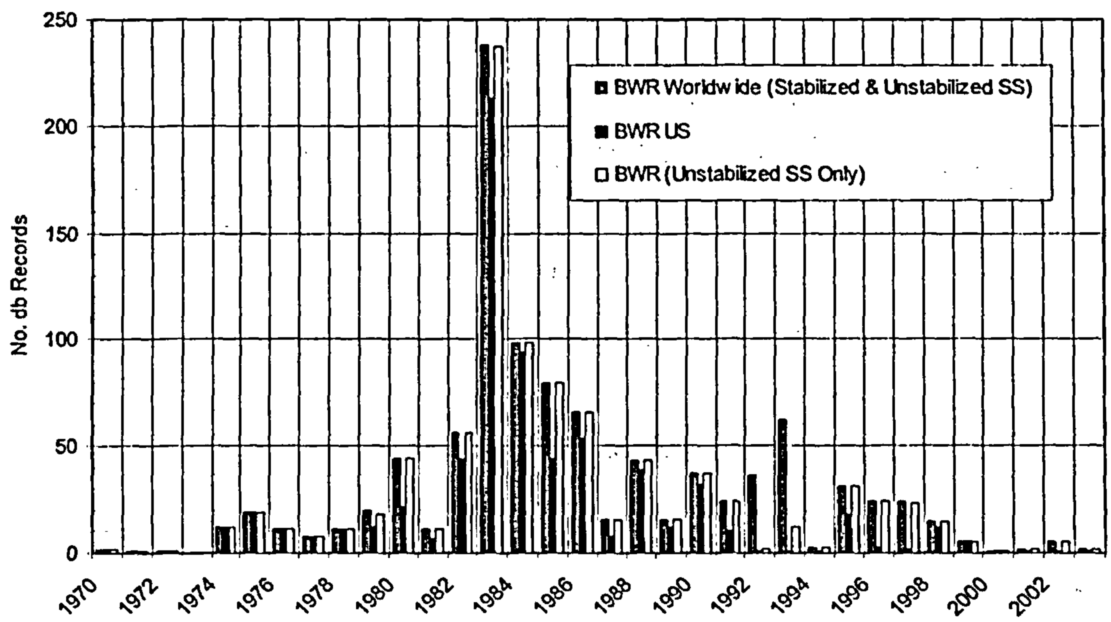


Figure D.1 Weld Failures Due to IGSCC in Code Class 1 & 2 Piping (1970-2002)

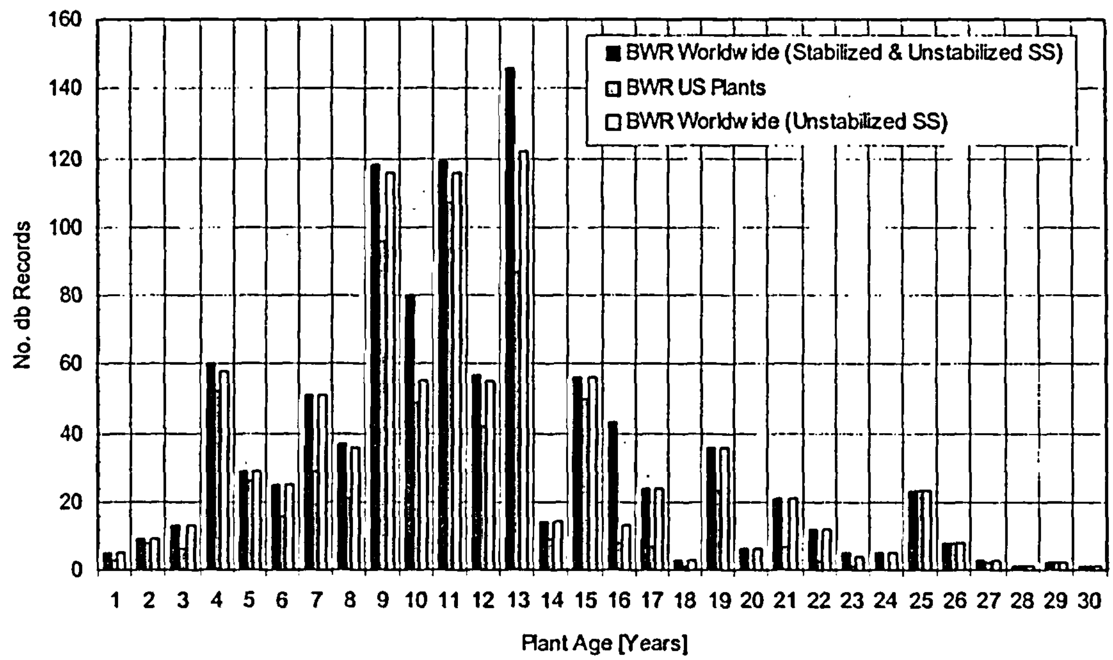


Figure D.2 IGSCC Experience by Year(s) of Operation

In Figure D. 3 the IGSCC data is organized by mode of failure (crack – pinhole leak – leak) and pipe size. Figure D. 4 shows the IGSCC data by size and material type.

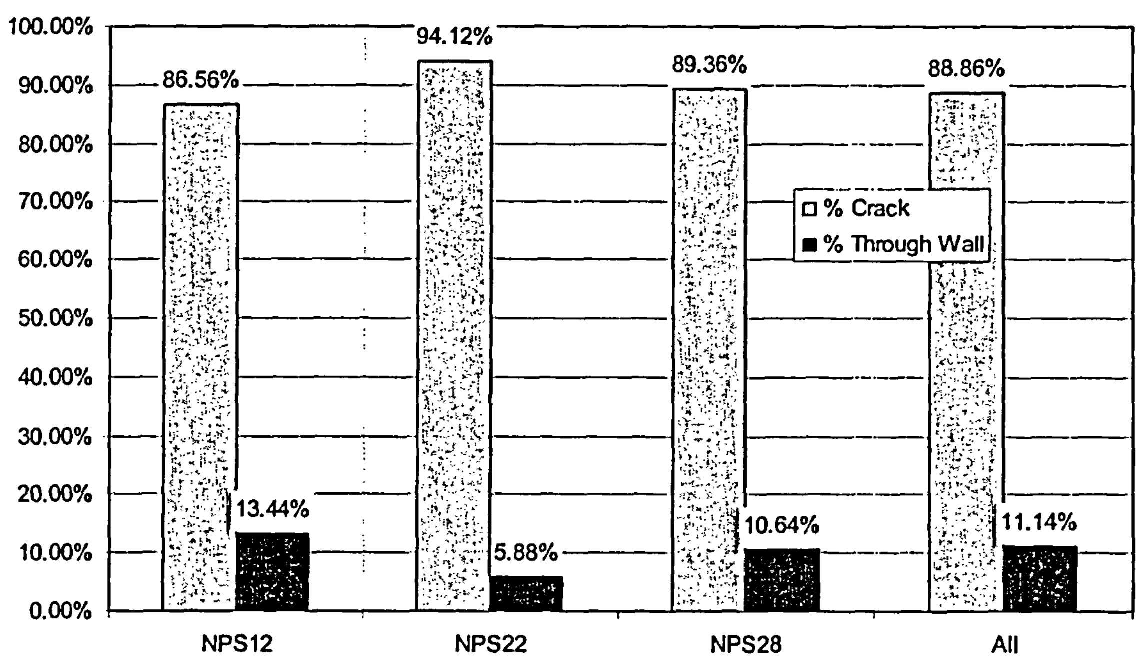


Figure D.3 IGSCC Data by Failure Mode

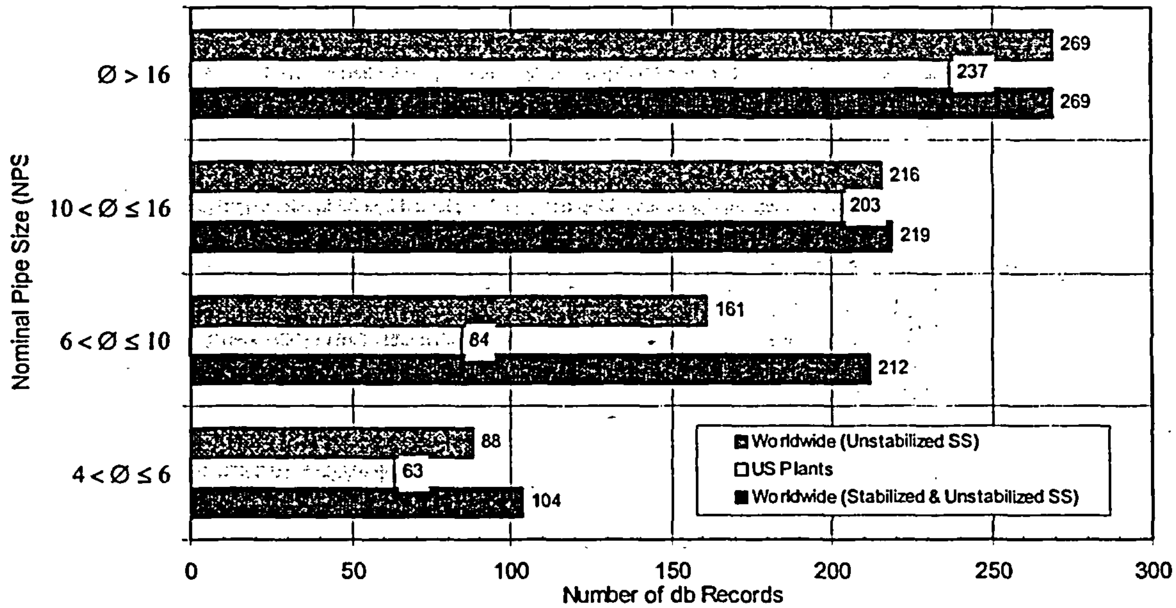


Figure D.4 IGSCC Experience by Pipe Size and Material Type

Figures D. 5 through D.7 include plots of crack depth versus crack length (L) in the circumferential (C) direction. An 'a/t-ratio' of 100% indicates a crack, which has penetrated the outside pipe wall. An 'L/C-ratio' of 100% indicates a crack, which spans the entire inside pipe circumference. Limited to part through-wall cracks, Figure D.8 summarizes the data by a/t-ratio.

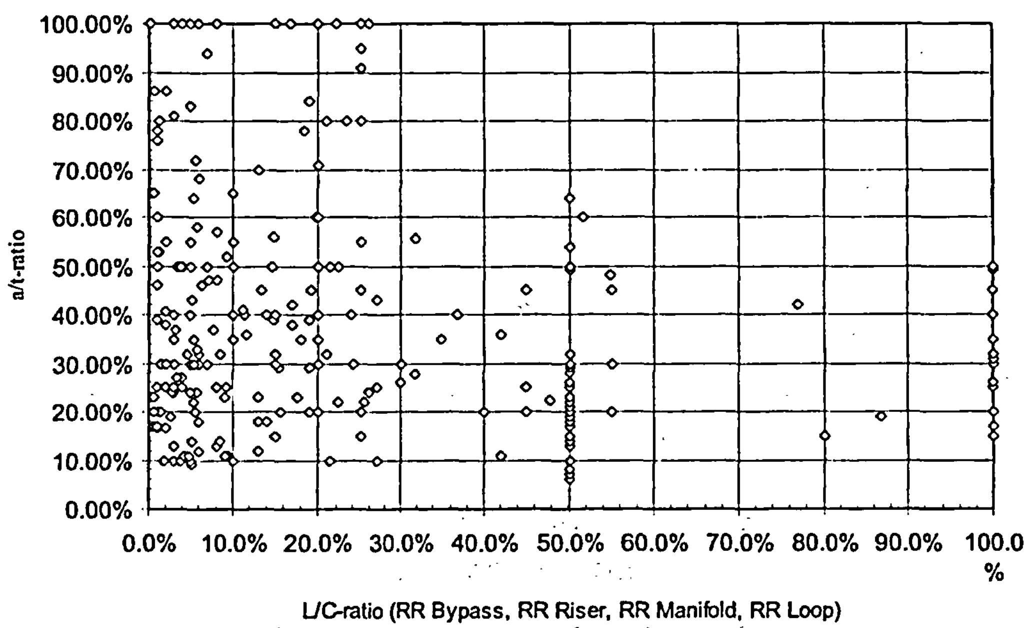


Figure D.5 Crack Depth Versus Crack Length in Austenitic Stainless Steel

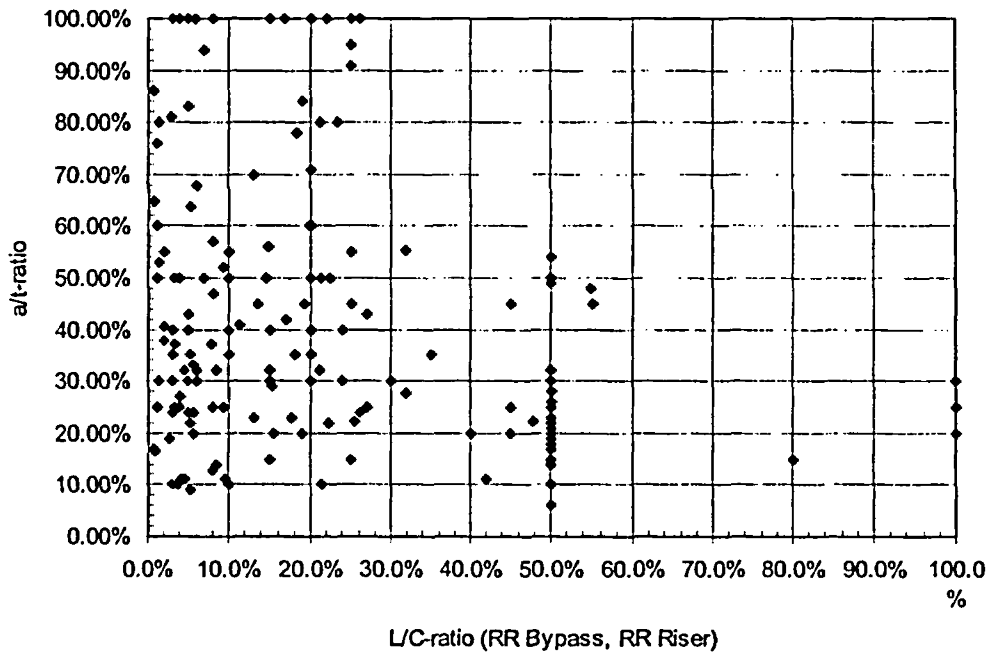


Figure D.6 Crack Depth Versus Crack Length in Austenitic Stainless Steel (NPS4, NPS12)

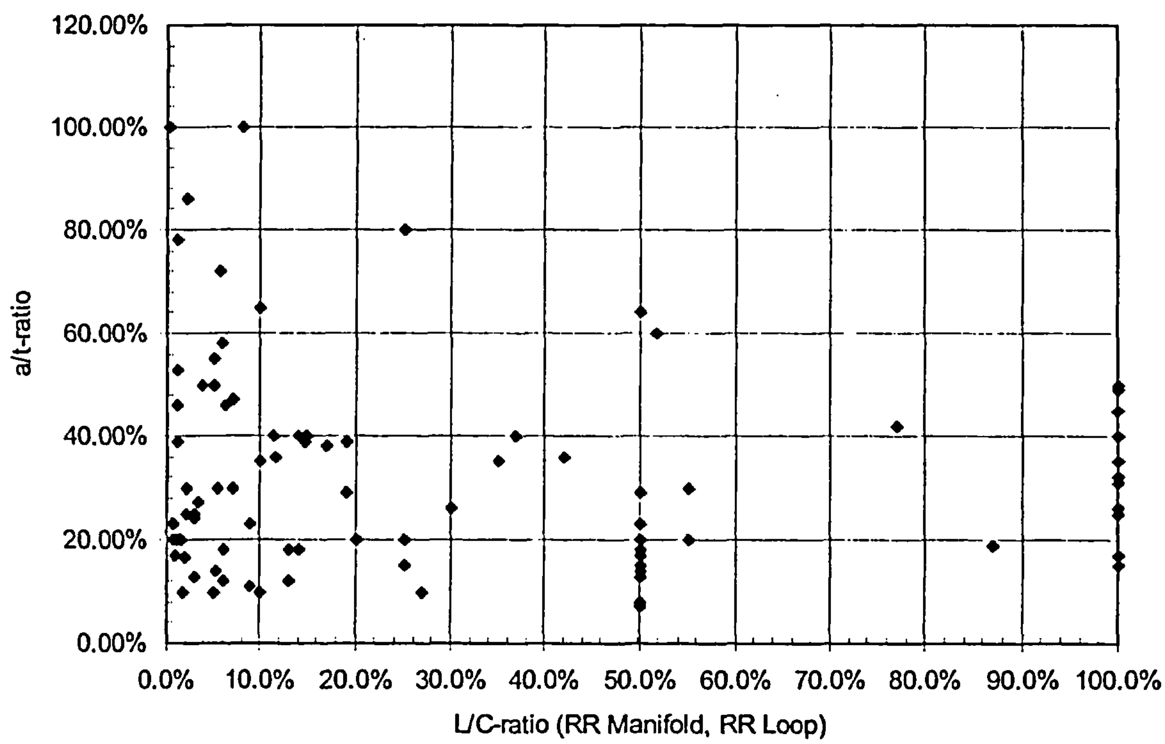


Figure D.7 Crack Depth Versus Crack Length in Austenitic Stainless Steel (NPS22, NPS28)

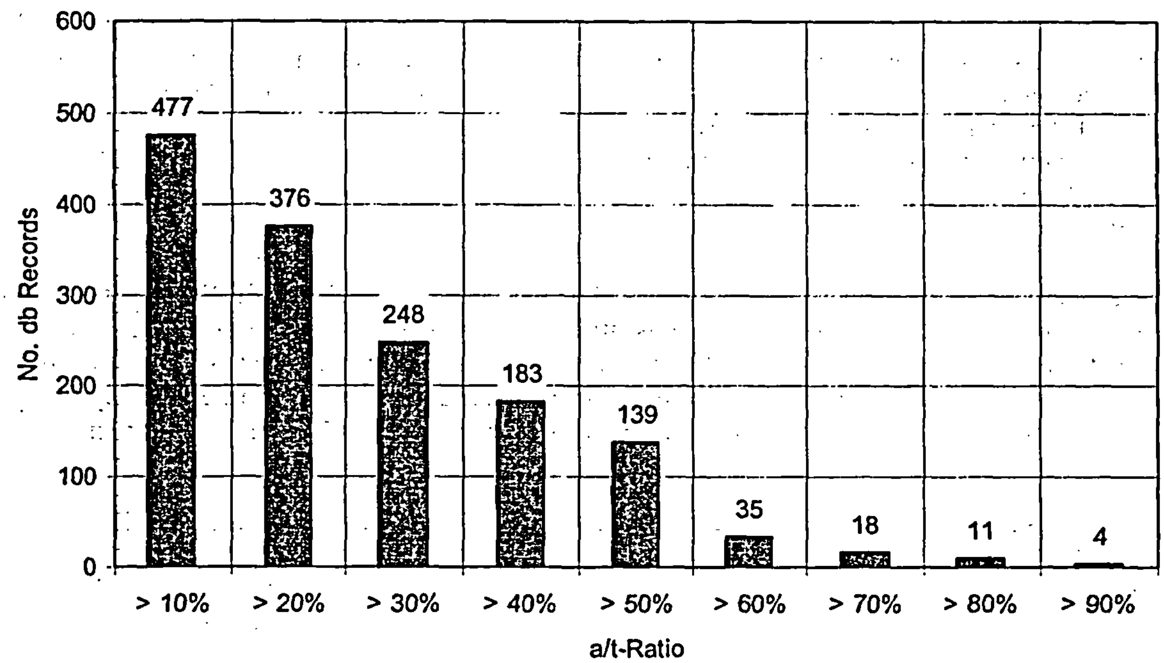


Figure D.8 Summary of IGSCC Non-Through Wall Flaws by 'a/t-Ratio'

In Figure D.5, the combination of a/t = 100% and L/C = 100% would indicate a case of DEGB where the pipe ends are separated from each other. As a rule-of-thumb, a through-wall crack (a/t = 100%) with L/C ≥ 40% is unstable and may exhibit unstable crack growth if it were to be left in place.[3]

As seen from the above, there have been a limited number of cases of leaks in large-diameter Reactor Recirculation piping. Only a small fraction of the total number of through-wall flaws have been active leaks; i.e., leaks that have developed during routine power operation. The majority of the through-wall flaws have been "non-active leaks." That is, leaks that have developed while shutting down for drywell inspection, during performance of weld crown grinding in preparation for ultrasonic examination ("ISI-leaks"), or during the performance of induction heat stress improvement (IHSI – "IHSI-leaks"). There are also some cases where leaks have been discovered during hydrostatic pressure testing to verify the integrity of weld repairs.

Like Figure D.5, Figure D. 8 includes data on all IGSCC-susceptible, Code Class 1 and 2 piping systems in BWR plants. While Figure D.5 includes approximately 300 data points, Figure D.8 includes on the order of 500 data points. This difference in the number of reports represented in respective chart is due to the fact that not all reports on IGSCC include complete details on the crack morphology (dimensions, orientation).

Where through-wall flaws have been observed leak rates have been small. In terms of leak rate and operational impact, so far the two most significant instances of IGSCC occurred at Duane Arnold in 1978 and at the Spanish plant Santa Maria de Garona in 1980. In the former case the leak rate was about 11 lpm (3 gpm) with L/C = 22%. In the latter case the observed leak rate was about 3.0 lpm (0.8 gpm) with L/C = 4.5%.

[3] See for example the report EPRI NP-2472 (The Growth and Stability of Stress Corrosion Cracks in Large-Diameter BWR Piping, July 1982).

D.3.2.2 FW Piping Service Experience - Figures D.9 and D.10 summarize the service experience with FW piping. With respect to plant designed by General Electric, the Code Class 1 portion of BWR carbon steel feedwater piping has performed well in the field. There are no reported leaks in medium-or large-diameter RCPB piping. Foreign plants have experienced (and in some cases, continue to experience) thermal fatigue damage due to thermal mixing and stratification. In fact, 80% of the degradation of the RCPB portions of FW piping has occurred in foreign plants with a piping system design that differs from that of U.S. BWR plants.

The U.S. service experience includes a few instances of non-through wall cracking of FW nozzle-to-safe-end (bimetallic) welds. The root cause of the cracking is attributed to weld defects from original construction. As documented in Information Notice 92-35 [D.19], Susquehanna Unit 1 has experienced flow-accelerated corrosion damage about 250 mm (10 inches) from a weld connecting NPS12 piping to a 20-inch by 12-inch reducing tee. There have been no reported flaws in any U.S. plant beyond T = 15 years of operation.

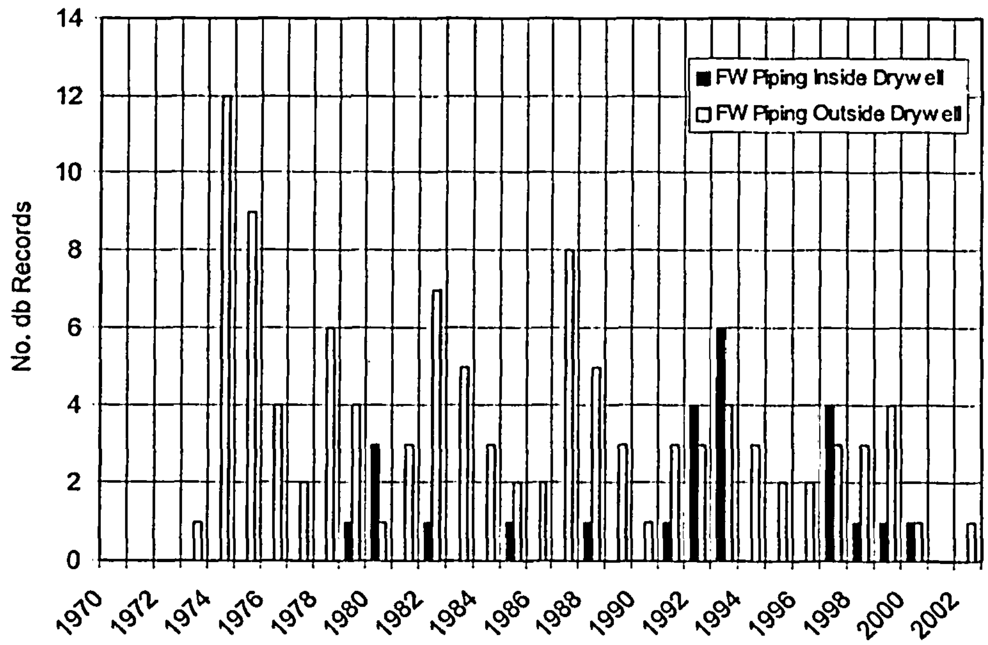


Figure D.9 Service Experience with FW Piping (i)

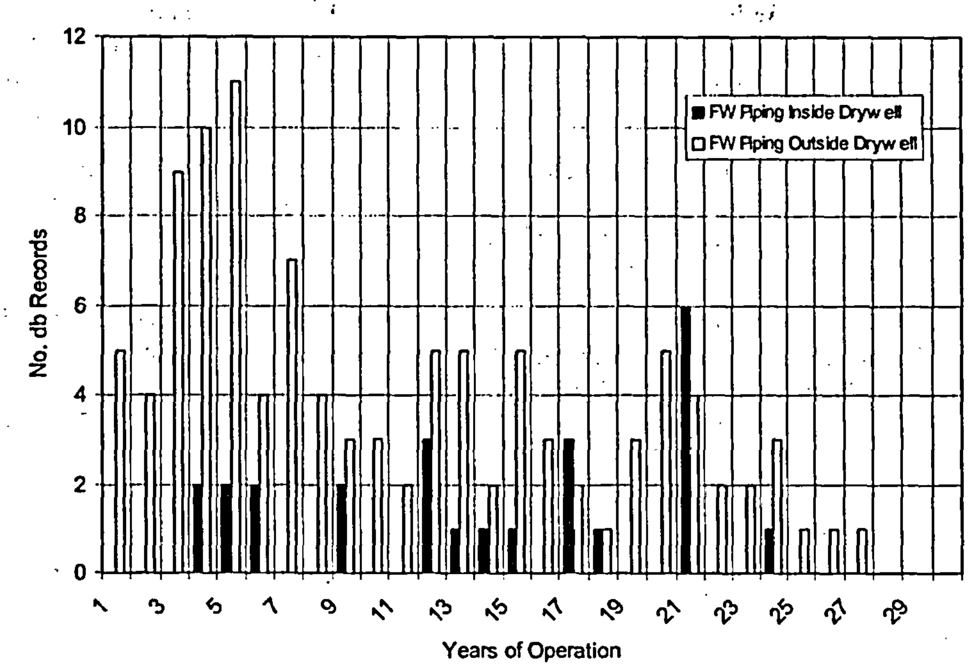


Figure D.10 Service Experience with FW Piping (ii)

D.3.3 Review of PWR-Specific Piping Service Experience

Limited to the PWR Base Case systems, this section summarizes the service experience with Code Class 1 piping. The results of this review are input to the pipe failure rate estimation.

D.3.3.1 RC & HPI/NMU Piping Service Experience - There have only been a limited number of events involving through-wall cracks in the large-diameter RC piping and the Class 1 portion of SI/CV piping. Evidence of axial primary water stress corrosion cracking (PWSCC) in the bimetallic safe-end to RPV nozzle welds of the RC-HL piping has been reported at Ringhals [D.20] and V.C. Summer [D.21].

During an eight-year period, the now decommissioned Trojan nuclear power plant experienced pressurizer surge line movement, which was attributed to thermal stratification [D.22]. In response, the NRC issued Bulletin 88-11 in December of 1988 [D.23] requesting that licensees perform visual inspections of the pressurizer surge line at the first available cold shutdown. Purpose of the inspections was to determine presence of any "gross discernible distress or structural damage in the entire pressurizer surge line, including piping, pipe supports, pipe whip restraints, and anchor bolts."

The current version (June 2004) of the PIPExp database includes four records associated with degradation of pressurizer surge lines:

- Record # 19849; during the Three Mile Island-1 2003 Refueling Outage (18-Oct-2003 to 3-Dec-2003), a UT examination found an axial flaw about 13 mm (0.5-inch) deep in the surge line nozzle-to-safe end interface in dissimilar metal weld No. SR0010BM. This weld connects a 10-inch Schedule 140, carbon steel nozzle to stainless steel safe end.
- Record # 19736; in November 2002 during UT examination of RC piping in the Belgian plant Tihange-2 (a 900 MWe series plant designed by Framatome), code rejectable indications were discovered in the 14-inch Inconel safe-end to nozzle weld. The flaw is believed to be an original construction defect.

- Record # 1119; while in hot shutdown condition, a non-isolable weld leak developed in a 1-inch drain line off the pressurizer surge line of Oconee-1 (LER 50-269/1998-002-01). The through-wall crack had initiated by TGSCC and propagated through-wall by vibratory fatigue. Small-diameter piping connecting to a pressurizer surge line is not part of the PWR-2 Base Case definition.
- Record # 420; during the 1988 annual refueling outage a pinhole leak was discovered in a 10-inch pressurizer surge line bi-metallic weld of Loviisa-1 (a Soviet designed WWER-440/213 plant located in Finland). The weld degradation was attributed to poor weld penetration and high residual stresses. This event was screened out from the data analysis.

Figures D.11 and D.12 summarize relevant service experience with medium- and large-diameter RC and safety injection (SI) and normal makeup (CV) piping. For comparison, Figure D.13 shows the service experience with small-diameter RC and SI/CV piping (≤ NPS2). Figure D.14 is a summary of the worldwide, PWR-specific data pipe failures that are attributed to thermal fatigue. In addition to RC-, SI- and CV-piping this figure includes failures in FW- and RHR-piping.

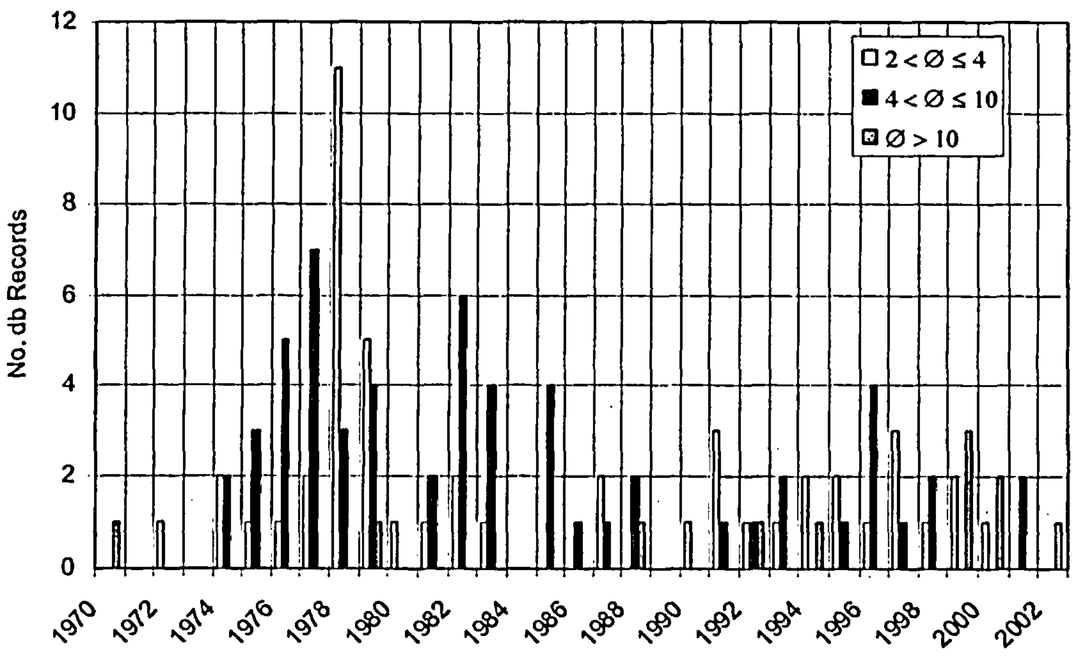


Figure D.11 Weld Failures in PWR RC-, CV- and SI-Piping (1970-2002)

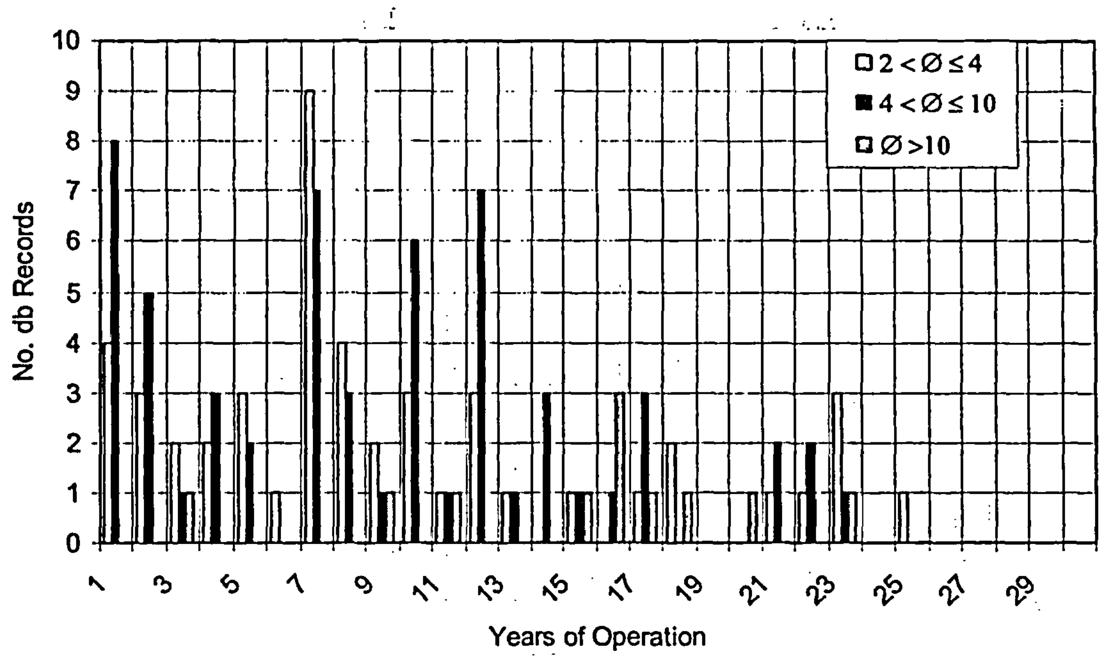


Figure D.12 Weld Failures in PWR RC-, CV- and SI-Piping (1970-2002)

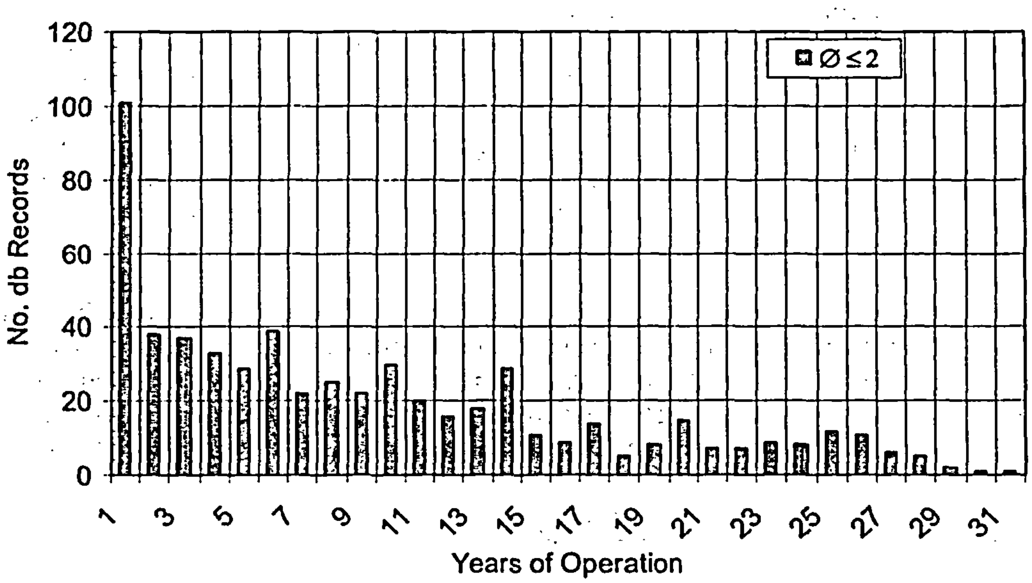


Figure D.13 Weld Failures in Small-Diameter PWR Piping

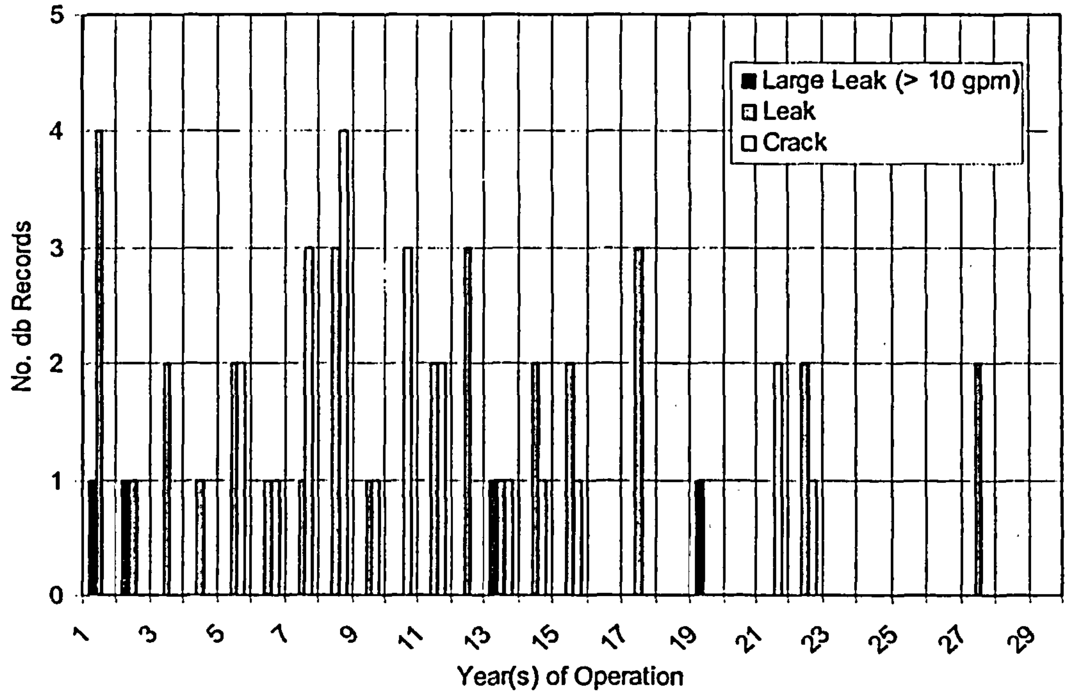


Figure D.14 Pipe Failures Attributed to Thermal Fatigue in PWRs Worldwide

Figure D.14 includes four significant (v > 10 gpm) events, three in foreign plants (Civaux-1 in France, Tsuruga-2 in Japan and Biblis-B in Germany) and one in a domestic plant (Oconee-2). The latter event involved a failure of a weld between the HPI/NMU and the RCS cold leg (= PWR Base Case Plant A.b). The plant operators correctly diagnosed the leak and brought the plant to safe shutdown. Subsequent to the weld failure in Oconee-2, limited to small-diameter piping the Electric Power Research Institute issued the "Interim Thermal Fatigue Guideline" [D.9] for evaluating and inspecting regions where there might be high potential for thermal fatigue cracking. Additional perspectives on thermal fatigue mitigation are included in an OECD-NEA report [D.24]. The Babcock &Wilcox-designed plants now include a new design thermal sleeve to mitigate or prevent thermal fatigue cracking of welds.

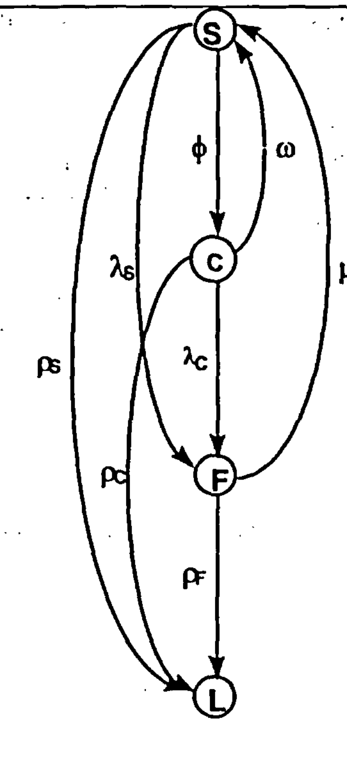
Prior to these 'four significant events', thermal fatigue damage occurred at Farley-2 and Tihange-1 (a Belgian plant) during 1988. At these plants, thermal fatigue initiated from cold water leaking through closed check or globe valves in safety injection lines. At Farley-2, the damage occurred in piping connected to the RCS cold leg, and at Tihange-1 in piping connected to the RCS hot leg. In these events the leak rates were 2.6 lpm (0.7 gpm) and 23 lpm (6 gpm), respectively. The U.S. NRC issued Bulletin 88-08 in response to these events.

D.4 Data Processing and Data Reduction

The objective of data processing is to extract from a pipe failure data collection relevant case histories that reflect specific combinations of reliability attributes and influence factors. Next, the data reduction prepares the input to the statistical parameter estimation in the form of event counts and exposure terms to develop Bayesian prior and posterior distributions.

D.4.1 Strategy for Data Processing and Data Reduction

Shown in Figure D.15 is a general four-state Markov model of piping reliability. All failure processes of this model can be evaluated using service data, assuming that such a data collection is of sufficient technical detail and completeness. This model is used in Section D.6 to develop time-dependent LOCA frequencies.



<u>Piping Reliability States</u>:

- <u>S</u> = Success (or undamaged state);
- <u>C</u> = Crack (non-through wall flaw);
- <u>F</u> = Leaking through-wall flaw (leak rate is within Technical Specification limit);
- <u>L</u> = Large leak (leak rate in excess, or well in excess of Technical Specification limit).

State Transitions:	
ϕ	Occurrence of non-through wall flaw
λ_C	Occurrence of small leak given a flaw ('C')
λ_F	Occurrence of large leak given a through-wall flaw ('F')
ρ_S	Occurrence of large leak given no flaw
ρ_C	Occurrence of large leak given a non-through wall flaw
ρ_F	Occurrence of large leak given a small leak
μ	Detect and repair a through-wall flaw
ω	Inspect and repair a non-through wall flaw

Figure D. 15 Four-State Markov Model of Piping Reliability[4]

In this and subsequent report sections, a pipe failure (F) is defined as a through-wall defect resulting in a non-active leak or small, active leak. The frequency of a large leak (L) in excess of Technical Specification limits is estimated using the following simple model:

$$F_L = \lambda \times p_{L|F} \tag{D.1}$$

$$\lambda = [No.\ Failures] / [Exposure\ (= T \times No.\ Components)] \tag{D.2}$$

Where:

F_L	=	Frequency of a large leak [1/Reactor-year].
λ	=	Failure frequency [1/Reactor-year.Extension]; where 'extension' refers to the piping component boundary definition. Depending on the intended application and type(s) of degradation mechanism, the extension could be formed by counts of bends, pipes, tees, welds or length of piping. In Equation (4.2), the exposure term reflects the total component population in the data survey.
T	=	Exposure time (or reactor operating years)
$p_{L\|F}$	=	Conditional probability of a large leak given a through-wall defect. Section D.5 includes a technical basis for estimating conditional failure probabilities.

[4] The figure is reproduced courtesy of K.N. Fleming (Technology Insights, Inc., San Diego, California).

The parameter estimation uses a Bayes' update process that begins with the development of prior distributions for each of the terms in equation (D.1). These prior distributions are shaped by our knowledge about the susceptibility of different piping systems to degradation. The input to the Bayes' update process comes from a small subset of PIPExp after it has been subjected to screening for pipe failures that meet certain selection criteria. A software tool (Bayesian Analysis Reliability Tool -BART™) is used to perform the updates.[5]

Piping reliability is a function of pipe size (diameter and wall thickness), and metallurgy, process medium, environment and design requirements; or attributes and influences, respectively. The purpose of data processing and data reduction is to extract from the total PIPExp database those subsets of service data that correspond to the attributes and influences of the Base Case definitions.

D.4.1.1 Informative versus Noninformative Prior Distributions - The type, extent and quality of applicable service data will determine the actual implementation of the Bayesian update process. Where sufficient service data is available an empirical Bayes approach is used. In this case classical estimation techniques are used to fit a prior distribution to the available data. When no or sparse service data is available a non-informative prior is defined. Relative to the five Base Cases the following approaches are used to determine the prior failure rates distributions:

- BWR Base Case – RR Loop B. There is ample service data on IGSCC. Our prior state-of-knowledge consists of service data before implementation of IGSCC mitigation strategies (mid-1980s). A prior failure rate is derived through classical statistical estimation.
- BWR Base Case – FW Loop B. Given the scarce service data, a lognormal distribution with a mean value of 1.0E-06 per weld-year and range factor (RF) equal to 100 is used. This is a noninfomative prior distribution.
- PWR Base Case – RC Hot Leg. The only available service data involves axial cracks in RPV nozzle-to-safe-end welds at three PWR units. A point estimate for the failure rates is calculated for the period 1970 through 2000. This point estimate is approximated by a lognormal distribution with range factor of 100; i.e., essentially a noninformative prior.
- PWR Base Case – RC Surge Line. For the pressurizer surge line there is no service data including non-through wall or through-wall cracking. Again, a lognormal distribution with a mean value of 1.0E-06 per weld-year and RF = 100 is used.
- PWR Base Case – HPI/NMU Line. Service data exists, which is directly applicable to this base case. To account for design changes that have been implemented post-1997, a non-informative prior is combined with B&W-specific failure data and exposure data through end of calendar year 1997. The resulting failure rate represents a prior distribution, which is applicable to this Base Case.

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[5] Details on Bayesian reliability analysis is found in text books on statistical analysis of reliability data; e.g., Martz and Waller (1991): Bayesian Reliability Analysis, Krieger Publishing Company, Malabar (FL), ISBN 0-89464-395-9. For conjugate functions like the gamma and beta distributions a Bayesian point estimator for the failure rate is the mean of respective posterior probability density function, or:

$\lambda = (\delta + r)/(\rho + T)$ – gamma

$\lambda = (\delta + r)/(\delta + \rho + n)$ – beta

Where, (δ, ρ) are the parameters of respective distribution and (r, T, n) correspond to new evidence (i.e., 'r' failures in 'T' hours, or 'r' failures in 'n' tests).

D.4.2 BWR-Specific Apriori Pipe Failure Rates

The failure rate development consists of determining an industry generic pipe failure rates for RR and FW piping, respectively. Next a Bayesian update is performed to generate failure rates that best represent the design and operating conditions of Plant B. Rather than taking an apriori failure from some published source, the approach in this study is to derive apriori failure information from the PIPExp database. The information that is summarized in Section D.3 provides some insights into the time-dependency of failure rates. These insights are explored further below.

D.4.2.1 RR Pipe Failure Data - Programs to mitigate the effects of certain degradation mechanisms strongly influence the achieved piping reliability. As an example, all BWR plants commissioned prior to the early to mid-1980s have experienced IGSCC. Industry initiatives to mitigate or eliminate the influence by IGSCC were implemented by the mid-1980s, and thereafter the rate of IGSCC has dropped sharply. The trend in the IGSCC rate is established by normalizing the data displayed in Figure D.2 (IGSCC by Years of Operation). Calculating the rate of IGSCC per weld-year for a given system and pipe size performs the normalization. Before performing this normalization the database is subjected to additional processing to exclude from further consideration any IGSCC data not directly applicable to the RR System that is representative of the Base Case. Similarly the part of the database including plant population and weld population data must be processed in such a way that an appropriate exposure term is developed commensurate with the failure data. In developing the RR-specific exposure term the following exclusion criteria were applied:

Plant Population Exclusion Criteria Applicable to RR Piping

- BWR plants without external recirculation loops;
- BWR plants in which the RR piping is fabricated from IGSCC resistant material e.g., Nuclear Grade stainless steel.

Table D.3 includes selected weld counts used to derive an exposure term according to Equation (D.2). Organized by pipe size and years of operation, Table D.4 is a summary of weld failures in RR piping. Noteworthy is the observation that there have been no reported through-wall defects in any plant beyond T = 15 years of operation. Using the information in Tables D.3 and D.4, Figure D.16 shows the calculated rate of RR pipe failure per weld-year.

The failure rates in Figure D.16 assume that all RR welds of a certain size to be equally susceptible to IGSCC. As was shown in SKI Report 98:30 [D.15], a correlation exists between weld failure rate and weld configuration. This correlation is assumed to be attributed to the piping layout, complexity of welding operation, and the associated weld residual stresses. The chart in Figure D.17 shows the weld configuration versus fraction of weld failures.

Table D.3 Selected Weld Counts in RR Piping

Plant ID (NSSS Type)	Weld Count by Pipe Size (NPS)								
	4	6	8	12	14	16	22	24	28
1 (AA/3)[6]	30	42	23	–	–	–	–	54	–
2 (BWR/4)	8	–	–	58	–	–	12	–	33
3 (BWR/4)	–	–	–	34	–	2	–	4	24
4 (BWR/5)	36	–	–	51	10	16	–	50	–
5 (BWR/5)	39	–	–	63	–	16	–	45	–
6 (BWR/5)	–	–	–	130	–	–	24	–	97
7 (BWR/5)	–	–	–	138	–	–	16	–	97
8 (BWR/4)	–	–	–	24	–	–	4	–	28
9 (BWR/4)	–	–	–	25	–	–	4	–	38
10 (BWR/4)	12	–	–	59	–	–	10	–	36
11 (BWR/4)	12	–	–	62	–	–	12	–	36
Plant B (BWR/4)	–	–	–	50	–	–	16	–	56
Mean:	23	–	–	63	–	–	12	–	49

Table D.4 Number of Through-Wall Flaws in RR Piping Attributed to IGSCC[7]

Pipe Diameter (Ø) [NPS]	Years of Operation															
	Total	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
3 < Ø ≤ 6	15	0	1	1	2	4	1	0	2	2	0	2	0	0	0	0
6 < Ø ≤ 12	9	0	0	0	0	1	0	1	0	1	1	2	1	1	0	1
12 < Ø ≤ 22	4	0	0	0	0	0	0	1	0	0	0	2	1	0	0	0
Ø > 22	16	0	0	0	0	0	0	0	0	5	0	6	4	0	1	0
	44	0	1	1	2	5	1	2	2	8	1	12	6	1	1	1

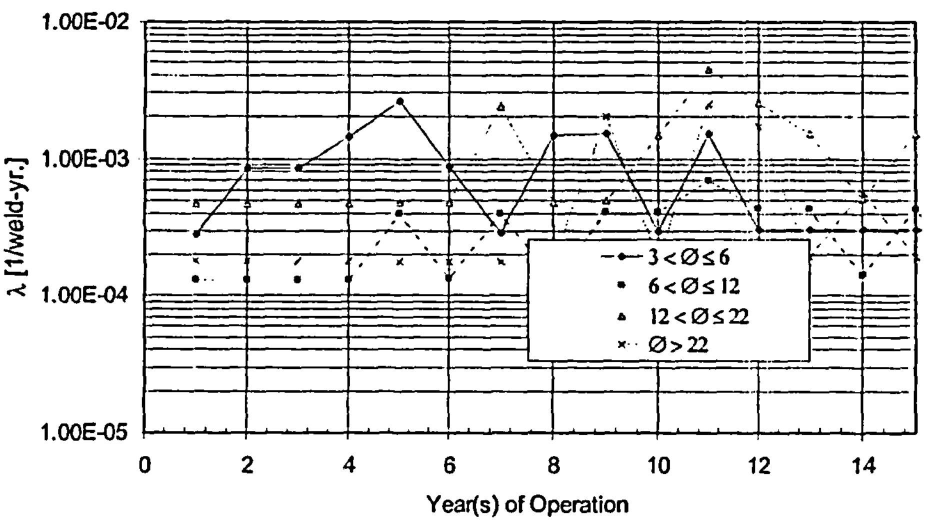

Figure D.16 Rate of IGSCC-Induced RR Pipe Failure ('Prior State-of-Knowledge')

[6] See SKI 98:30 [D.15] for details.

[7] This table includes active leaks (= leaks detected during routine power operation) and 'non-active' leaks (= leaks discovered during change of plant mode of operation), but it excludes 'ISI-leaks.' Appendix A, Table A-5 includes details on the through-wall cracks in NPS12, NPS22 and NPS28 Reactor Recirculation piping as included in Table D.4 above.

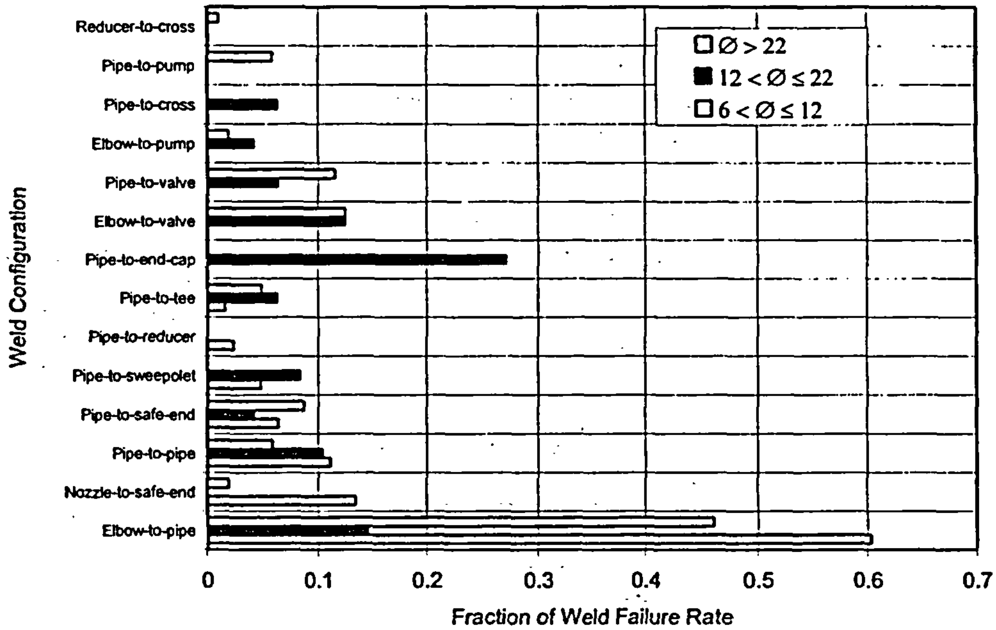

Figure D.17 RR Weld Failures as a Function Weld Configuration

In addition to weld configuration, the likelihood of failure is also a function of pipe size. For a weld of type "i" and size "j" the failure rate is expressed as follows:

$$\lambda_{ij} = F_{ij}/(W_{ij} \times T) \qquad \text{(D.3)}$$

and with

$$S_{ij} = F_{ij} / F_j \qquad \text{(D.4)}$$
$$A_{ij} = W_j/W_{ij} \qquad \text{(D.5)}$$

the failure rate of weld of type "i" and size "j" is expressed as

$$\lambda_{ij} = (F \times S_{ij})/(W_{ij} \times T) \qquad \text{(D.6)}$$
$$\lambda_{ij} = S_{ij} \times A_{ij} \times \lambda_j \qquad \text{(D.7)}$$

Where:

λ_{ij}	=	Failure rate of an IGSCC-susceptible weld of type "i", size "j"
λ_j	=	Failure rate of an IGSCC susceptible weld of size 'j'
F_j	=	Number of size "j" weld failures
F_{ij}	=	Number of type "i" and size "j" weld failures
W_j	=	Size "j" weld count
W_{ij}	=	Type "i" and size "j" weld count
Susceptibility (S_{ij})	=	The service experience shows the failure susceptibility to be correlated with the location of a weld relative to pipe fittings and other in-line components (flanges, pump casings, valve bodies). For a given pipe size and system, the susceptibility is expressed as the fraction of welds of type "ij" that failed due to a certain degradation mechanism). This fraction is established from the PIPExp database; Table D.5.
Attribute (A_{ij})	=	In the above expressions the attribute (A) is defined as the ratio of the total number of welds of size "j" to the number of welds of type "i". In expression (4.7), A_{ij} is a correction factor and accounts for the fact that piping system design & layout constraints impose limits on the number of welds of a certain type. For example, in a given system there tends to be more elbow-to-pipe welds than, say, pipe-to-tee welds.

Combining the global (or averaged) failure rates in Figure D.16 with the information summarized in Figure D.17 and Table D.5 provides the apriori failure rates that are input to Equation (D.1). The results are summarized in Section D.4.3.

Table D.5 IGSCC Susceptibility by Weld configuration – Selected Parameter Values

RR System [NPS]	Weld Configuration	Configuration Dependent Parameters	
		Susceptibility (S_{ij})	Attribute (A_{ij})
12	Elbow-to-pipe	6.03E-01	2.8
	Nozzle-to-safe-end	1.35E-01	5.0
	Pipe-to-reducer	2.38E-02	25.0
22	Pipe-to-end-cap	2.71E-01	4.0
	Pipe-to-sweepolet	8.33E-02	2.0
	Pipe-to-cross	6.25E-02	4.0
28	Elbow-to-pipe	4.62E-02	5.6
	Pipe-to-pipe	5.77E-02	3.1
	Cross-to-reducer	9.60E-03	28.8

D.4.2.2 FW Pipe Failure Data - The estimation of failure rates for FW piping uses a non-informative prior distribution together with the weld population data in Table D.6. This approach is selected based on the available, limited service experience with ASME XI Class 1 FW piping; Table D.7. In developing the data summary in Table D.7 the following FW exclusion criteria were used to develop a point estimate of the failure rate:

<u>Failure Data Exclusion Criteria Applicable to FW Piping</u>

- Piping external to the drywell containment structure;
- Non-US data.

Table D.6 Selected Weld Counts in ASME XI Class 1 FW Piping

Plant ID (NSSS Type)	Weld Count by Pipe Size (NPS)								
	8	10	12	14	16	18	20	22	24
1 (AA/3)	1	61	–	14	–	–	–	–	–
2 (BWR/4)	–	–	28	–	–	25	–	–	–
3 (BWR/4)	–	–	28	–	–	26	–	–	–
4 (BWR/5)	–	–	40	12	–	6	–	–	38
5 (BWR/5)	–	–	41	6		6	–	–	42
6 (BWR/5)	13	–	66	–	8	–	7	–	22
7 (BWR/5)	9	–	68	–	8	–	7	6	19
9 (BWR/4)	–	–	50	–	–	–	6	–	45
10 (BWR/4)	–	–	32	1	–	30	–	–	–
11 (BWR/4)	–	–	30	1	–	29	–	–	–
Plant B (BWR/4)	–	–	63	5	–	–	53	–	–
Mean:	–	–	42	–	–	–	41[8]	–	–

Table D.7 Summary of FW Pipe Failure Data

Pipe Size (Ø) [NPS]	Location		Data Origin		Failure Mode			
	Drywell	Ex-drywell	US	Non-US	Crack / Wall Thinning	P/H-leak	Leak	Rupture
3 < Ø ≤ 6	YES		YES		0	0	0	0
6 < Ø ≤ 12	YES		YES		4	0	0	0
Ø > 12	YES		YES		1	0	0	0
3 < Ø ≤ 6	YES			YES	5	0	0	0
6 < Ø ≤ 12	YES			YES	9	0	0	0
Ø > 12	YES			YES	5	0	0	0
3 < Ø ≤ 6		YES	YES		1	2	19	3
6 < Ø ≤ 12		YES	YES		1	1	7	0
Ø > 12		YES	YES		2	0	3	0
3 < Ø ≤ 6		YES		YES	3	0	1	0
6 < Ø ≤ 12		YES		YES	2	0	2	0
Ø > 12		YES		YES	0	1	1	0

D.4.3 PWR-Specific Apriori Pipe Failure Rates

As summarized in Section D.3.3, there have been only a few through-wall defects in Class 1 PWR piping. For the RC-HL, the rate of PWSCC per weld-year is established using the normalization process discussed in Section D.4.2.1 and with the following specializations:

RC-HL Apriori Failure Rate

- The apriori failure rate is derived using the PIPExp for the time-period 1970 through 2000 to include the consideration of the through-wall defect at V.C. Summer.
- Table D.8 includes the weld population data used to calculate λ_{NPS30} = 8.12E-05 per weld-year.

[8] The mean of weld count in NPS20-, 22- and 24-piping.

- Failure rate "post-processing" to account for different weld configuration susceptibilities to PWSCC is done consistent with Section D.4.2 and with the S_{ij} and A_{ij} assumed values shown in Table D.9.

Table D.8 Selected Weld Counts in Code Class 1 PWR Piping

Plant ID (NSSS Type)	Weld Count by Pipe Size [NPS]				
	3-¾	10	12	14	30[9]
1 (WEST/4)	–	24	18	1	84
2 (WEST/4	–	24	18	1	52
3 (WEST/4)	–	24	18	–	92
4 (WEST/4)	–	24	14	–	68
Plant A.a (WEST/3)	–	–	5	14	50
Plant A.b (B&W; HPI/NMU system only)	9	–	–	–	–

Table D.9 Degradation Susceptibility by Weld Configuration

RC System (NPS)	Weld Configuration	Configuration Dependent Parameters	
		Susceptibility (S_{ij})	Attribute (A_{ij})
30 (RC Hot Leg)	Nozzle-to-safe-end	8.00E-01	12.5
	Elbow-to-safe-end	8.00E-02	12.5
	Elbow-to-pump	5.00E-02	12.5
	Pipe-to-pump	4.00E-02	12.5
	Elbow-to-pipe	3.00E-02	1.5
14 (Surge Line)	Nozzle-to-safe-end	5.00E-01	14.0
	Pipe-to-safe-end	2.50E-01	14.0
	Branch-to-pipe	5.00E-02	14.0
	Branch-to-HL	1.50E-01	14.0
	Elbow-to-pipe-	5.00E-02	1.40
3-¾ (HPI/NMU)	Elbow-to-nozzle	8.50E-01	9.0
	Elbow-to-pipe	1.00E-01	2.25
	Elbow-to-valve	4.50E-02	3.0
	Pipe-to-pipe	5.00E-03	9.0

In contrast to the BWR weld susceptibility factors in Table D.5, the weld susceptibility factors in Table D.10 are assumed values that reflect the applicable service experience. As an example, for the RC Hot Leg the nozzle-to-safe-end weld is assigned the highest value in view of the available service experience; i.e., the Ringhals and V.C. Summer hot leg cracking as described in Section D.3.3.1. As another example, for the RC Surge Line, relatively high weld susceptibility factors are assigned the safe-end welds and Hot Leg branch connection. In view of the recent experience at TMI-1, the nozzle-to-safe-end weld is given a greater weight than other weld configurations. The uncertainty in the PWR weld susceptibility factors is not evaluated further in this study, however. EPRI TR-111880[10] is used for characterizing the prior knowledge about pipe failure due to thermal fatigue.

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[9] NPS30 is used to characterize the CL- and HL-piping.

[10] Table 2-3, page 2-10; λ_j = 1.34E-05 (RF = 100). TR-111880: Piping System Failure Rates and Rupture Frequencies for Use in Risk Informed In-Service Inspection (September 1999).

D.4.4 Prior Distributions for Bayesian Updating

Included for comparison, Figure D.18 shows the calculated flaw rates (non-through wall). Listed in Table D.10 are the RR and FW weld failure rates that represent the state-of-knowledge at T = 15 years of operation. Listed in Table D.11 are the RC and HPI/NMU prior weld failure rates. The failure rates represent the frequency per weld-year of a through-wall flaw resulting in a leakage of less than or equal to the Technical Specification limit for unidentified leakage. In summary, the derivation of prior weld failure rates includes the following steps:

- Determine the number of through-wall leaks from PIPExp database. Includes performing a trend analysis.
- From the PIPExp database, determine the appropriate exposure terms.
- Establish the susceptibility of different weld locations to degradation.
- Combine the output from the previous steps to determine prior failure rates applicable to welds in the RR and FW systems.

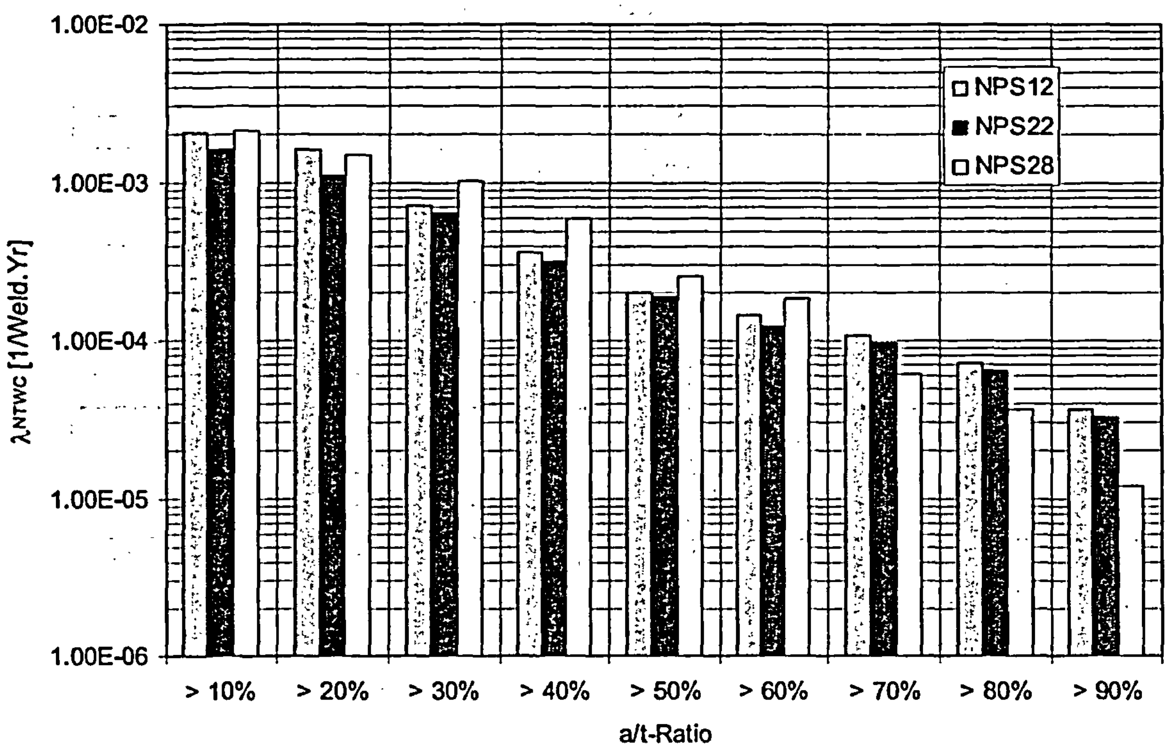

Figure D.18 Prior Frequency of Non-Through Wall IGSCC in RR Piping

Table D.10 Prior RR and FW Weld Failure Rates

System	Pipe Size [NPS]	Weld Configuration	Lognormal Distribution Parameters	
			Mean [1/Weld-yr]	Range Factor (RF)
RR	12	Elbow-to-pipe	3.95E-04	10
		Nozzle-to-safe-end	1.59E-04	10
		Pipe-to-safe-end	7.49E-05	15
		Pipe-to-sweepolet	7.02E-05	15
		Pipe-to-reducer	1.40E-04	15
RR	22	Pipe-to-end-cap	7.67E-04	10
RR	22	Pipe-to-sweepolet	1.18E-04	15
		Pipe-to-cross	1.77E-04	15
RR	28	Pipe-to-elbow	7.19E-04	10
		Nozzle-to-safe-end	1.50E-04	10
		Pipe-to-safe-end	6.74E-04	10
		Pipe-to-valve	2.25E-04	10
		Pipe-to-pump	2.25E-04	10
		Pipe-to-tee	1.25E-04	10
		Pipe-to-pipe	4.99E-05	20
		Pipe-to-cross	7.49E-05	20
		Pipe-to-reducer	7.49E-05	20
FW	12	Nozzle-to-safe-end	6.20E-06	100
		Elbow-to-pipe	3.32E-07	100
		Pipe-to-pipe	5.17E-07	100
		Pipe-to-reducer	1.55E-06	100
		Pipe-to-reducing-tee	4.65E-07	100
FW	20	Elbow-to-pipe	4.00E-06	100
		Elbow-to-valve	6.00E-07	100
		Pipe-to-reducing-tee	4.80E-07	100
		Pipe-to-reducer	7.20E-07	100
No failure rates derived for welds in NPS14 piping; it is assumed that the data on welds in NPS12 piping is also representative of welds in NPS14 piping.				

Table D.11 Prior RC and HPI/NMU Weld Failure Rates

System	Pipe Size [NPS]	Weld Configuration	Lognormal Distribution Parameters: Mean [1/Weld-yr]	RF
RC (Hot Leg)	30	Nozzle-to-safe-end	8.12E-04	100
		Elbow-to-pump	5.07E-05	100
		Pipe-to-pump	4.06E-05	100
		Elbow-to-pipe	3.65E-06	100
RC (Surge Line)	14	Branch-to-HL	2.10E-06	100
		Nozzle-to-safe-end	7.00E-06	100
		Pipe-to-safe-end	3.50E-06	100
		Branch-to-pipe	7.00E-07	100
		Elbow-to-pipe	7.00E-08	100
HPI/NMU	3-¾	Elbow-to-nozzle	9.86E-04	16
		Pipe-to-pipe	6.48E-06	100
		Elbow-to-pipe	1.90E-06	100
		Elbow-to-valve	1.31E-06	100

Note to Data on Pressurizer Surge Line:
- Susceptibility factors from Table D.9 applied directly to prior knowledge.

Notes to Data on HPI/NMU Line:
- Susceptibility factors from Table D.9 applied to prior knowledge.
- To develop a B&W-specific weld failure rate, the prior knowledge (data from TR-111880; see footnote #18) was combined with B&W-specific service data and exposure data from PIPExp through end of calendar year 1997. The failure rate for weld type 'elbow-to-nozzle' accounts for the weld failure at Oconee-2 in 1997.

D.5 Data for LOCA Frequency Estimation

Using the information in Section D.4, this section documents the input data to the LOCA frequency model. A Bayesian update is performed to develop posterior weld failure rates. The frequency of leaks exceeding Technical Specification limits are developed through estimates of the conditional probability of a large leak given a small through-wall flaw.

D.5.1 Posterior Weld Failure Rates

The failure rate calculation involves two factors, the number of applicable failures and the exposure data. To account for uncertainty in the exposure data, which could influence the failure rate calculation the following process is used. First, a best estimate update is performed using the appropriate number of failure events and the number of welds of exposure. To account for plant-to-plant variability in the weld exposure term, a second update is performed using the same failure data but an exposure estimate that is 50% higher, and a third update using an exposure estimate that is 50% lower. Each of the three updates is combined in a posterior weighting process using the following weights: 50% for the best estimate, 25% for the high exposure case and 25% for the low exposure case. The result is an uncertainty distribution for each failure rate, which reflects greater uncertainty than the best estimate data would imply alone. The results are given in Tables D.12 and D.13; Attachment B includes the input to the failure rate calculation. Figure D.19 displays posterior IGSCC flaw frequencies. Figures D.20-D.23 compare the prior and posterior non-through wall crack frequencies.

Table D.12 Posterior RR and FW Weld Failure Rate Distributions – BWR Base Cases

System	Pipe Size [NPS]	Weld Configuration	Failure Rate Uncertainty Distribution Parameters [(≤TS Leak)/Weld-yr] Mean	5%-tile	50%-tile	95%-tile
RR	12	Elbow-to-pipe	4.32E-05	8.48E-06	3.17E-05	1.16E-04
		Nozzle-to-safe-end	4.38E-05	5.52E-06	2.72E-05	1.36E-04
		Pipe-to-safe-end	2.99E-05	2.98E-06	1.70E-05	9.64E-05
		Pipe-to-sweepolet	3.14E-05	2.80E-06	1.71E-05	1.06E-04
		Pipe-to-reducer	7.82E-05	5.71E-06	3.97E-05	2.77E-04
RR	22	Pipe-to-end-cap	1.54E-04	2.28E-05	1.01E-04	4.52E-04
		Pipe-to-cross	4.24E-05	4.38E-06	2.47E-05	1.37E-04
		Pipe-to-sweepolet	7.37E-05	7.02E-06	4.09E-05	2.40E-04
RR	28	Pipe-to-elbow	8.52E-05	1.59E-05	6.07E-05	2.33E-04
		Nozzle-to-safe-end	6.55E-05	5.95E-06	3.61E-05	2.15E-04
		Pipe-to-safe-end	1.44E-04	2.11E-05	9.36E-05	4.28E-04
		Pipe-to-valve	5.96E-05	7.68E-06	3.75E-05	1.84E-04
		Pipe-to-pump	8.36E-05	8.68E-06	4.85E-05	2.71E-04
		Pipe-to-tee	5.78E-05	5.06E-06	3.13E-05	1.96E-04
		Pipe-to-pipe	1.29E-05	5.74E-07	5.25E-06	4.78E-05
		Pipe-to-cross	3.86E-05	7.89E-07	1.08E-05	1.50E-04
		Reducer-to-cross	3.86E-05	7.89E-07	1.08E-05	1.50E-04
FW	12	Nozzle-to-safe-end	2.29E-06	8.61E-10	6.88E-08	5.29E-06
		Elbow-to-pipe	1.75E-07	4.61E-11	4.28E-09	3.74E-07
		Pipe-to-pipe	2.78E-07	7.39E-11	6.89E-09	6.33E-07
		Pipe-to-safe-end	2.43E-07	6.20E-11	5.97E-09	5.50E-07
		Pipe-to-reducer	9.73E-07	2.38E-10	2.24E-08	2.12E-06
FW	12	Elbow-to-reducing-tee	3.33E-07	7.46E-11	7.11E-09	6.94E-07
FW	20	Pipe-to-elbow	1.62E-06	5.57E-10	4.61E-08	3.71E-06
		Pipe-to-pipe	4.10E-07	9.57E-11	8.77E-09	8.00E-07
		Pipe-to-valve	3.38E-07	7.09E-11	6.78E-09	6.53E-07
		Elbow-to-valve	3.54E-07	9.22E-11	8.80E-09	8.39E-07
		Pipe-to-tee	4.50E-07	7.33E-11	7.20E-09	7.25E-07
		Pipe-to-reducer	5.68E-07	1.17E-10	1.14E-08	1.14E-06
		Tee-to-valve	2.18E-07	4.06E-11	4.12E-09	4.06E-07

Table D.13 Posterior RC and HPI/NMU Weld Failure Rate Distributions – PWR Base Cases

System	Pipe Size [NPS]	Weld Configuration	Failure Rate Uncertainty Distribution Parameters [(≤TS Leak)/Weld-yr] Mean	5%-tile	50%-tile	95%-tile
RC (Hot Leg)	30	Nozzle-to-safe-end	7.64E-05	2.12E-07	7.34E-06	2.61E-04
		Elbow-to-pump	1.96E-05	7.59E-09	5.36E-07	4.02E-05
		Pipe-to-pump	1.24E-05	5.78E-09	4.47E-07	3.17E-05
		Elbow-to-pipe	1.05E-06	5.38E-10	3.94E-08	3.04E-06
RC (Surge Line)	14	Branch-to-CL	1.14E-06	2.90E-10	2.69E-08	2.42E-06
		Nozzle-to-safe-end	2.95E-06	9.94E-10	7.99E-08	6.49E-06
		Pipe-to-safe-end	1.75E-06	4.72E-10	4.00E-08	3.64E-06
		Branch-to-pipe	4.76E-07	1.04E-10	1.04E-08	9.87E-07
		Elbow-to-pipe	4.60E-08	1.06E-11	1.04E-09	9.66E-08
HPI/ NMU	3-¼	Pipe-to-safe-end	6.56E-04	1.45E-05	1.99E-04	2.53E-03
		Elbow-to-pipe	1.58E-06	3.30E-10	3.43E-08	3.39E-06
		Pipe-to-valve	1.96E-06	2.36E-10	2.39E-08	2.35E-06
		Pipe-to-pipe	4.55E-06	1.13E-09	1.13E-07	1.11E-05

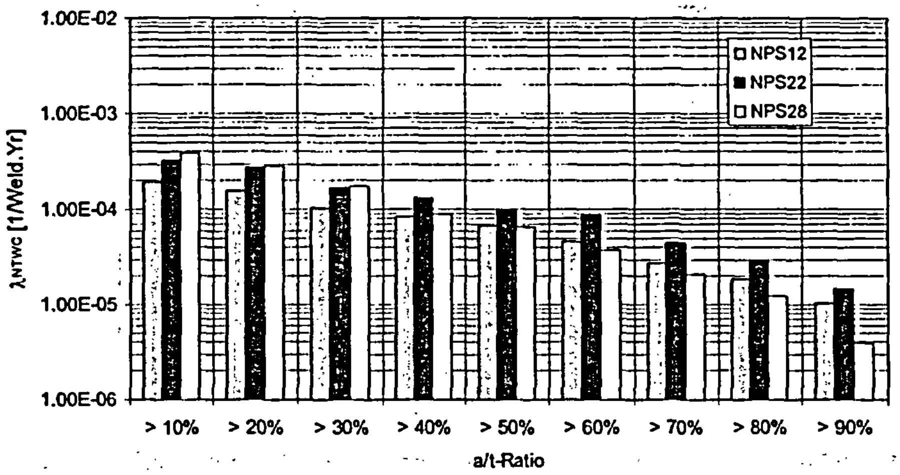

Figure D.19 Posterior IGSCC Frequency (Non-Through Wall)

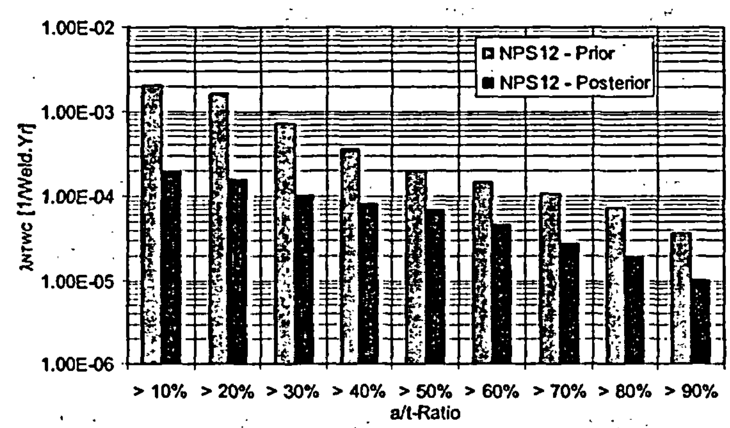

Figure D.20 Prior and Posterior IGSCC Frequency (Non-Through Wall) for NPS12 Welds

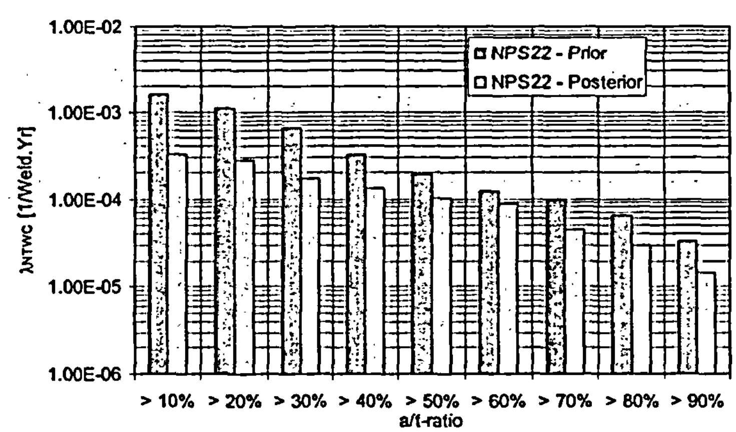

Figure D.21 Prior and Posterior IGSCC Frequency (Non-Through Wall) for NPS22 Welds

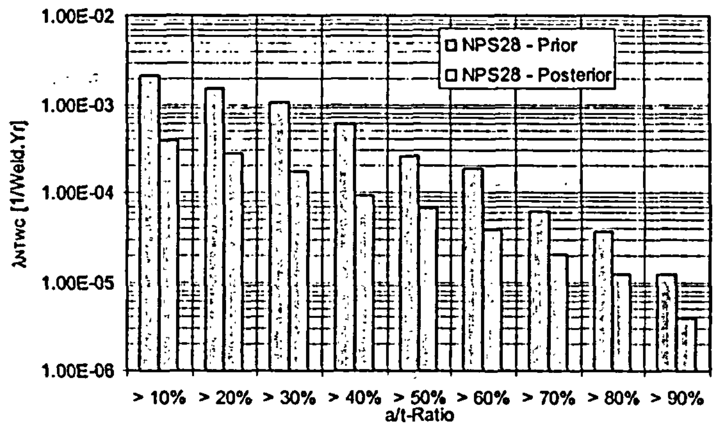

Figure D.22 Prior and Posterior IGSCC Frequency (Non-Through Wall) for NPS28 Welds

D.5.2 Conditional Weld Failure Probability

This section develops conditional failure probabilities for a "large leak," which is defined as a through-wall flaw with a leak/flow rate (ν) greater than 38 lpm (10 gpm) but less than or equal to 380 lpm (100 gpm). The process for developing conditional failure probabilities starts by deriving a point-estimate of p_{LIF} for small-diameter (≤ NPS1) stainless steel piping given presence of susceptibility to stress corrosion cracking (SCC). This point estimate is based on Jeffreys noninformative prior and service data from PIPExp. According to the latter, there are 42 through-wall flaws and zero large leaks in NPS1 BWR piping. This gives a point estimate of 1.16E-02, which is used as a constraint when developing conditional failure probabilities of medium- and large-diameter piping. The relationship between pipe size (diameter and wall thickness) and the conditional failure probability is assumed to follow a power law of the form:

$$p_{LIF} = a \times DN^{b} \qquad (D.8)$$

Where, DN is the nominal pipe size in [mm]. Decreasing trends correspond to negative values of b. Parameters 'a' and 'b' are determined for p_{LIF} = 1.16E-02 and DN25. Point estimates of p_{LIF} for other pipe sizes are shown in Figure D.23. The power law is assumed to be applicable to the RC-HL piping given circumferential cracking.

Unlike SCC, the service experience associated with piping susceptible to thermal fatigue damage indicates that a defect can rapidly propagate in the through-wall direction. Therefore, the conditional failure probabilities that are developed for SCC do not apply to thermal fatigue. Based on existing thermal fatigue data as recorded in PIPExp for BWR and PWR, a point estimate of p_{LIF} for small-diameter piping is on the order of 2.0E-02. This value is used as the constraint on the power law relationship for thermal fatigue.

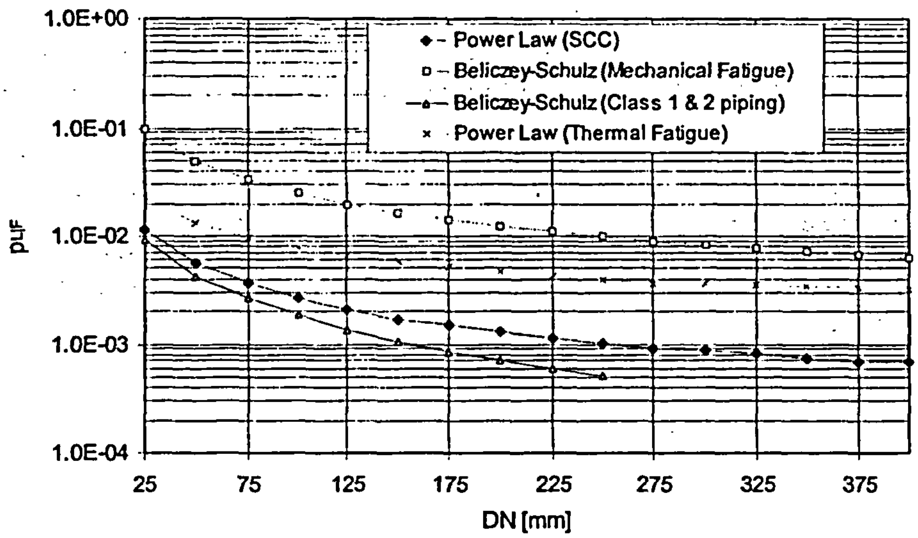

Figure D.23 Conditional Probability of Large Leak (= Cat 0 LOCA) Given a Through-Wall Flaw[11]

In Figure D.23 the power law model is compared with the Beliczey-Schulz (VF) correlation, which applies to piping subjected to mechanical fatigue. For each point estimate developed using the power law model a probability distribution is developed by using a Beta function (Table D.14). Because these Beta distributions are to be updated using Bayesian principles, they are constructed to be constrained non-informative priors. For SCC the A-parameter of each distribution is fixed at 1.0, and for thermal fatigue the A-parameter is fixed at 2.0. The B-parameter is selected such that the mean of the distribution is the point estimate. Fixing the A-parameter at 1.0 for SCC is assumed on the basis of the insights from the review of the service data (i.e., 0 large leaks in about 1,992 reactor-years and no large leaks are expected in the foreseeable future).

Table D.14 Prior Beta Parameters (Constrained Noninformative Prior)

Degradation Mechanism	Pipe Size		Parameter B in Beta Prior
	DN	NPS	
SCC (A = 1)	300	12	1262
	550	22	1496
	700	28	1700
Thermal Fatigue (A = 2)	90	3-¾	227
	350	14	592

For thermal fatigue, fixing the A-parameter at 2.0 is assumed in view of the service data, which includes large leak events. The service experience data indicates that flaws that are attributed to thermal fatigue tend to propagate at a faster rate in the through-wall direction SCC-induced flaws.

D.5.3 Conditional Failure Probability and Flow Rate

The conditional failure probabilities derived in the previous section are assumed applicable to Cat0 LOCA. It is furthermore assumed that for a *significant* primary piping breach to occur there has to be a through-wall flaw coinciding with a plant operational mode change or an unusual or severe loading condition such that the leakage exceeds a Cat0 LOCA. The service data collection (e.g., PIPExp) includes numerous examples where pressure pulses or spikes caused by changing flow conditions following a plant operational mode change

[11] Details on "Beliczey-Schulz (Class 1 & 2 Piping) are found in Nuclear Engineering and Design (102:431-438 and 110:229-232).

have resulted in non-active leaks[12] becoming active leaks. The physics of such transitions from non-active to active leaks are complex and location-dependent (e.g., function of flaw size and pipe stresses). Some published work exists on the correlation between crack propagation and plant transient history [D.25]. Using available empirical data, the uncertainties in such crack growth assessments are considerable, however.

In this analysis a simple parametric approach is applied to the estimation of weighted conditional failure probabilities (C_L) of a pressure boundary breach that exceeds a Cat0 flow rate threshold value. This approach is described through the event tree in Figure D.24. An undetected, or detected but monitored through-wall is exposed to a pressure pulse or unusual loading condition before a decision to perform manual, controlled reactor shutdown. The pressure pulse or unusual loading condition is characterized as a subjectively defined probability distribution.

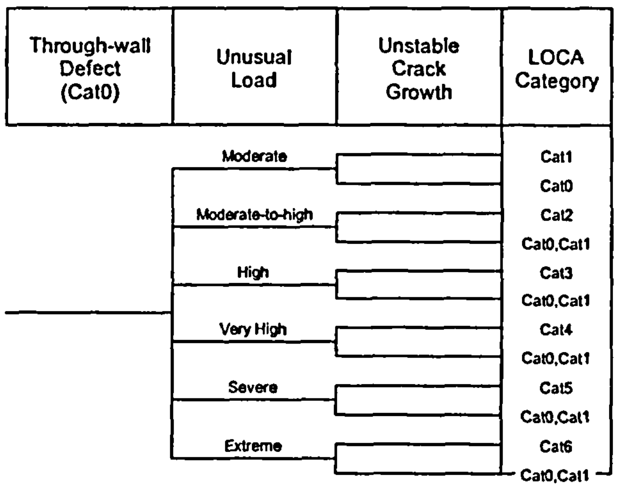

Figure D.24 Event Tree for Definition of LOCA Categories

In the cases of "moderate-to-high" to "extreme", the term "unusual" implies a loading condition beyond that resulting from anticipated transients including manual and automatic reactor/turbine trips. The conditional probability of an unusual or severe loading condition is described by five sets of subjective 3-bin discrete and overlapping probability distributions as summarized in Table D.15. These DPDs are combined with the weld failure rate distributions and conditional weld failure probability distributions by using a Monte Carlo merge technique.

Table D.15 Probability of LOCA Given Severe Overloading

Category	Flow Rate (v) Intervals [gpm]	DPD for Severe Loading					
		$C_{L\text{-}High}$	$C_{L\text{-}Med}$	$C_{L\text{-}Low}$	p_{High}	p_{Med}	p_{Low}
0	$10 < v \leq 100$	N/A	N/A	N/A	N/A	N/A	N/A
1	$100 < v \leq 1500$	.80	.50	.20	.2	.6	.2
2	$1500 < v \leq 5000$	.32	.20	.08	.2	.6	.2
3	$5000 < v \leq 25{,}000$	.13	.08	.03	.2	.6	.2
4	$25{,}000 < v \leq 100{,}000$	.05	.03	.01	.2	.6	.2
5	$100{,}000 < v \leq 500{,}000$	.02	.01	.005	.2	.6	.2
6	$v > 500{,}000$	.01	.005	.002	.2	.6	.2

[12] The term 'non-active leak' is taken to mean a through-wall flaw without visible leakage or with a small, detectable leakage that stays relatively constant over time.

Service data on water hammer events provides a justification for the chosen DPDs. From PIPExp, a point estimate for $C_{L\text{-}WH\text{-}Cat6}$ is approximately 4.9E-03, which is based on two events involving severe overloading (including plastic deformation) of a pipe section in 411 recorded water hammer events. This is taken as a best estimate C_L-value for calculating a Cat6 LOCA. Figure D.24 includes the rules for how the DPDs are applied to the LOCA frequency calculation. The Cat0 and Cat1 LOCAs include contributions from each loading condition associated with Cat2 or larger pressure boundary breach. In other words, the calculation accounts for the possibility that an 'unusual' loading condition may not result in a global or catastrophic pressure boundary breach. Given a through-wall flaw and severe overload, Figure D.25 shows the conditional failure probability as a function of pipe size.

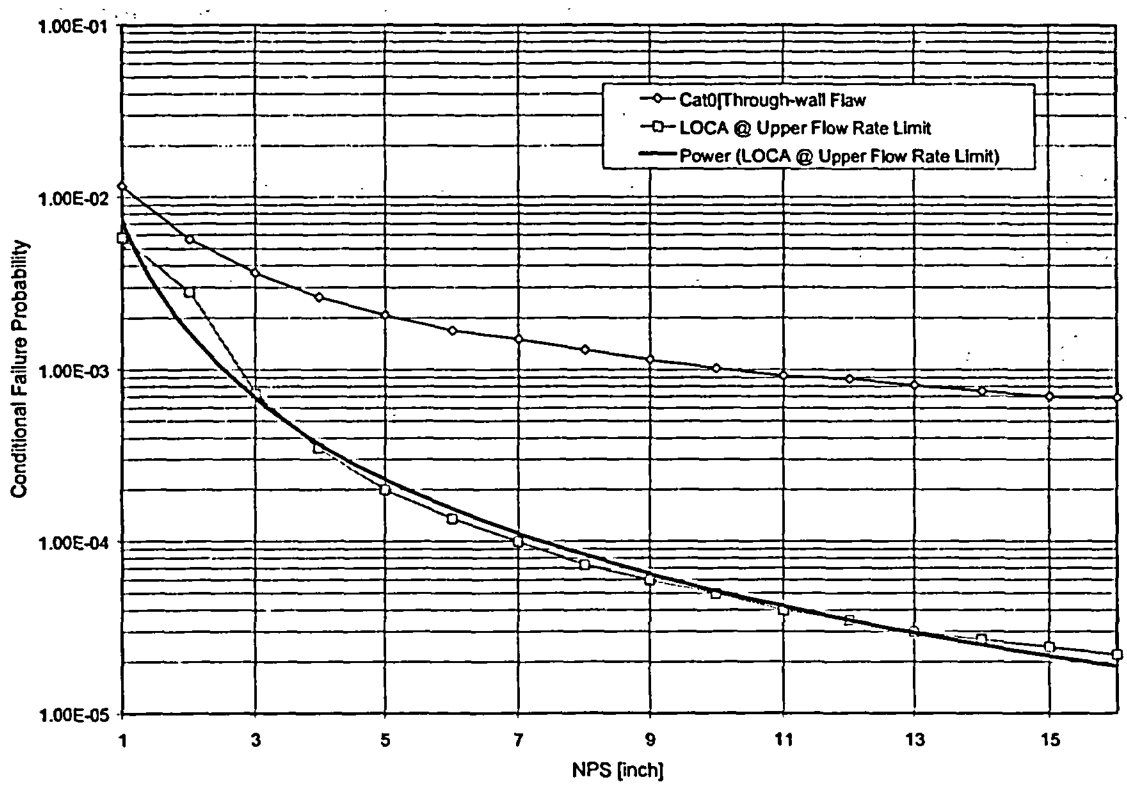

Figure D.25 Conditional Probability of Weld Failure Given Through-Wall Flaw and Severe Overloading

D.6 BWR and PWR Base Case LOCA Frequencies

The Base Case LOCA frequency models are based on three Excel files entitled "PlantBWelds" (BWR Base Case), "PlantA.aWelds" (PWR RC-HL and Surge Line), and "PlantA.bWelds" (PWR HPI/NMU). These Excel files are part of the plant design information supplied by members of the Expert Elicitation Panel. Adding information on weld type by location and weld failure parameters to respective Excel files provides the basis for calculating LOCA frequencies including uncertainty distributions.

D.6.1 LOCA Frequency Models

For the BWR Base Case, the original "PlantBWelds" includes two spreadsheets, one for the feedwater system and one for the reactor recirculation system. In a first step to create a LOCA frequency model, each spreadsheet was split into two, one for Loop A and one for Loop B of respective system. Next a new column was added to each new spreadsheet to include information on weld type by location. A review of isometric drawings provided the input to the new columns.

The statistical information that is summarized in Section D.5 of this report is included in five separate spreadsheets of the modified Excel file. The posterior weld failure rates are included on Tab "Parameters," each parameter assigned a unique variable name. The calculation of LOCA frequency, including Monte Carlo merge operations are performed on Tabs "Intermediate FW_A", "Intermediate FW_B", "Intermediate RR_A", and "Intermediate RR_B." These intermediate calculation sheets are linked to the design information for FW Loop A, FW Loop B, RR Loop A and RR Loop B, respectively. Using the variable names as defined on Tab "Parameters," each weld is assigned an appropriate failure rate (including uncertainty distribution as defined using a Crystal Ball "assumption"). Finally, an integrated calculation of LOCA frequency by leak threshold category (0 through 5) is performed on a separate spreadsheet, which is linked to the intermediate calculations. Each integrated LOCA frequency calculation is defined as a Crystal Ball "forecast."

For the PWR Base Case, the original "PlantAWelds" includes a single spreadsheet with the ASME XI Class 1 Category B-F/B-J welds. In a first step to create a LOCA frequency model, each spreadsheet was split into two, one for the RC-HL and one for the pressurizer surge line. A third spreadsheet with the HPI/NMU weld listed was added to form a new Excel-file corresponding to the PWR Base Case LOCA frequency model.

D.6.2 BWR Base Case LOCA Frequencies at T = 25

Summarized in Table D.16 are the LOCA frequency uncertainty distributions that are derived for BWR Base Case Cat0 through Cat6 LOCA. The results are representative of Plant B after 25 years of operation (T = 25).

Table D.16 Plant B LOCA Frequencies at T = 25 Years

System	LOCA Category	Uncertainty Distribution			
		Mean	5%-tile	50%-tile	95%-tile
RR Loop A NPS12	Cat0	2.06E-06	9.72E-08	1.38E-06	6.28E-06
	Cat1	1.33E-06	6.29E-08	8.95E-07	4.09E-06
	Cat2	1.44E-07	6.04E-09	8.66E-08	4.73E-07
	Cat3	5.81E-08	2.34E-09	3.51E-08	1.91E-07
	Cat4	2.31E-08	9.08E-10	1.39E-08	7.54E-08
	Cat5	N/A	N/A	N/A	N/A
RR Loop B NPS12	Cat0	2.07E-06	1.00E-07	1.36E-06	6.41E-06
	Cat1	1.34E-06	6.45E-08	8.81E-07	4.15E-06
	Cat2	1.44E-07	5.85E-09	8.65E-08	4.72E-07
	Cat3	5.84E-08	2.26E-09	3.47E-08	1.95E-07
	Cat4	2.31E-08	9.15E-10	1.40E-08	7.48E-08
	Cat5	N/A	N/A	N/A	N/A
RR Loop A NPS22	Cat0	1.23E-06	4.88E-08	7.30E-07	4.07E-06
	Cat1	8.58E-07	3.37E-08	5.12E-07	2.89E-06
	Cat2	7.36E-07	2.42E-09	3.94E-08	2.57E-07
	Cat3	2.95E-08	1.01E-09	1.60E-08	1.01E-07
	Cat4	1.16E-08	3.94E-10	6.33E-09	4.04E-08
	Cat5	4.72E-09	1.57E-10	2.52E-09	1.63E-08
RR Loop B NPS22	Cat0	1.24E-06	4.97E-08	7.26E-07	4.06E-06
	Cat1	8.66E-07	3.39E-08	5.10E-07	2.88E-06
	Cat2	7.37E-08	2.48E-09	3.95E-08	2.59E-07
	Cat3	2.91E-08	9.77E-10	1.59E-08	1.01E-07
	Cat4	1.18E-08	3.86E-10	6.36E-09	3.97E-08
	Cat5	4.73E-09	1.56E-10	2.55E-09	1.67E-08
RR Loop A NPS28	Cat0	2.73E-06	1.31E-07	1.80E-06	8.41E-06
	Cat1	1.91E-06	9.05E-08	1.24E-06	5.91E-06
	Cat2	1.64E-07	6.27E-09	9.69E-08	5.38E-07
	Cat3	6.54E-08	2.49E-09	3.92E-08	2.16E-07
	Cat4	2.61E-08	1.03E-09	1.55E-08	8.65E-08
	Cat5	1.05E-08	4.24E-10	6.20E-09	3.50E-08
RR Loop B NPS28	Cat0	2.76E-06	1.35E-07	1.82E-06	8.44E-06
	Cat1	1.93E-06	9.31E-08	1.27E-06	5.92E-06
	Cat2	1.65E-07	6.32E-09	9.73E-08	5.42E-07
	Cat3	6.62E-08	2.57E-09	3.93E-08	2.18E-07
	Cat4	2.64E-08	1.10E-09	1.56E-08	8.74E-08
	Cat5	1.06E-08	4.40E-10	6.33E-09	3.40E-08
FW Loop A NPS12	Cat0	6.69E-07	2.78E-08	4.05E-07	2.20E-06
	Cat1	4.34E-07	1.80E-08	2.61E-07	1.42E-06
	Cat2	4.75E-08	1.61E-09	2.58E-08	1.61E-07
	Cat3	1.89E-08	6.41E-10	1.03E-08	6.49E-08
	Cat4	7.53E-09	2.62E-10	4.07E-09	2.53E-08
	Cat5	N/A	N/A	N/A	N/A
FW Loop B NPS12	Cat0	6.83E-07	2.89E-08	4.16E-07	2.24E-06
	Cat1	4.44E-07	1.90E-08	2.68E-07	1.45E-06
	Cat2	4.86E-08	1.70E-09	2.64E-08	1.67E-07
	Cat3	1.93E-08	6.58E-10	1.06E-08	6.71E-08
	Cat4	7.75E-09	2.80E-10	4.20E-09	2.62E-08
	Cat5	N/A	N/A	N/A	N/A
FW Loop B NPS14	Cat0	6.02E-08	1.52E-09	2.61E-08	2.26E-07
	Cat1	4.19E-08	1.06E-09	1.84E-08	1.55E-07
	Cat2	3.59E-09	7.64E-11	1.42E-09	1.37E-08
	Cat3	1.45E-09	3.12E-11	5.69E-10	5.51E-09

Table D.16 Plant B LOCA Frequencies at T = 25 Years

System	LOCA Category	Uncertainty Distribution			
		Mean	5%-tile	50%-tile	95%-tile
	Cat4	5.80E-10	1.19E-11	2.30E-10	2.21E-09
	Cat5	2.32E-10	4.74E-12	9.11E-11	8.83E-10
FW Loop A NPS20	Cat0	8.43E-07	3.47E-08	5.08E-07	2.74E-06
	Cat1	5.92E-07	2.37E-08	3.55E-07	1.95E-06
	Cat2	5.08E-08	1.75E-09	2.75E-08	1.74E-07
	Cat3	2.03E-08	7.15E-10	1.10E-08	6.83E-08
	Cat4	8.10E-09	2.77E-10	4.33E-09	2.81E-08
	Cat5	3.26E-09	1.12E-10	1.74E-09	1.13E-08
FW Loop B NPS20	Cat0	1.00E-06	4.40E-08	6.13E-07	3.24E-06
	Cat1	7.014-07	3.01E-08	4.28E-07	2.12E-06
	Cat2	6.03E-08	1.03E-09	3.32E-08	2.04E-07
	Cat3	2.41E-08	8.45E-10	1.33E-08	8.00E-08
	Cat4	9.64E-09	3.44E-10	5.27E-09	3.32E-09
	Cat5	3.84E-09	1.38E-10	2.14E-09	1.31E-08
RR Total Loops A & B	Cat0	1.21E-05	3.02E-06	1.03E-05	2.70E-05
	Cat1	8.24E-06	2.02E-06	6.98E-06	1.86E-05
	Cat2	7.64E-07	1.40E-07	6.07E-07	1.92E-06
	Cat3	3.07E-07	5.43E-08	2.44E-07	7.79E-07
	Cat4	1.22E-07	2.22E-08	9.73E-08	3.06E-07
	Cat5	3.05E-08	3.59E-09	2.19E-08	8.52E-08
FW Total Loops A & B	Cat0	3.26E-06	5.55E-07	2.60E-06	8.13E-06
	Cat1	2.21E-06	2.72E-07	1.75E-06	5.56E-06
	Cat2	2.10E-07	2.70E-08	1.54E-07	5.85E-07
	Cat3	8.40E-08	1.10E-08	6.18E-08	2.30E-07
	Cat4	3.36E-08	4.40E-09	2.47E-08	9.35E-08
	Cat5	7.33E-09	3.95E-10	4.33E-09	2.44E-08
RR + FW Total	Cat0	1.53E-05	5.24E-06	1.37E-05	3.10E-05
	Cat1	1.05E-05	3.51E-06	9.30E-06	2.14E-05
	Cat2	9.75E-07	2.24E-07	8.24E-07	2.26E-06
	Cat3	3.90E-07	9.00E-08	3.32E-07	9.05E-07
	Cat4	1.56E-07	3.64E-08	1.31E-07	3.57E-07
	Cat5	3.78E-08	6.43E-09	2.93E-08	9.73E-08

D.6.3 PWR Base Case LOCA Frequencies at T = 25

The PWR Base Case Cat0 through Cat6 LOCA frequencies including uncertainty distributions, are summarized in Table D.17. These results are representative of Plant A.a/A.b after 25 years of operation (T = 25).

Table D.17 Plant A.a/A.b LOCA Frequencies at T = 25 Years

System	LOCA Category	Uncertainty Distribution			
		Mean	5%-tile	50%-tile	95%-tile
RC Hot Leg (3-of-3); Plant A.a	Cat0	8.94E-07	4.84E-09	1.27E-07	2.88E-06
	Cat1	6.65E-07	3.55E-09	9.39E-08	2.14E-06
	Cat2	4.87E-08	2.10E-10	6.15E-09	1.49E-07
	Cat3	1.83E-08	8.33E-11	2.42E-09	5.95E-08
	Cat4	6.99E-09	3.03E-11	8.93E-10	2.21E-08
	Cat5	2.55E-09	1.16E-11	3.29E-10	8.29E-09
	Cat6	1.26E-09	5.44E-12	1.58E-10	4.04E-09
RC Surge Line Plant A.a	Cat0	1.44E-07	2.65E-09	2.98E-08	5.02E-07
	Cat1	1.14E-07	2.13E-09	2.36E-08	3.94E-07
	Cat2	9.60E-09	1.48E-10	1.88E-09	3.46E-08
	Cat3	3.84E-09	5.78E-11	3.50E-10	1.35E-08
	Cat4	1.44E-09	2.01E-11	2.77E-10	5.06E-09
	Cat5	5.31E-10	8.23E-12	1.01E-10	1.87E-09
	Cat6	N/A	N/A	N/A	N/A
HPI/NMU (2-of-2) Plant A.b	Cat0	2.72E-05	4.64E-07	6.90E-06	1.07E-04
	Cat1	1.60E-05	2.62E-07	3.93E-06	6.09E-05
	Cat2	2.33E-06	3.30E-08	5.40E-07	9.02E-06
	Cat3	9.22E-07	1.28E-08	2.14E-07	3.59E-06
	Cat4	N/A	N/A	N/A	N/A
	Cat5	N/A	N/A	N/A	N/A
	Cat6	N/A	N/A	N/A	N/A

D.6.4 Time-Dependency of LOCA Frequency Results

For respective Base Case Plant, the LOCA frequencies are determined for three time periods: T= = 25 years after plant startup (corresponding to today's state-of-knowledge), T = 40 years after plant startup (corresponding to original design life), and T = 60 years after plant startup (corresponding to end-of-life extension). The time-dependent analysis is performed in two different ways. First a 'prospective analysis' is performed based on a Markov model of piping reliability (Figure D.15). Second, a 'retrospective analysis' is performed by using Bayesian statistics.

D.6.4.1 Use of Markov Model to Determine Time-Dependency - According to the Markov model diagram in Figure D.15, a piping component can be in four mutually exclusive states: S (= Success), C (= Cracked), F (= Leaking, non-active leakage, or active leakage with leak rate within Technical Specification Limit) or L (= Leaking, with leak rate in excess of Technical Specification Limit). The time-dependent probability that a piping component is in each state S, C, F, or L is described by a differential equation. Under the assumption that all the state transition rates are constant the Markov model equations will consist of a set of coupled linear differential equations with constant coefficients. The reliability term needed to represent LOCA frequency is the system failure rate or hazard rate $h\{t\}$, which is time-dependent. The hazard rate is defined as:

$$h\{t\} = (1/(1 - L\{t\})) \times dL\{t\}/dt \qquad \text{(D.9)}$$

Where:

$$1 - L\{t\} = S\{t\} + C\{t\} + F\{t\} \qquad \text{(D.10)}$$

The hazard rate is a function of time and the parameters of the Markov model; $h\{t\}$ is the time-dependent frequency of pipe rupture. Reference [D.14] provides solutions to the Markov model and derives an expression for $h\{t\}$ as a function of the six parameters associated with the 4-state Markov model: An occurrence rate for detectable flaws (ϕ), a failure rate for leaks given the existence of a flaw (λ_F), two rupture frequencies including one from the initial state of a flaw (ρ_F) and one from the initial state of a leak (ρ_L), a repair rate for detectable flaws (ω), and a repair rate for leaks (μ). The latter two parameters dealing with repair are further developed by the following simple models.

$$\omega = \frac{P_{FI} P_{FD}}{(T_{FI} + T_R)} \tag{D.11}$$

Where:

P_{FI} = probability that a piping element with a flaw will be inspected per inspection interval. This parameter has a value of 0 if it is not in the inspection program and 1 if it is in the inspection program. For the inspected elements, a value of 1 is used for any ISI inspection case and 0 for the case of no ISI. The element may be selected for inspection directly by being included in the sections sampled for ISI inspection, or indirectly by having a rule such that if degradation is detected anywhere in the system, the search will be expanded to include examination of that element.

P_{FD} = probability that a flaw will be detected given this element is inspected. This is the reliability of the inspection program and is equivalent to the term used by NDE experts, "Probability of detection (POD)." This probability is conditioned on the occurrence of one or more detectable flaws in the segment according to the assumptions of the model. Also note that

T_{FI} = mean time between inspections for flaws, (inspection interval).

T_R = mean time to repair once detected. Depending on the location of the weld to be repaired, the actual weld repair could take on the order of several days to much more than a week. Accounting for time to prepare for repair, NDE, root cause evaluation, etc., the total outage time attributed to the repair of a Class 1 weld is on the order of 1 month or more. However, since this term is always combined with T_{FI}, and T_{FI} could be 10 years, in practice the results are insensitive to assumptions regarding T_R

Similarly, estimates of the repair rate for leaks can be estimated according to:

$$\mu = \frac{P_{LD}}{(T_{LI} + T_R)} \tag{D.12}$$

Where:

P_{LD} = probability that the leak in the element will be detected per leak inspection or detection period

T_{LI} = mean time between inspections for leaks. For RCPB piping the time interval between leaks can be essentially instantaneous if the leak is picked up by radiation alarms, to as long as the time period between leak tests performed on the system.

T_R = as defined above but for full power applications, this time should be the minimum of the actual repair time and the time associated with cooldown to enable repair and any waiting time for replacement piping.

A summary of the root input parameters of the Markov model and the general strategy for estimation of each parameter is presented in Table D.18.

Table D.18 Four-State Markov Model Root Input Parameters

Parameter	Assumed or Estimated Value	Basis
ω	2.1×10^{-2}/year {=(.25) × (.90)/(10+(200/8760))}	Element assumed to have a 25% chance of being inspected for flaws every 10 years with a 90% detection probability. In the given example detected flaws will be repaired in 200 hours
μ	7.92×10^{-1}/ year {=(.90) × (.90)/(1+(200/8760))}	Element is assumed to have a 90% chance of being inspected for leaks once a year with a 90% leak detection probability
ρ_C	Table D.13, D.14 and D.15	The basis is developed in Sections D.4 and D.5.
λ_C	Table D.13 and D.14	The basis is developed in Sections D.4 and D.5.
ρ_F	2.0×10^{-2}/year	If the element is already leaking, the conditional frequency of ruptures is assumed to be determined by the frequency of severe overloading events; the given value is equal to the frequency of severe water hammer (from PIPExp database).
ϕ	Variable (for IGSCC $\phi = 7.58 \times (\lambda_C + \rho_C)$)	The occurrence rate of a flaw is estimated from service data. As an example, IGSCC in the BWR operating environment will create ca. 7.58 flaws for every through-wall leak that is observed.
P_{FI}	1 or 0	Probability per inspection interval that the pipe element will be included in the inspection program.
P_{FD}	Variable (see text above for details)	Probability per inspection interval that an existing flaw will be detected. A chosen estimate is based on NDE reliability performance demonstration results and difficulty and accessibility of inspection for particular weld.
P_{LD}	Variable (0 – no leak detection to 0.9 for leak detection using current methods/technology)	Probability per detection interval that an existing leak will be detected. Estimate based on system, presence and type of leak detection system, and locations and accessibility.
T_{FI}	10 years (per ASME XI)	Flaw inspection interval, mean time between inservice inspections.
T_{LD}	Variable (1.5 – once per refueling outage / 1.92E-2 – weekly / 9.13E-4 – each shift)	Leak detection interval, mean time between leak detections. Estimate based on method of leak detection; ranges from immediate/ continuous to frequency of routine inspections for leaks (incl. hydrostatic pressure testing).
T_R	Variable (see text above for details)	Mean time to repair the affected piping element given detection of a critical flaw or leak. Estimate of time to tag out, isolate, prepare, repair, leak test and tag into service.

In addition to generating a time-dependent LOCA frequency, the Markov model provides a basis for investigating the sensitivity of LOCA frequency to different inservice inspection and leak detection strategies. The Markov model determines the inspection effectiveness factor, I, which is the ratio of the LOCA frequency with credit for inspections to that given no credit for inspections:

$$I_j = \frac{h_{25}\{inspprog' j'\}}{h_{25}\{noinsp\}} \tag{D.13}$$

Where:

$h_{25}\{inspprog\ 'j'\}$ = hazard rate at T = 25 given inspection strategy 'j.'
$h_{25}\{noinsp\}$ = hazard rate given no inspections.

The solutions to the Markov model for time dependent hazard rates are developed in terms of closed form analytical solutions using an Excel spreadsheet. In this study the time-dependent LOCA frequencies are determined for twelve cases that are defined by varying the following parameters (Table D.19):

- Whether or not the piping segment is subjected to any ISI program;
- The extent of the ISI program ('Caused-Based' vs. 'Extensive', all encompassing);
- The inspection interval of the ISI program;
- Type and frequency of leak detection. The different leak detection methods include primary system mass balance calculations, visual observation (through video monitor), (PWR) containment sump level and flow rate monitors, airborne particulate radioactivity and gaseous radioactivity monitors, and different main control room monitors for primary system temperature, pressure, etc.

Table D.19 Inspection Cases Evaluated for Selected Pipe Segments

Leak Inspection Strategy	In-Service Inspection Strategy		
	None	Cause-Based [P_{FD} = 0.50]	Comprehensive [P_{FD} = 0.90]
None	Case 1	Case 5	Case 9
Refueling Cycle (Hydro Test)	Case 2	Case 6	Case 10
Weekly	Case 3	Case 7	Case 11
8 Hour Shift	Case 4	Case 8	Case 12

The time-dependent LOCA frequencies associated with the five Base Cases are summarized in Figures D.25 through D.39. Figure D.25 is assumed to be representative of Base Case 1; the LOCA frequency at T = 25 years is equal to the calculated point estimate of 8.24E-06 per reactor-year under an assumption of "caused-based" ISI with POD = 0.5 and leak detection (e.g., hydrostatic pressure testing prior to exiting a refueling outage). This assumption is applied to the other base cases as well (Figures D.28, D.31, D.34, and D.37).

It is noted that the service data input to the calculation is associated with piping that has been subjected to different inspection strategies. In some cases flaws have been detected fortuitously and in other cases the flaw detection has resulted from augmented IGSCC inspection programs. The results in Figure D.25 are based on an assumed 'cause-based' inspection strategy whereby the inspection sample is determined by an initial discovery of a flaw. If a flaw is found, the inspection is immediately expanded to cover other similar locations. The combination of inspection sample and rules for expanded search for flaws are sufficient to result in an average probability of detection (POD) of 0.50. The analysis also considers what in this study is termed "comprehensive" ISI, which implies 100% ISI coverage using state-of-the-art NDE technology. Such a program is assumed to result in an average probability of detection of 0.90.

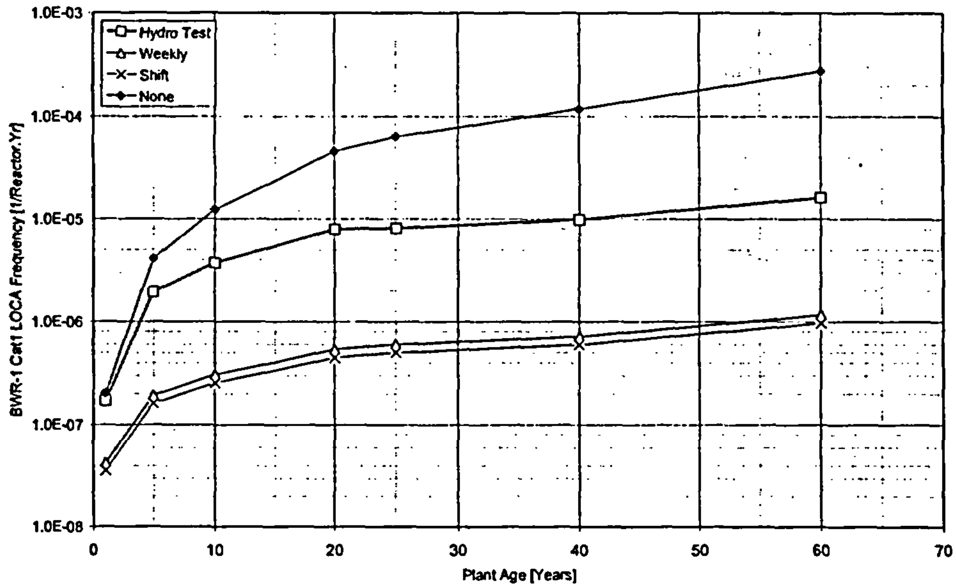

Figure D.26 Time-Dependent BWR-1 Cat 1 LOCA Frequency Given 'Cause-Based' ISI

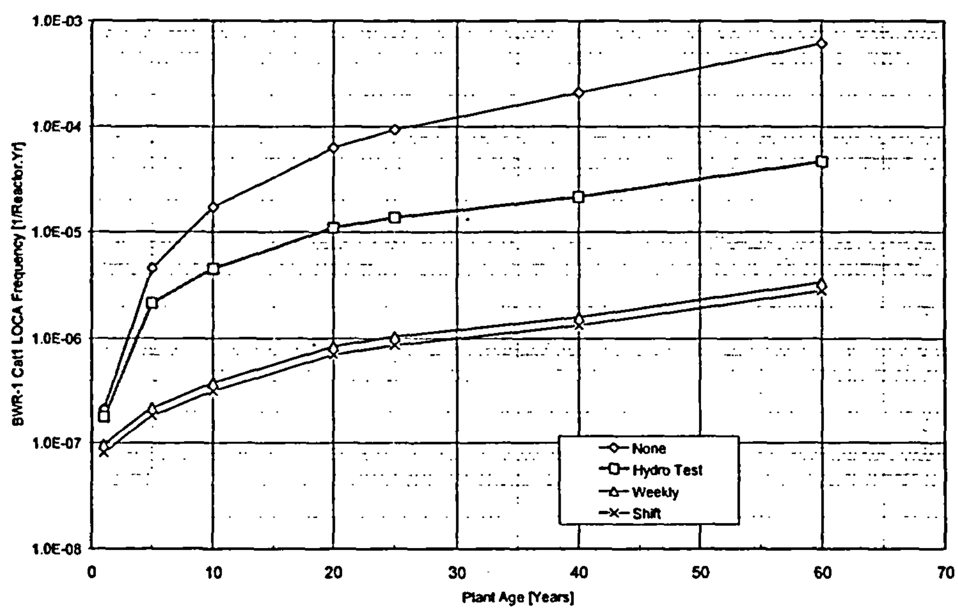

Figure D.27 Time-Dependent BWR-1 Cat 1 LOCA Frequency Assuming no ISI

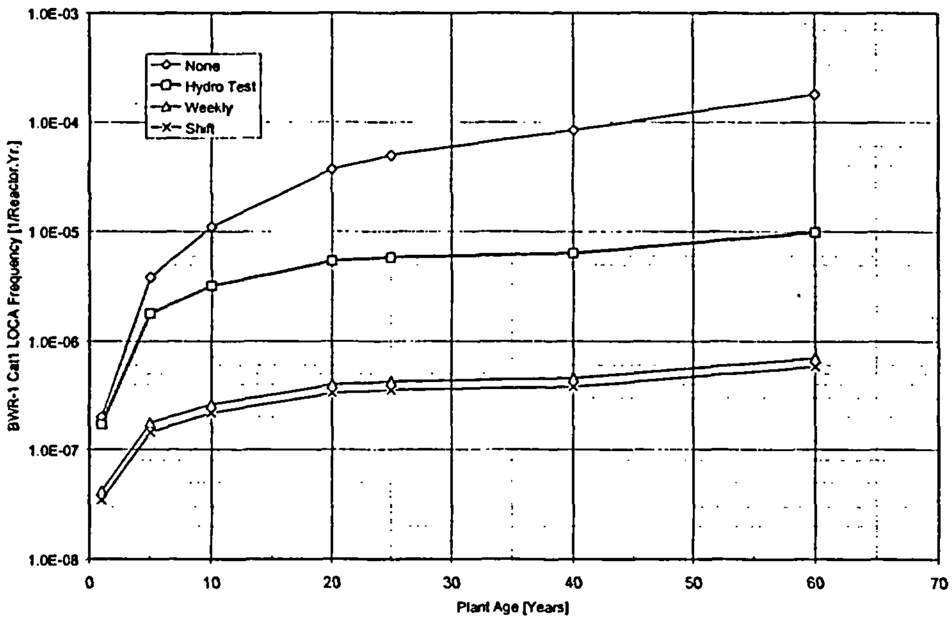

Figure D.28 Time-Dependent BWR-1 Cat 1 LOCA Frequency Given 'Comprehensive ISI'

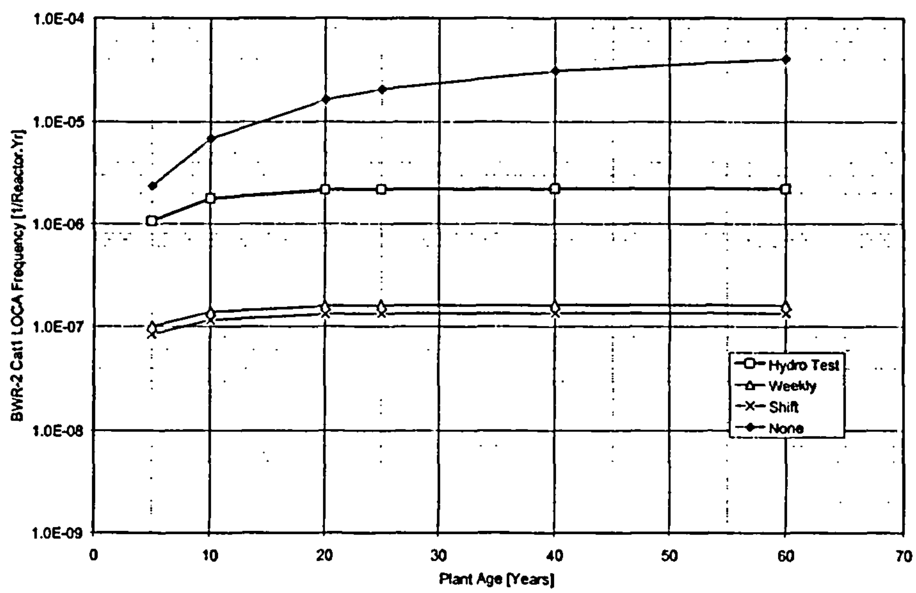

Figure D.29 Time-Dependent BWR-2 Cat 1 LOCA Frequency Given 'Cause-Based' ISI

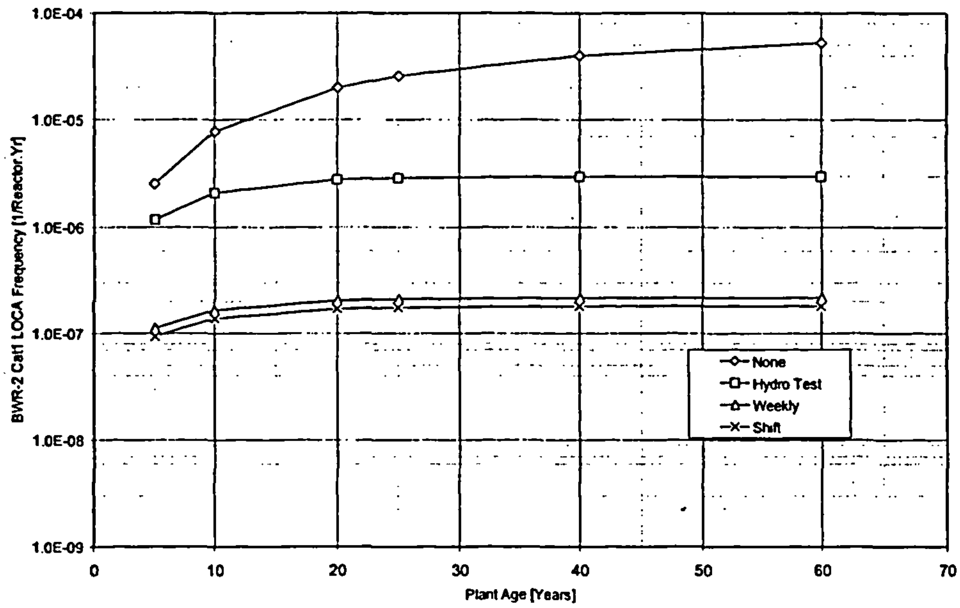

Figure D.30 Time-Dependent BWR-2 Cat 1 LOCA Frequency Assuming no ISI

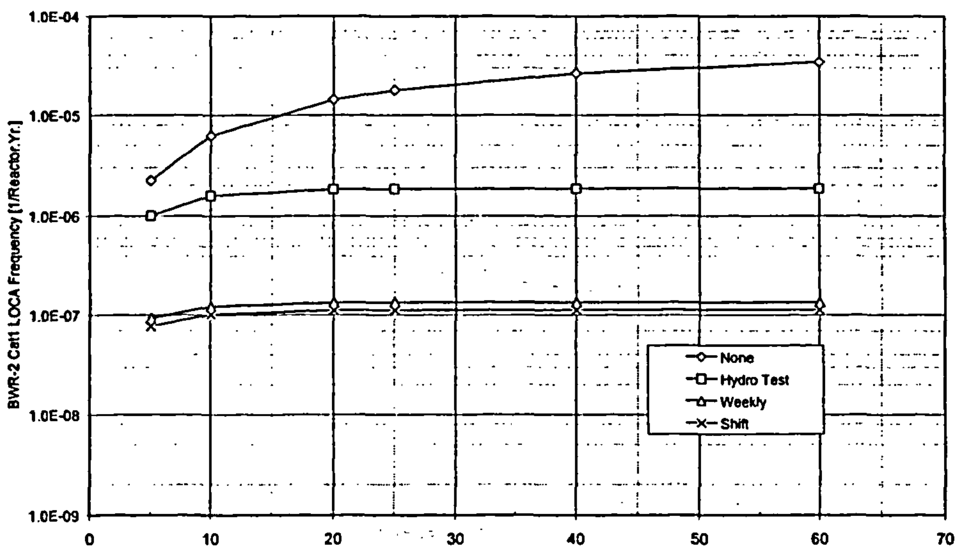

Figure D.31 Time-Dependent BWR-2 Cat 1 LOCA Frequency Given 'Comprehensive ISI'

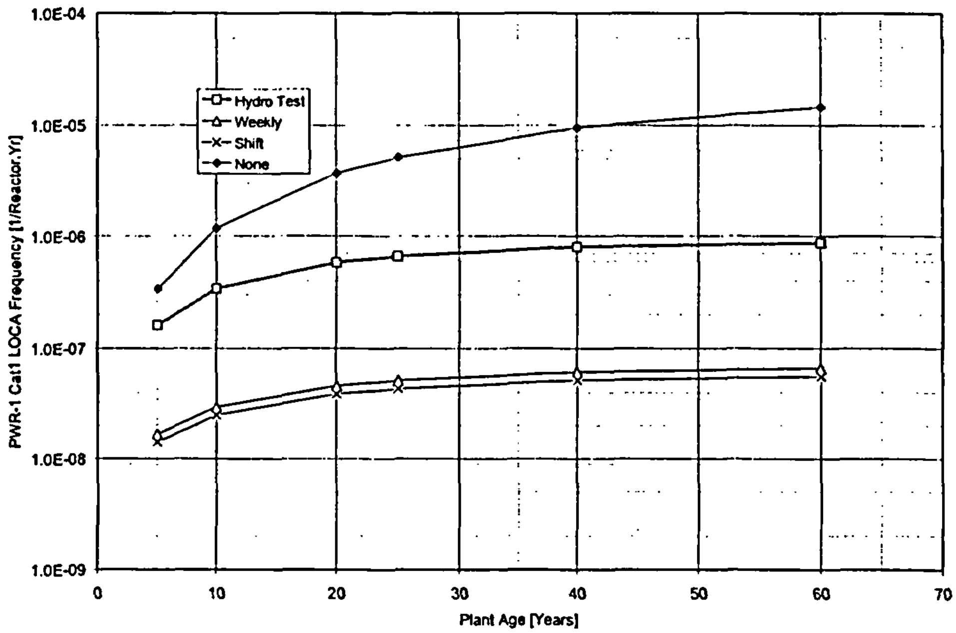

Figure D.32 Time-Dependent PWR-1 Cat 1 LOCA Frequency Given 'Cause-Based' ISI

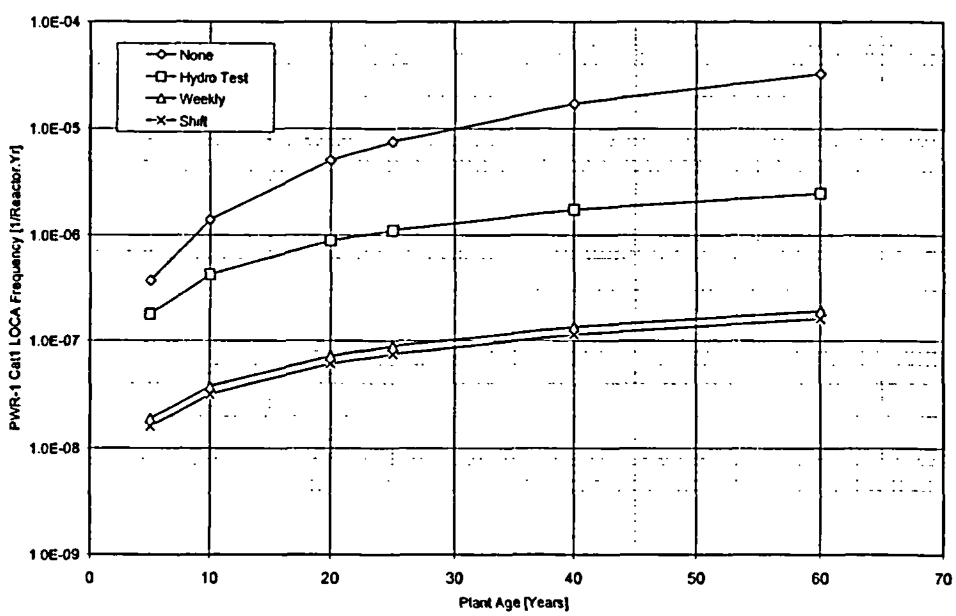

Figure D.33 Time-Dependent PWR-1 Cat 1 LOCA Frequency Assuming no ISI

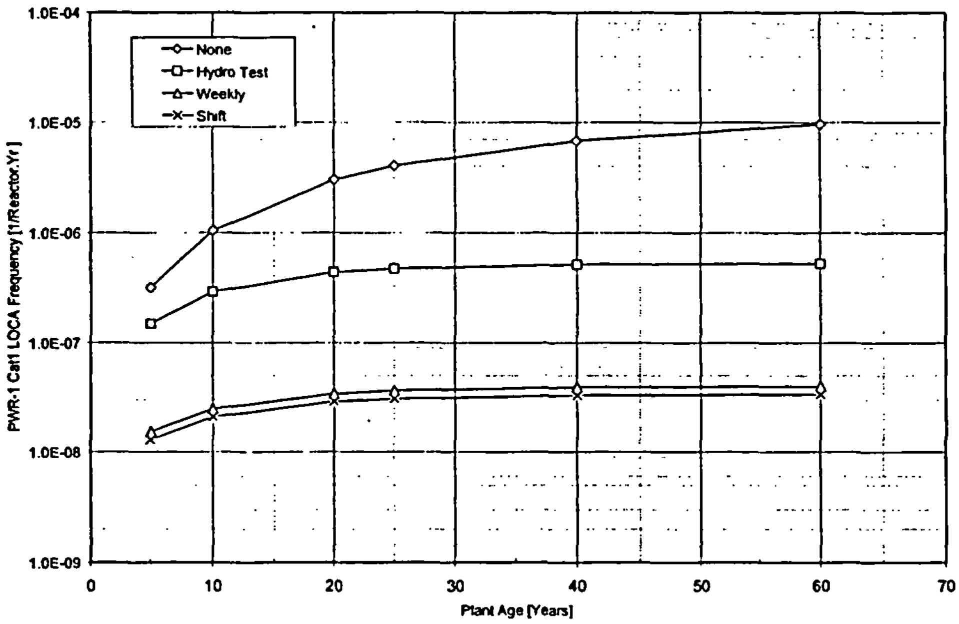

Figure D.34 Time-Dependent PWR-1 Cat 1 LOCA Frequency Given 'Comprehensive ISI'

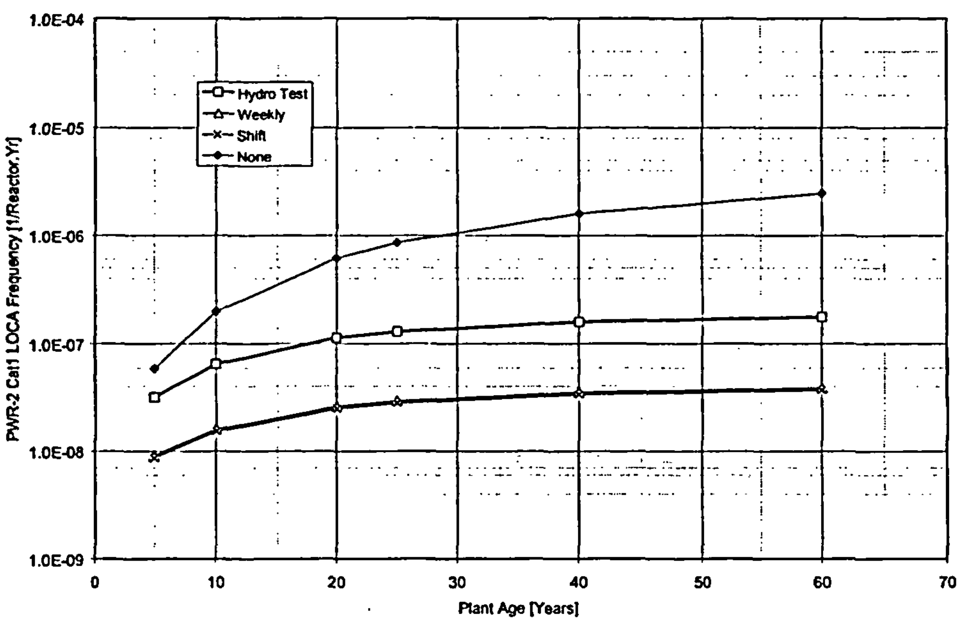

Figure D.35 Time-Dependent PWR-2 Cat 1 LOCA Frequency Given 'Cause-Based' ISI

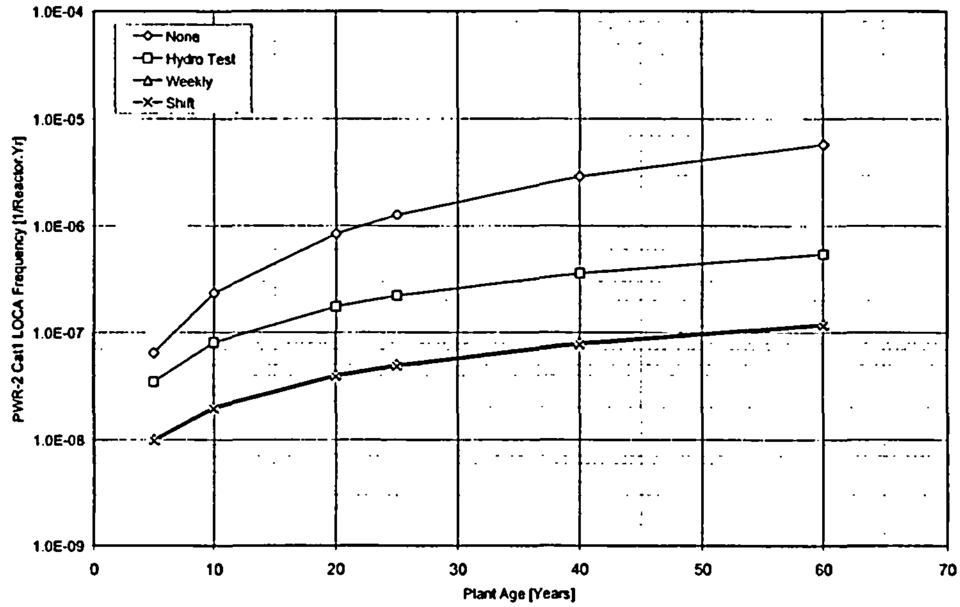

Figure D.36 Time-Dependent PWR-2 Cat 1 LOCA Frequency Assuming no ISI

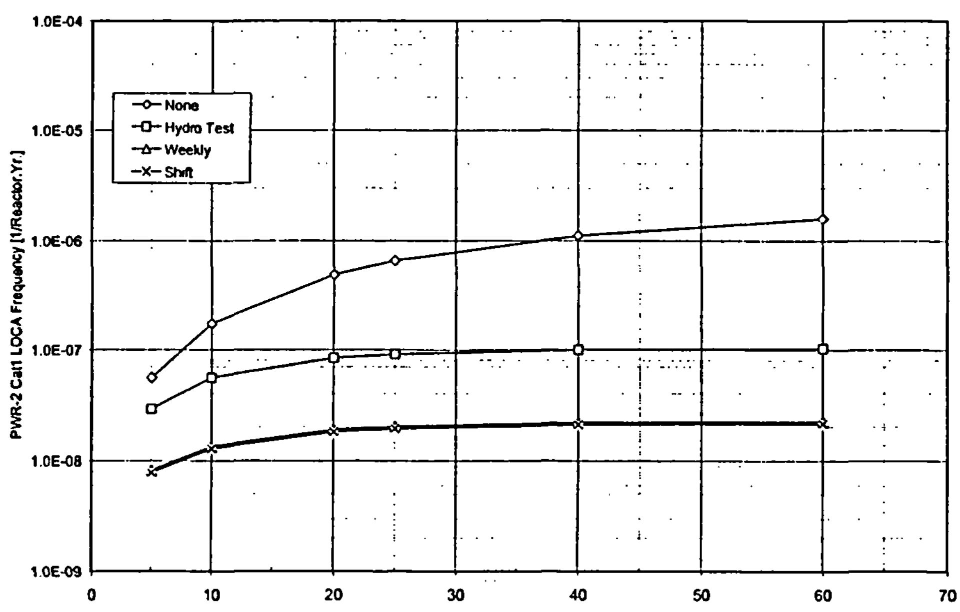

Figure D.37 Time-Dependent PWR-2 Cat 1 LOCA Frequency Given 'Comprehensive ISI'

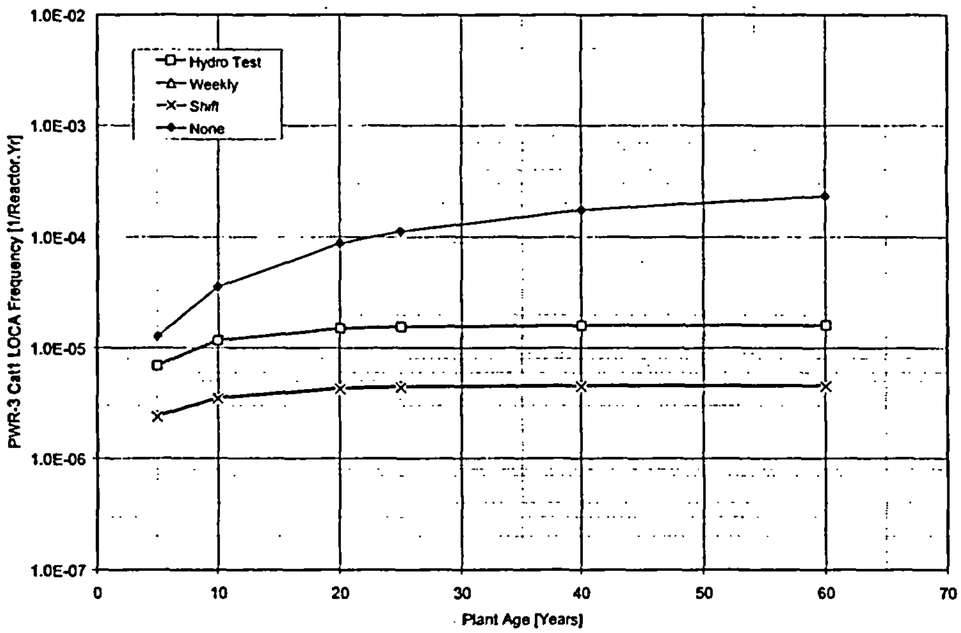

Figure D.38 Time-Dependent PWR-3 Cat 1 LOCA Frequency Given 'Cause-Based' ISI

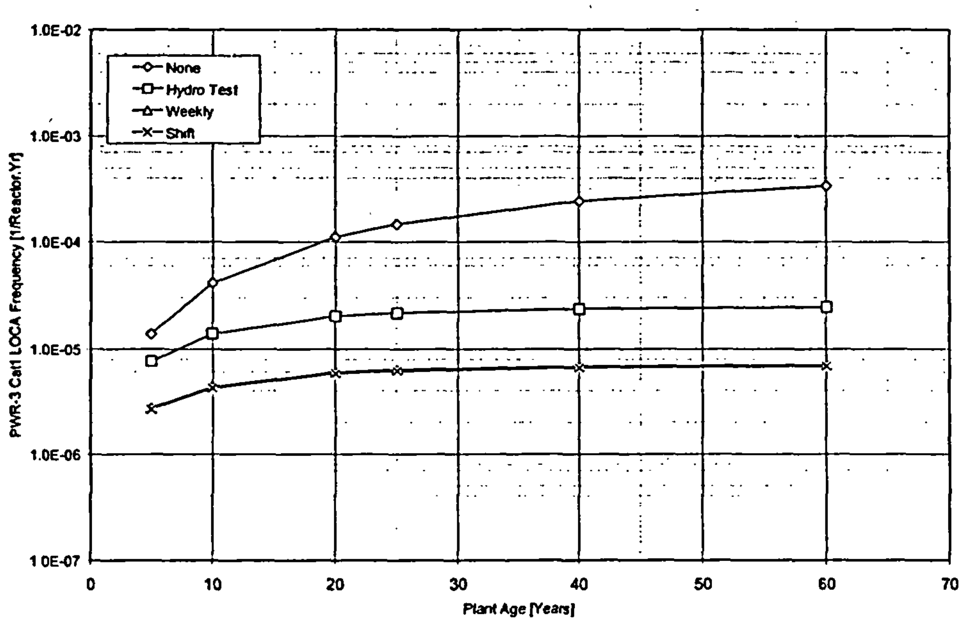

Figure D.39 Time-Dependent PWR-3 Cat 1 LOCA Frequency Assuming no ISI

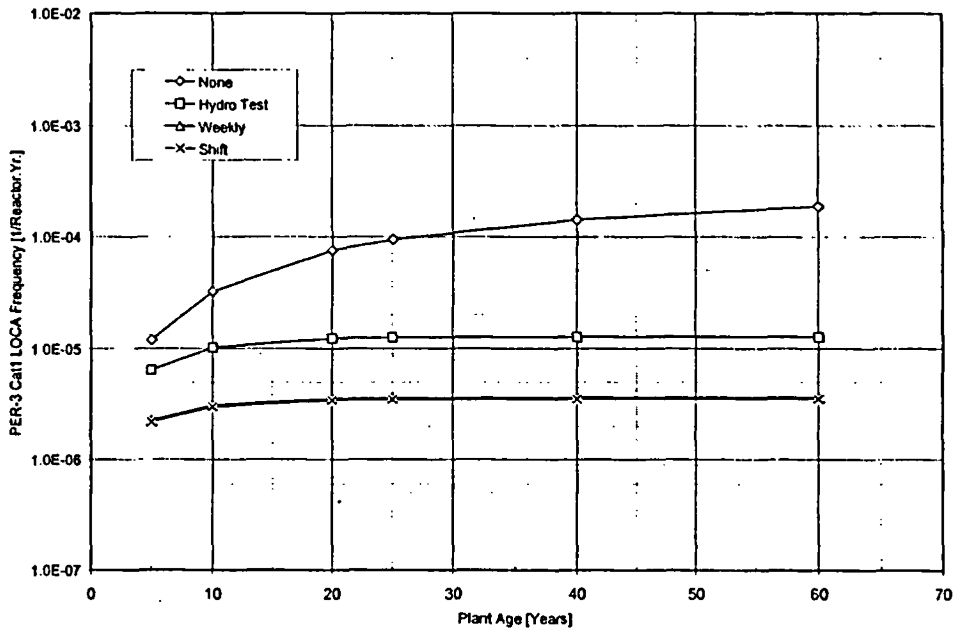

Figure D.40 Time-Dependent PWR-3 Cat 1 LOCA Frequency Given 'Comprehensive ISI'

D.6.4.2 Speculative LOCA Frequency at T = 40 & T = 60 - A retrospective evaluation is performed through a Bayesian update process whereby the exposure term in Equation 4.1 is modified to account for the longer exposure time. The analysis is performed by assuming that the service history at T = 40 and T = 60 years is known; zero (0) weld failures during the intervals ΔT = 15 years (T40 to T25) and ΔT = 35 years (T60 to T25). This is a purely speculative assumption implying that the ISI/NDE technologies and other piping reliability management programs remain at least as effective as at the present and that no unexpected material aging occurs. The extrapolated LOCA frequencies are summarized in Table D.20. Under the given assumptions the LOCA frequency would be expected to decrease with time.

Table D.20 Base Case LOCA Frequency Results (T = 25, 40 & 60 Years)

Base Case	LOCA Frequency – Statistical Mean [1/Reactor-year] Flow Rate Interval [gpm]				
	100 < v ≤ 1500	1500 < v ≤ 5000	5000 < v ≤ 25,000	25,000 < v ≤ 100,000	100,000 < v ≤ 500,000
BWR-1, T = 25	8.24E-06	7.64E-07	3.07E-07	1.22E-07	3.05E-08
BWR-1; T = 40	2.67E-06	2.29E-07	9.14E-08	3.64E-08	1.45E-08
BWR-1; T = 60	2.44E-06	2.08E-07	8.38E-08	3.34E-08	1.34E-08
BWR-2, T = 25	2.21E-06	2.11E-07	8.40E-08	3.36E-08	7.33E-09
BWR-2, T = 40	2.07E-06	2.03E-07	8.05E-08	3.13E-08	6.61E-09
BWR-2, T = 60	1.87E-06	1.85E-07	7.35E-08	2.97E-08	6.09E-09
PWR-1, T = 25	6.65E-07	4.87E-08	1.83E-08	6.99E-09	2.55E-09
PWR-1, T = 40	2.14E-07	1.49E-08	6.10E-09	2.24E-09	8.14E-10
PWR-1, T = 60	1.19E-07	8.34E-09	3.38E-09	1.26E-09	4.62E-10
PWR-2, T = 25	1.14E-07	9.60E-09	3.84E-09	1.44E-09	5.31E-10
PWR-2, T = 40	1.07E-07	9.22E-09	3.68E-09	1.34E-09	4.79E-10
PWR-2, T = 60	9.67E-08	8.31E-09	3.36E-09	1.27E-09	4.41E-10
PWR-3, T = 25	1.60E-05	2.33E-06	9.22E-07	N/A	N/A
PWR-3, T = 40	1.08E-05	1.58E-06	6.31E-07	N/A	N/A
PWR-3, T = 60	8.23E-06	1.20E-06	4.81E-07	N/A	N/A

Note 1: PWR-1 in this table accounts for 3-of-3 hot legs.
Note 2: PWR-3 in this table accounts for 2-of-2 HPI/NMU lines.

D.6.5 Influence of Service Data on LOCA Frequency

The LOCA frequencies in this Base Case Report are derived from service data on Code Class 1 piping. In this section we investigate how the LOCA frequencies relate to two data issues: 1) completeness of the pipe failure data collection, and 2) data interpretations. The former remains an ever-present issue in probabilistic safety assessment. Completeness is addressed by having in place an active and rigorous data collection process (*c.f.* Appendix A). Two sensitivity cases (SC:s) are defined to demonstrate how changes in the input to the failure rate calculations affect the estimated LOCA frequency. The sensitivity cases are defined as:

1. SC1: A small leak (≤ T.S. limit for unidentified RCPB-leakage) is assumed to have occurred in a pipe-to-safe-end weld in a BWR NPS28 reactor recirculation pipe during the time period 1988 – 2002. This evidence is used to modify the posterior weld failure rates.
2. SC2: This sensitivity case is concerned with an assumed large leak (= Cat0 LOCA) in a NPS28 BWR reactor recirculation pipe. Again, the large leak is assumed to have occurred in the time period 1988 – 2002. This evidence is used to modify the posterior weld failure rates and the conditional failure probability.

The results of the sensitivity analysis are summarized in Table D.21. These sensitivity cases are hypothetical in that they do not account for effects on piping reliability by the anticipated industry and regulatory actions that invariably would arise in response to the results of root cause analysis to determine the reasons behind a significant RCPB degradation such as defined by SC1 or SC2.

Table D.21 BWR LOCA Frequency Sensitivity Analysis Results

Base Case	LOCA Frequency – Statistical Mean [1/Reactor-year] Flow Rate Interval [gpm]				
	Cat1 100 < v ≤ 1500	Cat2 1500 < v ≤ 5000	Cat3 5000 < v ≤ 25,000	Cat4: 25,000 < v ≤ 100,000	Cat5: 100,000 < v ≤ 500,000
Base-1	8.24E-06	7.64E-07	3.07E-07	1.22E-07	3.05E-08
Base-1 – SC1	8.70E-06	8.07E-07	3.27E-07	1.29E-07	3.29E-08
Base-1 – SC2	1.30E-05	1.17E-06	4.77E-07	1.87E-07	5.63E-07

D.6.6 Service Data and Conditional Failure Probabilities

There is no service data associated with Cat0 LOCA events. Therefore, the estimation of conditional failure probabilities is based on zero-failure statistics. Since not all flaws propagate through-wall if left unattended, an alternative to the approach in Section D.5.2 (constrained noninformative prior) would be to use Jeffrey's noninformative prior and to assume all flaws (non-through wall and through wall) as pressure boundary integrity challenges. The result would be conditional failure probabilities that are closely approximated by the power law (Equation D.8), however. It is acknowledged that this is just one way of representing the current state-of-knowledge with respect to gross Code Class 1 pipe failure. It is not a physical model of flaw propagation given its interactions with certain loading conditions and pipe stresses.

D.7 Summary of Results

An application of a parametric attribute/influence method has yielded results as summarized in this section. Central to the method is the processing and interpretation of service data on Code Class 1 piping. A Markov model of piping reliability is used to develop time-dependent LOCA frequencies.

D.7.1 Discussion of Assumptions

A parametric attribute/influence method is applied to five base cases. Three types of assumptions are made in the analysis; global assumptions (applicable to all five base cases), BWR-specific assumptions and PWR-specific assumptions:

Global Assumptions

- Pipe failure results from observable degradation mechanisms and loading conditions. A statistical evaluation of service experience data therefore provides a sufficiently accurate basis for piping reliability analysis.
- The PIPExp database is of sufficient completeness and depth to support an application of the parametric attribute/influence methodology. This database addresses piping performance in response to both anticipated and unanticipated loading conditions.
- The effect on piping reliability from pressure, deadweight, weld residual stresses, thermal loading, and thermal stratification is implicitly accounted for in the PIPExp database. This database also accounts for the effects from inadvertent over-pressurization and relief valve actuation, water hammer and seismic[13] events.

[13] The database includes a single event involving the fracture of a small-diameter steam line due to seismic event (Fukushima-Daiichi Unit 6 on 07-21-2000).

BWR-Specific Assumptions

- The BWR-specific LOCA frequencies are assumed to be representative of a plant with IGSCC Category D and E welds operating with normal water chemistry (NWC). The pipe failure database includes plants with hydrogen water chemistry (HWC) and NWC. This study did not differentiate between plants with weld overlays and HWC versus plants with weld overlays and NWC, however. This study shows improved water chemistry together with weld reinforcements to lower the weld failure rates by about a factor of ten (10).
- Because of service conditions and piping arrangements, flow accelerated corrosion (FAC) is not viewed as a significant degradation mechanism affecting Code Class 1 feedwater piping. Degradation involving wall thinning is therefore not viewed as having an effect on the time-dependent LOCA frequency.

PWR-Specific Assumptions

- The estimation of RC-HL weld failure rates is based on the assumption that the observed (in 4th quarter 2000) weld degradation at V.C. Summer is a circumferential flaw in the RPV nozzle-to-safe-end weld. This assumption is believed to result in an over-estimation of the actual weld failure rate.
- Relative to PWRs of Westinghouse design, the pipe failure database includes no records on through-wall flaws in large-diameter pressurizer surge line welds. The analysis assumes that the piping is susceptible to thermal fatigue of sufficient magnitude to potentially cause a flaw in the through-wall direction.

D.7.2 Summary of Input Data and Results

Tables D.22 and D.23 summarize the input data to the LOCA frequency calculation. Tables D.24 through D.26 give the results at T = 25, 40 and 60 years, respectively. Consistent with the LOCA frequency elicitation structure, Table D.27 is summary of mid values (MV, or 50th percentiles) at T = 25 years rather than mean values, however. Figure D.41 shows the time-dependent LOCA frequencies. Figure D.42 shows selected weld failure rates. Figures D.43 through D.46 show the contribution to LOCA frequency by respective Base Case. Note that Figure D.45 includes the contribution to LOCA frequency by PWR-1 (Reactor Coolant System Hot Legs; all 3 loops are accounted for in this figure) and PWR-2 (Pressurizer Surge Line). Note that the Base Case results used in Table E.1 in the main body can be obtained from Tables D.16, D.17, and D.20 in this report.

Table D.22 Summary of Key BWR Base Case Input Data

Input Data	Base Case					
	BWR-1			BWR-2		
	NPS12	NPS22	NPS28	NPS12	NPS14	NPS20
Weld count	50	16	56	63	5	53
Weld failure rate Dominant [1/Reactor-yr.]	6.50E-05	1.54E-04	1.44E-04	2.20E-06	2.20E-06	1.58E-06
Weld failure rate Minimum [1/Reactor-yr.]	2.37E-05	3.32E-05	1.29E-05	1.77E-07	1.77E-07	1.73E-07

Table D.23 Summary of Key PWR Base Case Input Data

Input Data	Base Case		
	PWR-1	PWR-2	PWR-3
	NPS30	NPS14	NPS3-¾
Weld count	50	14	9
Weld failure rate Dominant [1/Reactor-yr.]	7.64E-05	1.56E-06	6.56E-04
Weld failure rate Minimum [1/Reactor-yr.]	1.05E-06	4.60E-08	1.58E-06

Table D.24 Calculated LOCA Frequencies (T = 25 Years)

Base Case	LOCA Frequency – Statistical Mean [1/Reactor-year] Flow Rate Threshold Value [gpm]					
	Cat1 v > 100	Cat2 v > 1,500	Cat3 v > 5,000	Cat4 v > 25,000	Cat5 v > 100,000	Cat6 v > 500,000
BWR-1[14]	9.46E-06	1.22E-06	4.60E-07	1.53E-07	3.05E-08	N/A[15]
BWR-2[16]	2.54E-06	3.36E-07	1.25E-07	4.09E-08	7.33E-09	N/A
PWR-1[17]	7.42E-07	7.62E-08	2.93E-08	1.09E-08	3.77E-09	1.26E-09
PWR-2	1.29E-07	1.50E-08	5.40E-09	1.56E-09	5.31E-10	N/A
PWR-3[18]	1.60E-05	2.32E-06	9.22E-07	N/A	N/A	N/A

Table D.25 Calculated LOCA Frequencies (T = 40 Years)

Base Case	LOCA Frequency – Statistical Mean [1/Reactor-year] Flow Rate Threshold Value [gpm]					
	Cat1 v > 100	Cat2 v > 1,500	Cat3 v > 5,000	Cat4 v > 25,000	Cat5 v > 100,000	Cat6 v > 500,000
BWR-1	1.14E-05	1.47E-06	5.54E-07	1.84E-07	3.78E-08	N/A
BWR-2	2.56E-06	3.39E-07	1.26E-07	4.13E-08	7.40E-09	N/A
PWR-1	8.96E-07	9.20E-08	3.54E-08	1.32E-08	4.55E-09	1.45E-09
PWR-2	1.60E-07	1.86E-08	6.70E-09	1.93E-09	6.59E-10	N/A
PWR-3	1.95E-05	3.30E-06	9.44E-07	N/A	N/A	N/A

Table D.26 Calculated LOCA Frequencies (T = 60 Years)

Base Case	LOCA Frequency – Statistical Mean [1/Reactor-year] Flow Rate Threshold Value [gpm]					
	Cat1 v > 100	Cat2 v > 1,500	Cat3 v > 5,000	Cat4 v > 25,000	Cat5 v > 100,000	Cat6 v > 500,000
BWR-1	1.88E-05	2.43E-06	9.16E-07	3.05E-07	6.07E-08	N/A
BWR-2	2.56E-06	3.39E-07	1.26E-07	4.13E-08	7.40E-09	N/A
PWR-1	9.74E-07	1.00E-07	3.85E-08	1.43E-08	4.95E-09	1.57E-09
PWR-2	1.77E-07	2.06E-08	7.41E-09	2.14E-09	7.29E-10	N/A
PWR-3	1.96E-05	3.32E-06	9.50E-07	N/A	N/A	N/A

[14] BWR-1 is the combination of RR Loop A and B.
[15] N/A = not applicable.
[16] The results are for FW Loop A and B.
[17] The results are for 3-of-3 RC hot legs.
[18] The results are for 2-of-2 HPI/NMU lines.

Table D.27 BWR and PWR LOCA Frequency Elicitation Anchor (MV) Values (T = 25 Years)

Base Case	Median (MV) LOCA Frequency [1/Reactor-year]					
	Flow Rate Threshold Value [gpm]					
	Cat1 v > 100	Cat2 v > 1,500	Cat3 v > 5,000	Cat4 v > 25,000	Cat5 v > 100,000	Cat6 v > 500,000
BWR-1	8.23E-06	1.08E-06	4.03E-07	1.29E-07	2.19E-08	N/A
BWR-2	1.09E-06	1.35E-07	5.03E-08	1.65E-08	2.10E-09	N/A
PWR-1	1.54E-07	2.25E-08	8.33E-09	2.85E-09	8.53E-10	1.58E-10
PWR-2	1.37E-08	1.39E-09	5.15E-10	1.54E-10	5.46E-11	N/A
PWR-3	6.87E-06	1.15E-06	2.14E-07	N/A	N/A	N/A

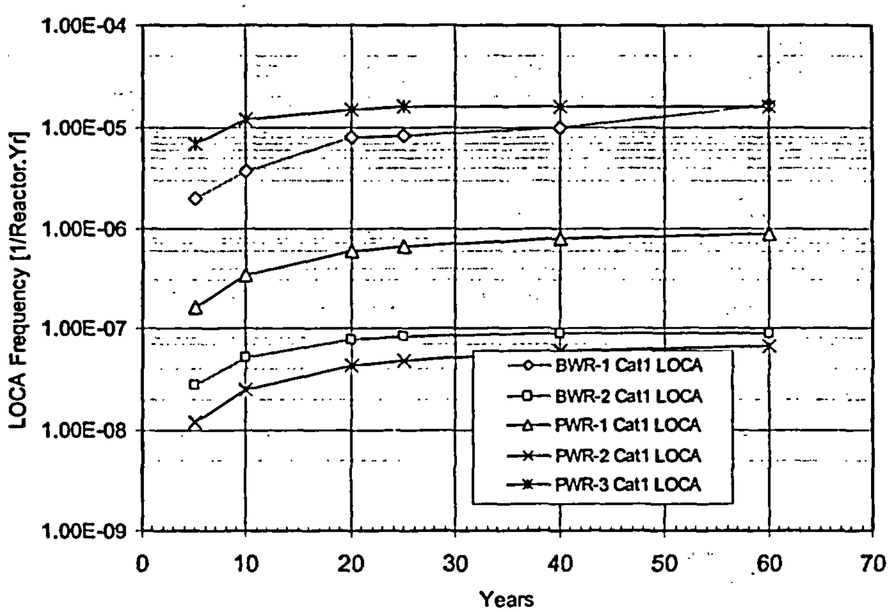

Figure D.41 Time-Dependent Cat 1 LOCA Frequencies

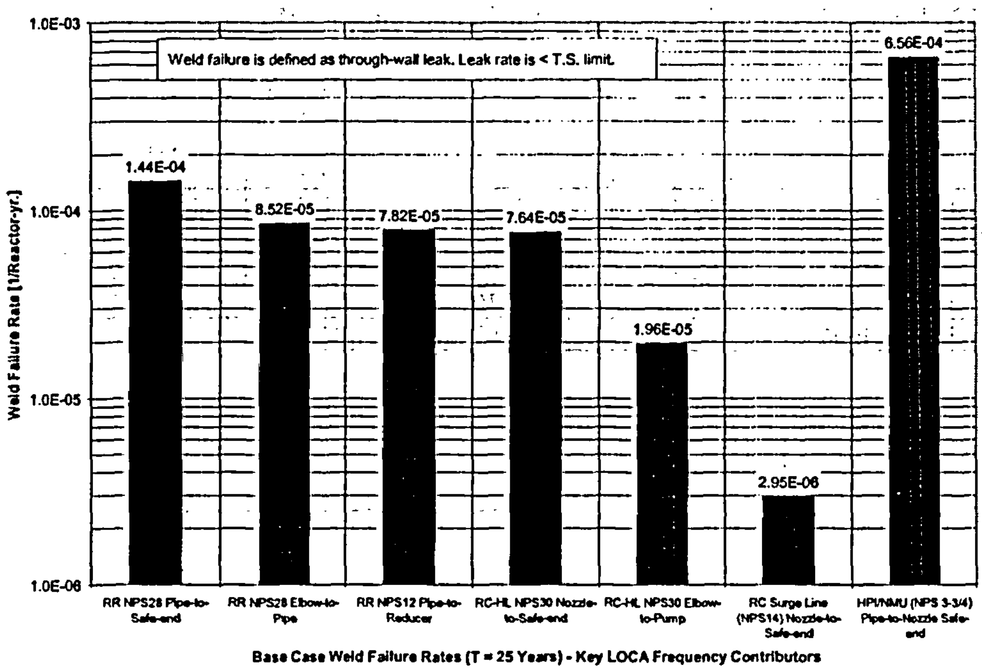

Figure D.42 Selected Base Case Weld Failure Rates

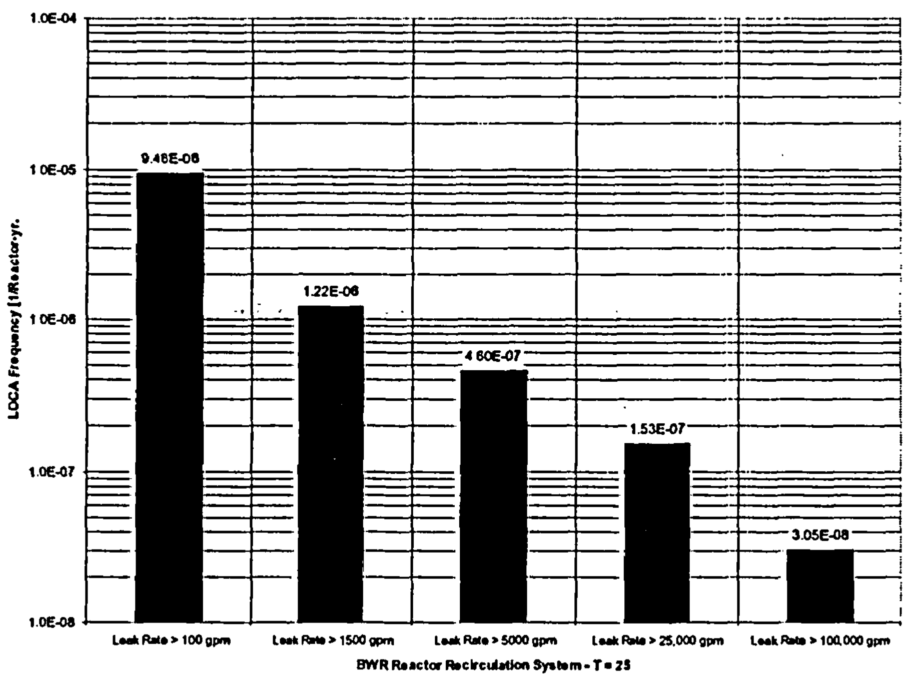

Figure D.43 BWR-1 LOCA Frequency

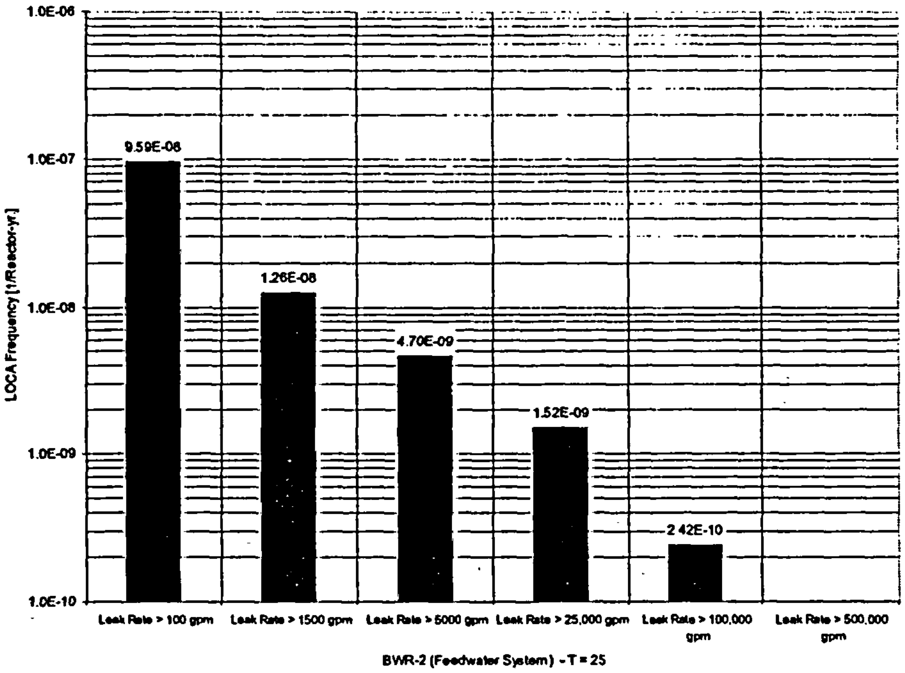

Figure D.44 BWR-2 LOCA Frequency (Feedwater System Loop A & B)

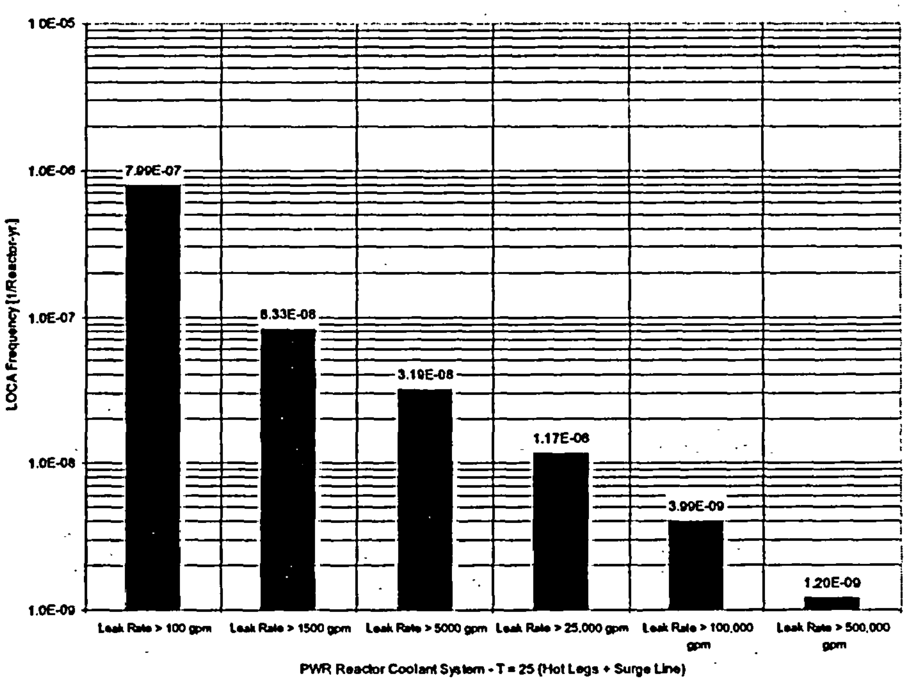

Figure D.45 PWR-1 and PWR-2 LOCA Frequency

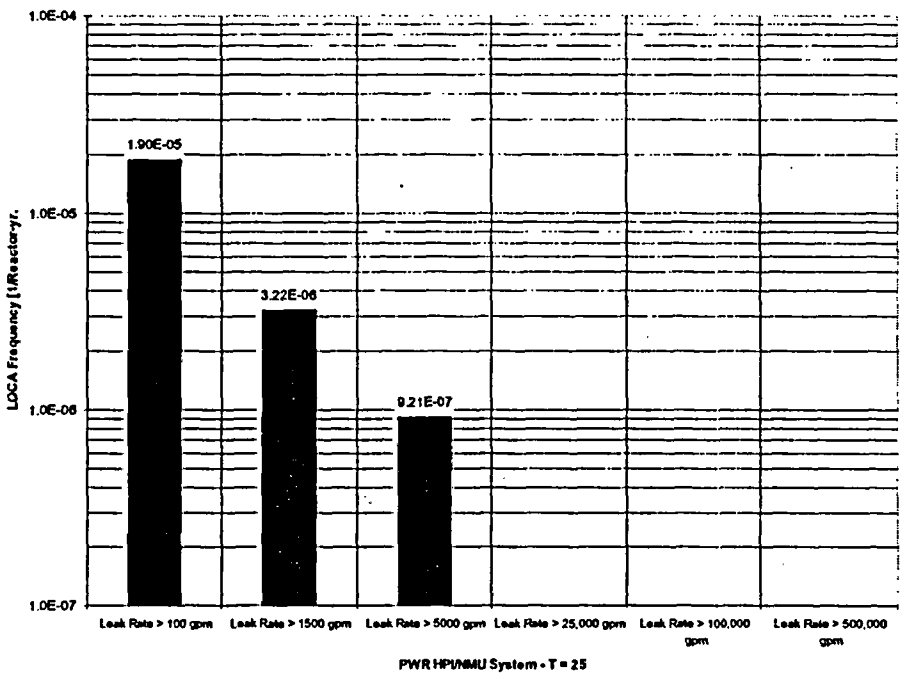

Figure D.46 PWR-3 LOCA Frequency (ASME Code Class 1 HPI/NMU System)

Figure D.47 displays the results of a sensitivity analysis associated with the BWR Base Cases. It is concerned with the influence by service data on the calculated LOCA frequency.

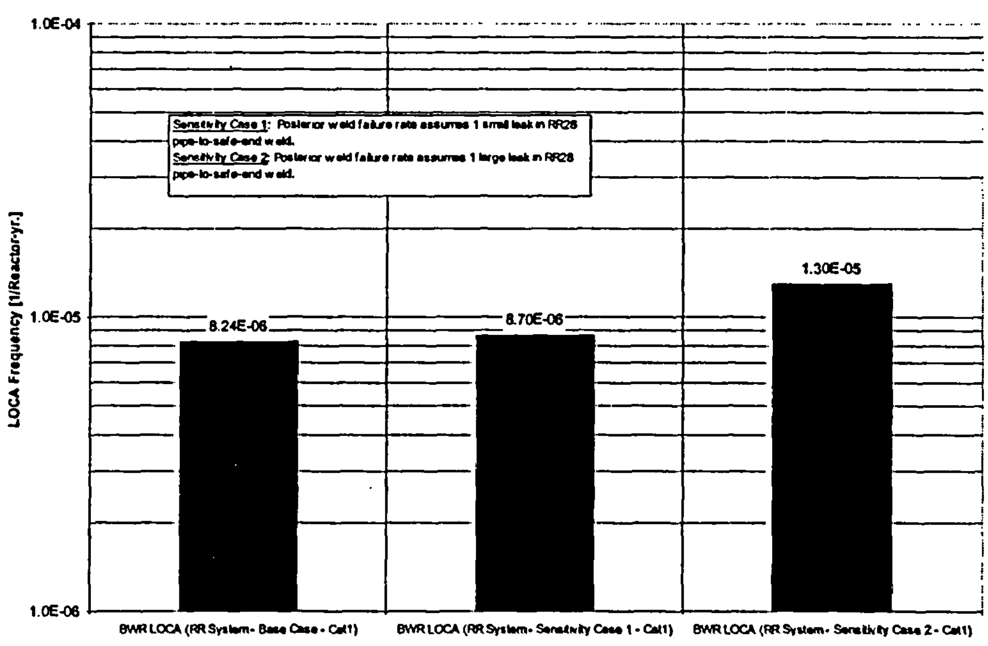

Figure D.47 Selected BWR Base Case Sensitivity Analysis Results – Cat 1 LOCA

Figures D.48 (BWR) and D.49 (PWR) show the influence of in-service inspection on the time-dependent LOCA frequency; no ISI and ISI with POD = 0.5 and 0.9, respectively.

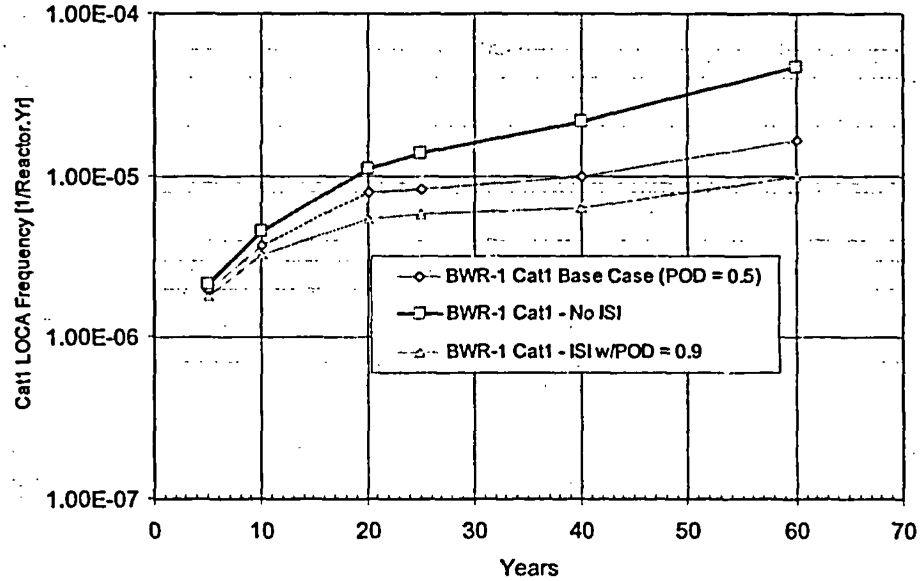

Figure D.48 Influence of ISI on Time-Dependent BWR-1 Cat 1 LOCA Frequency

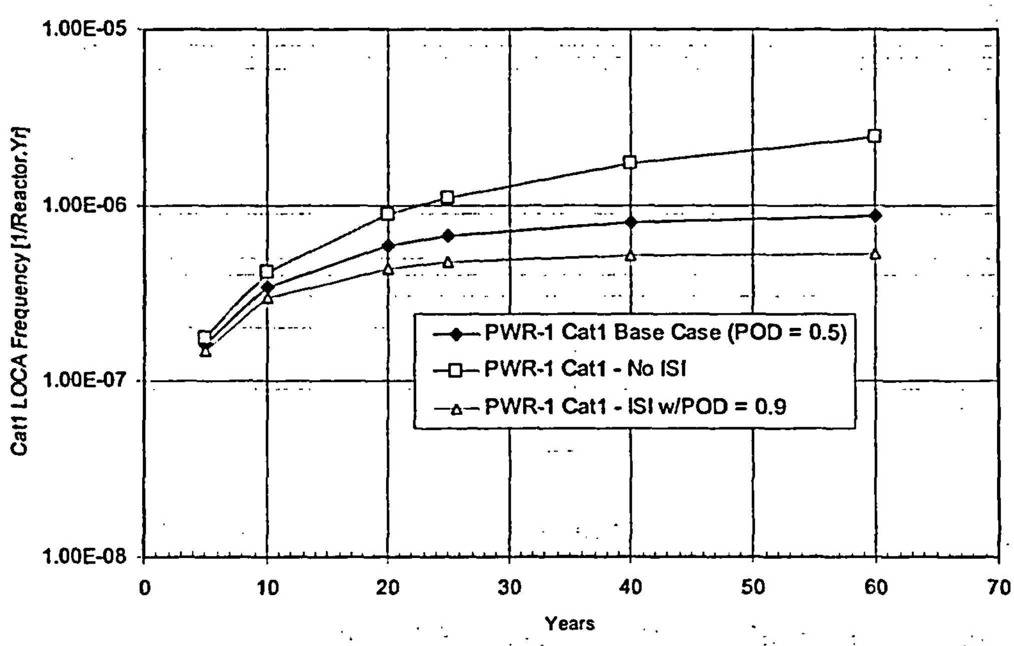

Figure D.49 Influence of ISI on Time-Dependent PWR-1 Cat 1 LOCA Frequency

D.7.3 Benchmarking

A limited scope benchmarking exercise was performed to compare predicted weld failure rates with the reported service experience. The benchmarking was limited to NPS12 BWR reactor recirculation welds susceptible to IGSCC. Probabilistic fracture mechanics (PFM) calculations using the WinPRAISE computer code generated predictions about the weld failure rate for different assumptions about the normal operating stresses (σ_{NO}).[19] Bayesian reliability analysis was used to derive weld failure rates from service experience data. Figure D.50 shows the results of the benchmarking exercise. Table D.28 includes a description of the different cases of the benchmarking exercise.

Table D.28 Benchmarking of WinPRAISE Versus Service Experience

Case	Definition
BOYL (PIPExp)	Table D.13 (this report). NPS12 Reactor Recirculation pipe-to-reducer weld with weld overlay. T = 25 years. This weld configuration has the highest predicted failure rate.
DOH-1 (D.O. Harris)	NPS12 reactor recirculation system weld with normal operating stress, σ_{NO} = 10 ksi[20]
DOH-2	NPS12 reactor recirculation system weld; σ_{NO} = 12 ksi
DOH-3	NPS12 reactor recirculation system weld; σ_{NO} = 15 ksi
DOH-4	NPS12 reactor recirculation system weld; σ_{NO} = 20 ksi

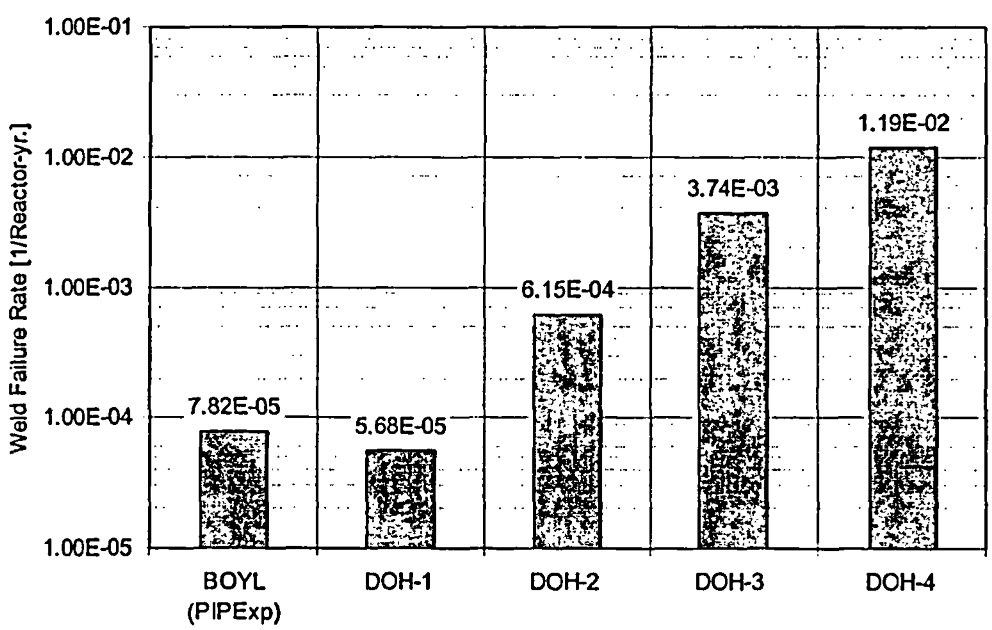

Figure D.50 Predicted (WinPRAISE) Versus Observed Weld Failure Rate (PIPExp)

D.7.4 Comparison to Historical LOCA Frequency Estimates

Figures D.51 and D.52 compare the Base Case results to historical LOCA frequency estimates. Direct (one-to-one) comparisons are not feasible due to different LOCA definitions and estimation approaches. Listed below are the selected BWR and PWR LOCA frequency references.

[19] D.O. Harris, "Progress in Benchmarking SCC for 12 Inch Recirculation Line, July 1, 2003.
[20] 1 ksi = 6.9 MPa

BWR Large (≥ Cat3) LOCA Frequency Estimates (Figure D. 51)

- SKI 98:30 (FW/RR); the displayed value range is taken from Reference [D.18]. It excludes contribution from thermal fatigue in Code Class 1 feedwater system piping. The feedwater system design is unique to the pilot plant in SKI Report 98:30 and it is therefore not applicable to BWR-2.
- NUREG/CR-5750 (Appendix J) provides recommended pipe LOCA frequencies. The given value range accounts for all Code Class 1 pipe failure contributions.
- GRS-98 is a probabilistic safety assessment of the German plant Gundremmingen; a BWR plant designed and built by Kraftwerk Union. This reactor design has no external recirculation loops; the given LOCA frequency value range accounts only for contributions from Code Class 1 feedwater pipe failure.
- BFN-1 (NUREG/CR-2802) is the 1982 probabilistic safety assessment of Browns Ferry Unit 1 performed as part of the NRC-sponsored Interim Reliability Evaluation Program. The given LOCA frequency value range accounts for Reactor Recirculation pump suction piping failure.

PWR Large (≥ Cat3) LOCA Frequency Estimates (Figure D. 52)

- NUREG/CR-5750 (Appendix J) provides recommended pipe LOCA frequencies. The given value range accounts for all Code Class 1 pipe failure contributions.
- Surry-1 (1990 Expert Elicitation). Surry-1 is a 3-loop Westinghouse reactor, similar to the PWR-1/PWR-2 reference design. The given LOCA frequency value range applies to RCS pipe failure and resulted from a NRC-sponsored expert elicitation.[21]
- EPRI TR-100380 (Piping Failures in U.S. Commercial Nuclear Power Plants, 1992) includes recommended BWR and PWR LOCA frequencies that are based on statistical analysis of service data. The given LOCA frequency value range accounts for all Code Class 1 pipe failure contributions.

[21] See for example T.V. Vo et al (1991). "Estimates of Rupture Probabilities for Nuclear Power Plant Components: Expert Judgment Elicitation," Nuclear Technology, 96:259-270.

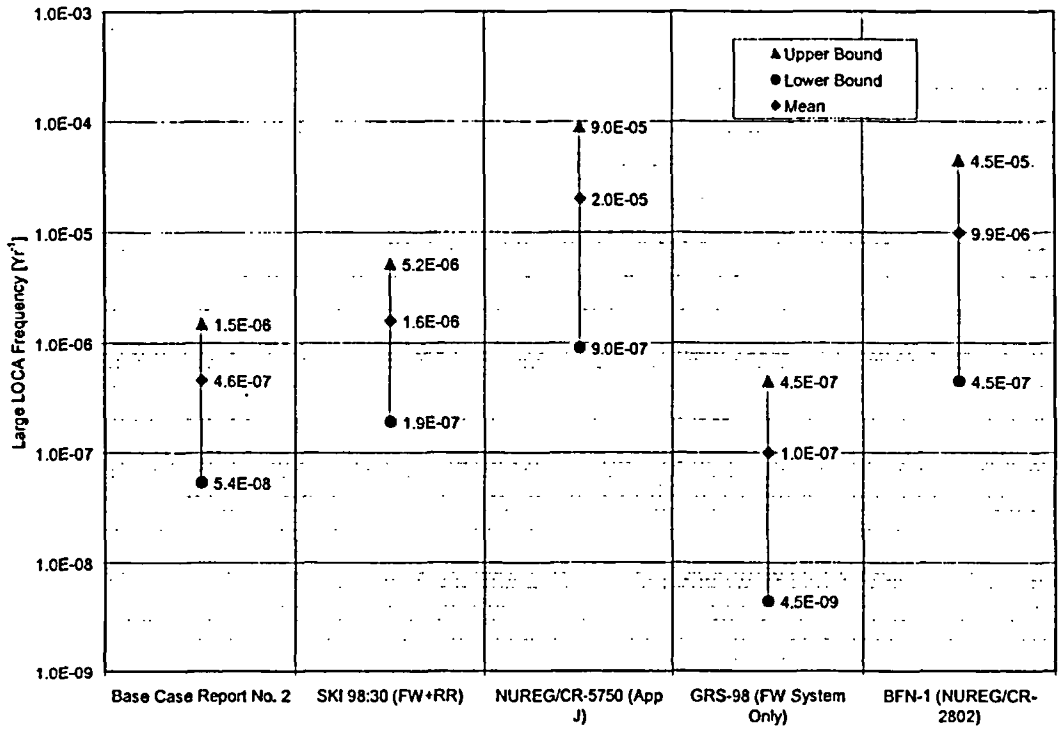

Figure D.51 Comparison of Selected BWR Large LOCA Frequency Estimates

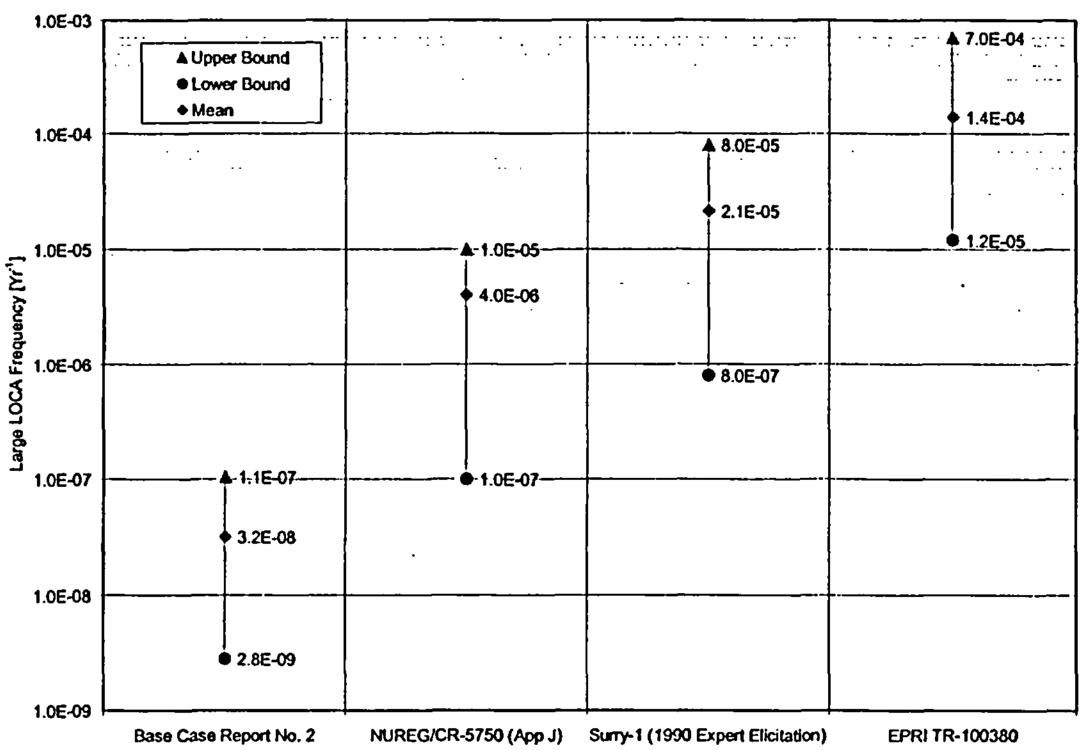

Figure D.52 Comparison of Selected PWR Large LOCA Frequency Estimates

D.8 References

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ATTACHMENT A TO APPENDIX D

SUMMARY OF PIPEXP DATABASE

The PIPExp database has evolved over a period of about ten years. A first database version was developed with financial support from the Swedish Nuclear Power Inspectorate (SKI). Since the conclusion of the initial R&D effort in 1998 an active maintenance program has supported the database.

D.A.1 Database Structure

Designed in Access, the database consists of searchable free-format text fields and a large number of data fields that are used as data filters in support of a range of data processing needs. Table D.A.1 is a summary of text and data fields.

D.A.2 Completeness and Quality Management

The completeness of the pipe failure data is addressed through a continuous database management program. Extracted from Monthly Summary Reports, Table D.A.2 provides snapshots of the database evolution from 1998 to the present. In PIPExp, each record is assigned a 'Quality Index' (Table D.A.1, item #4, and Table D.A.3) as one means of monitoring the completeness and technical accuracy of source information as well as the process of classifying and coding of the source information.

Table D.A.1 Description of Data Fields in PIPExp

Item No.	Field Name	Type	Description
1	UPDATE	Date	Date of the most recent update.
2	MER	Yes/No[22]	Multiple Events Report; some reports include information on more than one crack/leak in one system. Used to identify events where a discovery resulted in an investigation (e.g., augmented ISI) to identify further piping degradation due to a common cause. A new record is added if additional degradation is positively identified (by component socket).
3	DDA	Text	Data filter used to classify a record as either 'public' (= Licensee Event Report), 'restricted' or 'proprietary.'
4	QA-Index	Number	QA-Index of '1' signifies a data entry determined to be 'complete.' By contrast, a QA-Index of '6' signifies a database entry for which only a LER (or equivalent) abstract was available.
5	EVENT DATE	Date	Event date (MM/DD/YY); date of discovery (in case of ISI).
6	PLANT TYPE	Text	Plant type; e.g., BWR, PWR, used as data filter.
7	DESIGN	Text	NSSS design/design generation; keyword using generally accepted or standard nomenclature. This field is used as a data filter.
8	NSSS-VENDOR	Text	Reactor vendor; e.g., ABB-Atom, KWU/Siemens, Westinghouse; used as data filter.
9	PLANT NAME	Text	Plant name
10	COUNTRY	Text	Two-letter code based on the ISO 3166-1-alpha-2 code elements.
11	CONSTRUCTOR	Text	Name of company responsible for the original piping system design. The default name is the architect engineering firm. Used as data filter.
12	COD	Date	Date (MM/DD/YY) of commercial operation as default. If known, date of initial criticality. For U.S. data, based on NUREG-0020
13	PLANT	Text	Plant operational state (at the time of discovery); keyword using generally accepted or standard nomenclature. This field is used as a data

[22] A check box without check mark implies 'No' or 'Unknown/Pending.'

Table D.A.1 Description of Data Fields in PIPExp

Item No.	Field Name	Type	Description
	OPERATIONAL STATE - POS		filter. Pulldown menu with the following options: ▪ CSD – Cold Shutdown ▪ HSD – Hot Shutdown ▪ HSB – Hot Standby ▪ Refueling ▪ Shutting Down ▪ Starting Up ▪ Power Operation
14	REFERENCE-1	Text	Primary reference
15	REFERENCE-2	Text	Secondary (or supplemental) reference
16	REFERENCE-3	Text	Tertiary (or supplemental) reference
17	LER-RO?	Yes/No	Check if the information source is a Licensee Event Report (or equivalent); i.e., from a regulatory reporting system.
18	EVENT TYPE	Text	Event type; 'Crack', 'Wall Thinning', 'P/H-leak' (P/H = pinhole), 'Leak', 'Severance', 'Rupture.' Used as data filter.
19	FAILURE-ON-DEMAND	Yes/No	Check if pipe failure occurred when a demand was placed on the affected system (e.g., standby system). Used as data filter.
20	SYNERGY	Yes/No	Check if the pipe failure was caused by multiple degradation mechanisms; e.g., crack initiation through IGSCC and crack propagation through thermal fatigue. Used as data filter.
21	DEGRADATION+ LOADING	Yes/No	Check if the pipe failure resulted from the combined effect of a degradation mechanism (e.g., flow-accelerated corrosion, FAC) and a severe (or unusual) loading condition. Used as data filter.
22	ECA	Text	Event Category. Used as data filter. This database field is used to characterize actual or potential impact on plant risk by a degradation or failure. The following options are available: ▪ S-/M-/L-LOCA (implies that a pressure boundary failure resulted in ESF actuation); ▪ S-/M-/L-LOCA Precursor (implies mitigation of a pressure boundary failure through prompt operator response; e.g., plant shutdown prior to reaching ESF actuation setpoint); ▪ Internal Flooding (spill rate in excess of room/compartment floor drain capacity); ▪ Internal Flooding Precursor (accumulation of large water volumes prevented through prompt operator response); ▪ Common Cause Initiating (CCI) Event (pressure boundary failure results in spatial effects through spraying or steaming of safety equipment); ▪ CCI Precursor (pressure boundary failure results in spraying or steaming but prompt operator action prevents safety equipment from being affected); ▪ System Disabled (pressure boundary failure is large enough to incapacitate a system function); ▪ System Degraded (default used for at-power events that result in an entry into a Technical Specification Action Statement).
23	CCC	Yes/No	Check if event is considered to be a 'common cause candidate' (CCC) event. Used as data filter.
24	CA	Text	Corrective Action. Used as data filter. The following types of corrective action are defined: ▪ REPAIR (used in a generic sense); ▪ REPLACEMENT ▪ REPLACEMENT – IN-KIND ▪ REPLACEMENT – NEW MATERIAL ▪ TEMP. REPAIR (temporary repair to allow continued operation until next refueling outage or major maintenance outage at which time a Code-repair (which would require system isolation and

Table D.A.1 Description of Data Fields in PIPExp

Item No.	Field Name	Type	Description
			draining) or a replacement is performed. ▪ WOR (= weld overlay repair); primarily applies to ASME Section XI Class 1 or 2 (or equivalent) piping.
25	ISS	Yes/No	Safety system actuation; check if pipe failure resulted automatic actuation of a make-up system or other safety system. Used as data filter.
26	IRT	Yes/No	Automatic reactor trip; check if pipe failure resulted in automatic reactor trip/turbine trip. Used as data filter.
27	IPO	Text	Impact of pipe failure on plant operation; e.g., power reduction, manual reactor trip. Used as data filter.
28	TTR	Number	Repair time in hours.
29	TTR-Class	Number	A data filter: 1: TTR ≤ 8 hours; 2: 8 < TTR ≤ 24 hours; 3: 24 < TTR ≤ 96 hours; 4: 96 < TTR ≤168 hours; 5: TTR > 168 hours.
30	NARRATIVE	Memo	Event narrative; includes details on plant condition prior to event and plant response during event, method of detection, corrective action plan. This field should include sufficient information to support independent verification of the event data classification.
31	LQT	Number	Quantity of process medium released [kg]
32	DOL	Text	Duration of release
33	LRT	Number	Leak rate [kg/s]
34	GPM	Number	Leak rate [U.S. gallons/minute]
35	LEAK CLASS	Number	A data filter: 1: Leak Rate (LR) ≤ 1 gpm; 2: 1 < LR ≤ 5 gpm; 3: 5 < LR ≤ 10 gpm; 4: 10 < LR ≤ 50 gpm; 5: LR > 50 gpm.
36	FLO	Text	Location of crack/leak/rupture; description of where in the piping system a degradation or failure occurred. Include sufficient detail to support the consequence evaluation/classification.
37	K1	Yes/No	Data filter; steamline break outside containment.
38	K2	Yes/No	Data filter; feedwater line break
39	K3	Yes/No	Data filter; steamline break inside containment.
40	IMPULSE-LINE	Yes/No	Check (= 'Yes') if affected line is a valve impulse line.
41	INSTR. LINE	Yes/No	Check (= 'Yes') if affected line is an instrument sensing line.
42	ISOMETRIC DRAWING #	Text	Isometric drawing number
43	P&ID #	Text	Piping and instrument drawing number.
44	MSA	Text	Name of the affected plant system
45	SHARED	Yes/No	Check (= 'Yes') if affected piping is shared by two reactor units. Mainly applies to support systems (e.g., Service Water, Instrument Air) where sections of a piping system may be shared by two reactor units; this is relatively common in the U.S.
46	OSA	Text	Name of other systems affected by the degradation or failure. Secondary effects of piping failure
47	S-TYPE	Text	Category of system affected by the degradation or failure. Used as data filter. The following types are used: • RCPB (Reactor Coolant Pressure Boundary); • SIR (Safety Injection & Recirculation); includes emergency core cooling systems & decay heat removal). • CS (Containment Spray) • AUX (Reactor Auxiliary Systems); includes component cooling water, chemical & volume control, reactor water cleanup, control rod drive, containment heat removal, standby liquid control, radwaste control, spent fuel pool cooling. • FWC (Feedwater & Condensate Systems) • STEAM (Main Steam System) • SUPPORT (Service Water & Instrument Air systems) • PCS (turbine generator)

Table D.A.1 Description of Data Fields in PIPExp

Item No.	Field Name	Type	Description
			• FIRE (Fire Protection).
48	ISO	Yes/No	Check if the affected pipe section can be isolated to prevent or mitigate direct/indirect impacts.
49	DET	Text	Method of detection; e.g., ISI, WT = walk-through inspection, leak detection system in combination with control room indication and/or alarm. Used as data filter. Pulldown menu with the following options: ▪ Walk-through ▪ UT-examination ▪ Liquid penetrant testing ▪ Hydrotesting ▪ Leak detection ▪ Containment/drywell inspection ▪ Control Room Indication
50	DRYWELL ENTRY	Yes/No	BWR-specific data field. Checked for 'at-power', unidentified P/H-leak or leak requiring power reduction or reactor shutdown for containment drywell entry to determine leak source. Used as a data filter. Also, this data could be input to plant availability models.
51	CONTAINMENT ENTRY	Yes/No	Check if power reduction initiated to allow for containment entry to identify source of leakage. Used for other than BWR plants.
52	CRS	Text	Verbal description of crack morphology; orientation and size/ geometry of crack or fracture
53	CRACK-DEPTH	Number	Crack depth in percent of wall thickness (a/t-ratio)
54	AXIAL-LENGTH	Number	Axial crack length in [mm].
55	CRACK-LENGTH	Number	Circumferential crack length as percent of inside diameter
56	ASPECT-RATIO	Number	Ratio of crack depth (a) to flaw length (L)
57	WELD-CONFIG	Text	Configuration of the affected weld in a piping system; e.g., BP = bend-to-pipe weld, PP = pipe-to-pipe weld, etc.
58	INSIDE CONTAINMENT	Yes/No	Check if pipe failure located inside containment
59	AUXILIARY BUILDING	Yes/No	Check if pipe failure located in Auxiliary Building (PWR)
60	REACTOR BUILDING	Yes/No	Check if pipe failure located in Reactor Building (BWR)
61	TURBINE BUILDING	Yes/No	Check if pipe failure located in Turbine Building
62	Not used	N/A	N/A
63	Not used	N/A	N/A
64	CTA	Text	Component Type; pulldown menu with the following options: ▪ Bend ▪ Elbow ▪ Elbow – 45-degree ▪ Elbow – 90-degree ▪ Elbow – LR (Long Radius) ▪ Pipe ▪ Reducer ▪ Tee ▪ Weld ▪ Socket weld
65	ASME Class	Number	Differentiate between 1, 2, 3 and 4 (= non-Code Class)
66	BELOW-GRADE	Yes/No	Check if 'Yes'; Below Grade / Underground Piping. Used as data filter.
67	FIELD-WELD	Yes/No	Check if 'Yes'. Used as data filter.
68	SHOP-WELD	Yes/No	Check if 'Yes'. Used as data filter.
69	CONCRETE-LINED	Yes/No	Check if 'Yes'; could apply to essential or non-essential service water (or equivalent) system piping. Used as data filter.
70	REPLACEMENT	Yes/No	Check if piping replaced using new material.

Table D.A.1 Description of Data Fields in PIPExp

Item No.	Field Name	Type	Description
71	REPL-DATE	Date/Time	Date of component (e.g., weld and spool piece) replacement. Used in hazard plotting.
72	YOO	Number	Years of commercial operation when failure occurred. Used in aging analysis.
73	AGE	Number	Age of component socket [hours]. Used in hazard plotting. For additional information.
74	CLASS	Number	Based on diameter; events grouped in six diameter classes; 1 = (≤ DN15), 2 = (15 < DN ≤ 25), 3 = (25 < DN ≤ 50), 4 = (50 < DN ≤ 100), 5 = (100 < DN ≤ 250), 6 = (> DN250). DN = nominal diameter in [mm]. This field is used as a data filter.
75	THOMAS	Number	Ratio of diameter and pipe wall thickness ([CSI/WTK]); for details, see the paper by H.M. Thomas (1981): "Pipe and Vessel Failure Probability," *Reliability Engineering*, 2:83-124. This field is used as a data filter.
76	CSI	Number	Nominal diameter [DN] in [mm]. Used as data filter.
77	WTK	Number	Wall thickness [mm]
78	SCHEDULE	Number	Pipe schedule number
79	DIS-MET	Yes/No	Dissimilar metal weld; check if 'yes'. Used as data filter.
80	MTR	Text	Material; e.g., carbon steel, stainless steel, etc. Used as data filter.
81	MTR-DES	Text	Material designation according to national standard; e.g., AISI 304, SS2343, etc. Used as data filter.
82	PMD	Text	Process medium. Used as data filter.
83	RAW WATER	Text	Source of raw water (applies to Fire Protection and Service Water piping); differentiate between LAKE – RIVER – SEA-BRACKISH. Used as data filter.
84	STG	Yes/No	Normally stagnant process medium? Used as data filter.
85	HWC	Yes/No	For BWRs; hydrogen water chemistry; check if 'Yes'. Used as data filter (e.g., in factor-of-influence assessments).
86	HWC-START	Date/Time	Date when HWC was introduced
87	NMCA	Yes/No	Check if Noble Metal Chemical Addition. Used as data filter.
88	NMCA-Start	Date/Time	Date when NMCA started.
89	IHSI	Yes/No	Induction heat stress improvement; check if 'Yes'. Used as data filter.
90	IHSI-DATE	Date/Time	Date when IHSI was performed
91	MSIP	Yes/No	Check if Mechanical Stress Improvement Process applied to weld. Used as data filter.
92	MSIP-Date	Date/Time	Date of MSIP application
93	S-A	Number	Stress intensity allowance; ratio of the critical stress intensity factor to the assessed stress intensity factor given a flaw. This information is extracted from fracture mechanics evaluations.
94	OPA	Number	Operating temperature [°C]
95	DPA	Number	Design temperature [°C]
96	OPB	Number	Operating pressure [MPa]
97	DPB	Number	Design pressure [MPa]
98	OPC	Text	Process medium chemistry (for primary system); e.g., NWC = normal water chemistry, HWC = hydrogen water chemistry. Used as data filter.
99	MPR	Text	Method of fabrication; e.g., cold formed, hot formed. Used as data filter.
100	SYS	Yes/No	Systematic failure? Used as data filter to enable queries that address the effectiveness of remedial actions (e.g., preventing recurring failures).
101	RFL	Text	Description of the extent and nature of a systematic failure
102	REST	Yes/No	Failure due to deficient system restoration?; e.g., no venting prior to fill procedure, etc. Used as data filter.
103	CEA	Text	Apparent cause of failure; e.g., IGSCC, PWSCC, TGSCC, etc. Used as data filter. Pull down menu with the following options: • B/A SCC (B/A = boric acid) • Corrosion (general, pitting or crevice corrosion)

Table D.A.1 Description of Data Fields in PIPExp

Item No.	Field Name	Type	Description
			• Corrosion-fatigue • Erosion • Erosion-cavitation • Flow accelerated corrosion (FAC) • Fretting • HF: Construction/installation error • HF: Human error • HF: Repair/maintenance error • HF: Welding error • HPSCC (High Potential SCC) • IGSCC • MIC (Microbiologically induced corrosion) • Overpressurization • PWSCC • Severe overloading (other than water hammer) • SICC (Strain-rate induced SCC) • TGSCC • Thermal fatigue • Unknown • Unreported • Vibration • Vibration-fatigue • Water hammer
104	EPRI-CODE	Text	Failure code as used in the EPRI '97 / EPRI '98 databases (see for example EPRI TR-110161 (Piping System Reliability and Failure Rate Estimation Models for Use in Risk-Informed In-Service Inspection Applications, December 1998). Used as data filter. Pull down menu with the following options: ▪ CF – Corrosion-fatigue ▪ COR – General corrosion, microbiologically induced corrosion (MIC), pitting corrosion ▪ COR-EXT – external corrosion ▪ D&C – Design & Construction errors ▪ E-C – Erosion-cavitation ▪ E/C – Erosion-corrosion ▪ FRET – Fretting ▪ HE – Human Error ▪ OVP – Overpressurization ▪ SCC – Stress corrosion cracking ▪ TF – Thermal Fatigue ▪ UNK – Unknown ▪ VF – Vibration-fatigue ▪ WH – Water hammer
105	RC1	Text	Contributing factor number 1
106	RC2	Text	Contributing factor number 2
107	CEC	Memo	Description of events and causal factors. Include sufficient technical detail from the root cause analysis process so that recurrence may be prevented.
108	CMT	Memo	Any other information of relevance to understanding of underlying causal factors. Also, information on the type and extent of repair/replacement. The purpose of this free-format database field is to facilitate future applications, for example, by codifying the information on piping replacements.

Table D.A.1 Description of Data Fields in PIPExp

Item No.	Field Name	Type	Description
109	ISI	Yes/No	Deficient ISI; e.g., ISI not performed, or ISI failed to detect a flaw. Used as data filter to identify events caused by ISI program deficiencies (e.g., affected component should have been included in program) or an inspection prior to failure missed a degradation that propagated in the through-wall direction.
110	ISI-CMT	Memo	Comments on ISI history; e.g., date of last inspection, details on examination technique(s).
111	mCF	Yes/No	Check if there are multiple circumferential flaws in a weld.
112	Number of Flaws	Number	
113	D0-1	Number	Distance from 12 o'clock position to the first circumferential flaw; this field is repeated for up to nine flaws.
114	CF-1	Number	Length of the first circumferential flaw (counted from the 12 o'clock position

Table D.A.2 Summary of PIPExp Database Development

Plant Type	Database as of 12-31-1998 No. Damage Reports by QA-Index[23]							Database as of 12-31-1999 No. Damage Reports by QA-Index						
	Totals	1	2	3	4	5	6	Totals	1	2	3	4	5	6
BWR	673	210	66	3	74	7	277	1595	1000	168	2	146	53	226
PHWR	100	30	3	--	56	1	10	100	30	3	--	56	1	10
PWR	1376	386	123	6	152	84	746	1656	645	176	5	211	208	411
RBMK	57	3	6	--	19	28	1	66	7	6	--	22	31	--
	2291	629	198	9	301	120	1034	3417	1682	253	7	435	243	647

Plant Type	Database as of 12-31-2000 No. Damage Reports by QA-Index							Database as of 12-31-2001 No. Damage Reports by QA-Index						
	Totals	1	2	3	4	5	6	Totals	1	2	3	4	5	6
BWR	1711	1111	164	2	175	58	201	1784	1172	166	2	197	63	184
PHWR	95	43	1	--	41	10	--	96	44	1	--	41	10	--
PWR	1748	696	181	5	260	209	397	1952	811	194	5	329	221	392
RBMK	125	12	5	1	77	30	--	125	12	5	1	77	30	--
	3679	1862	351	8	553	307	598	3957	2039	366	8	644	324	576

Plant Type	Database as of 12-31-2002 No. Damage Reports by QA-Index							Database as of 06-01-2004 No. Damage Reports by QA-Index						
	Totals	1	2	3	4	5	6	Totals	1	2	3	4	5	6
BWR	1872	1216	174	12	219	75	176	2033	1370	189	77	233	93	71
GCR, HWLWR	--	--	--	--	--	--	--	12	12	--	--	--	--	--
PHWR	106	51	2	--	42	11	--	101	47	1	--	41	12	--
PWR	2077	1011	198	6	351	233	278	2280	1213	218	103	344	270	132
RBMK	160	48	--	--	18	81	--	160	12	5	4	109	30	--
	4215	2290	379	22	721	349	454	4586	2654	413	184	727	405	203

[23] See Table A-3 for a definition of QA-Index.

Table D.A.3 Definition of QA Index for Database Management

QA-Index	Definition
1	Validated – all source data has been accessed & reviewed – no further action required
2	Validated – source data may be missing some, non-critical information – no further action anticipated
3	Validated – incomplete source data – assumptions made about material grade and/or exact flaw location – no further action anticipated
4	Validation based on incomplete information – depending on application requirements, further action may be necessary
5	Validation based on available, incomplete information – further action expected (e.g., retrieval of additional source data)
6	Not validated – validation is pending, or record is subject to deletion from database

Table D.A.4 Summary of Through-Wall Cracks in BWR Reactor Recirculation Piping[24]

EID	Date of Detection	Plant	NPS	Comment
1978	06-14-1978	Duane Arnold	10	LER 50-331/78-030. Active, at power leak
1067	01-26-1983	Brunswick-1	12	Weld B32-RR-12-BR-H4
2419	02-12-1980	Santa Maria de Garona	10	Active, 0.8 gpm leak
1397	02-21-1985	Browns Ferry-2	12	Riser-to-manifold weld
404	11-02-1982	Monticello	12	LER 50-220/82-013; Riser-to-safe-end weld
7414	01-15-1986	Hatch-1	12	Weld 1B31-1RC-12BR-B-3
3231	11-22-1982	Monticello	12	Leak detected during hydrostatic testing
2681	04-01-1985	Quad Cities-2	12	Weld 02M-F7
1848	11-05-1986	Quad Cities-2	12	Weld 02K-S3
2085	10-20-1982	Monticello	22	Pipe-to-safe-end weld
443	10-31-1982	Monticello	22	Pipe-to-safe-end weld
2850	11-06-1982	Hatch-1	22	LER 50-321/82-089; End cap-to-manifold
3217	03-10-1985	Duane Arnold	22	Weld RHB-J1
7543	03-23-1982	Nine Mile Point-1	28	LER 50-220/82-009; Weld P32-FW-17-W
7542	03-23-1982	Nine Mile Point-1	28	Weld P32-FW-22-W
7538	04-15-1982	Nine Mile Point-1	28	Weld P32-FW-27-W
437	04-15-1985	Nine Mile Point-1	28	Weld P32-FW-42-W
3224	03-10-1985	Duane Arnold	28	Weld RCB-J36
2839	07-18-1985	Brunswick-1	28	Weld 1B32-RR-28-A-4
2838	07-18-1985	Brunswick-1	28	Weld 1B32-RR-28-A-15
2837	07-18-1985	Brunswick-1	28	Weld 1B32-RR-28-B-8
2836	07-18-1985	Brunswick-1	28	Weld 1B32-RR-28-B-4
1711	07-18-1985	Brunswick-1	28	Weld 1B32-RR-28-A-14
3183	01-09-1986	Brunswick-2	28	Weld 2B32-RR-28-B-3
3182	01-09-1986	Brunswick-2	28	Weld 2B32-RR-28-B-4
3181	01-09-1986	Brunswick-2	28	Weld 2B32-RR-28-B-5
3180	01-09-1986	Brunswick-2	28	Weld 2B32-RR-28-B-11
1723	01-09-1986	Brunswick-2	28	Weld 2B32-RR-28-A-4

[24] Basis for Table D.5 (Section D.4).

ATTACHMENT B TO APPENDIX D

BASIS FOR LOCA FREQUENCY MODELS

Attached as Tables D.B.1 (RR System Loop B) and D.B.2 (FW System Loop B) are the Excel spreadsheets on which the BWR LOCA frequency model is based. Attached as Tables D.B.3 (RC-HL), D.B.4 (RC Surge Line), and D.B.5 (HPI/NMU Line) are the spreadsheets on which the PWR LOCA frequency model is based.

The input to the calculation of a selection of posterior weld failure rates is summarized in Tables D.B.6 (BWR-1, NPS28), D.B.7 (BWR-2, NPS12), D.B.8 (PWR-1), D.B.9 (PWR-2), and D.B.10 (PWR-3).

Table D.B.1 BWR-1 – RR System Loop B Weld List

System ID	Loop	Exam Category	Category Item	Line Number	Weld Order	Component ID	NPS (In)	Wall Thk (In) or Schedule	Configuration	Description
B31	B	B-F	B5.10	5358-5(2)	5	101-304E	12	Schd. 80	Nozzle-safe-end	RRI Nozzle-to-Safe End Butt Weld (N2E)
B31	B	B-F	B5.10	5358-5(5)	5	2-303A	12	Schd. 80	Nozzle-safe-end	RRI Nozzle-to-Safe End Butt Weld (N2A)
B31	B	B-F	B5.10	5358-5(4)	5	2-303B	12	Schd. 80	Nozzle-safe-end	RRI Nozzle-to-Safe End Butt Weld (N2B)
B31	B	B-F	B5.10	5358-5(6)	4	2-303C	12	Schd. 80	Nozzle-safe-end	RRI Nozzle-to-Safe End Butt Weld (N2C)
B31	B	B-F	B5.10	5358-5(3)	5	2-303D	12	Schd. 80	Nozzle-safe-end	RRI Nozzle-to-Safe End Butt Weld (N2D)
B31	B	B-J	B9.11	5358-5(5)	1	FW-RD-2-B10	12	Schd. 80	Pipe-sweepolet	B31-Reactor Recirc - Loop B Circ Weld
B31	B	B-J	B9.11	5358-5(4)	1	FW-RD-2-B11	12	Schd. 80	Pipe-sweepolet	B31-Reactor Recirc - Loop B Circ Weld
B31	B	B-J	B9.11	5358-5(6)	0.5	FW-RD-2-B12	12	Schd. 80	Pipe-reducer	
B31	B	B-J	B9.11	5358-5(3)	1	FW-RD-2-B13	12	Schd. 80	Pipe-sweepolet	B31-Reactor Recirc - Loop B Circ Weld
B31	B	B-J	B9.11	5358-5(2)	1	FW-RD-2-B14	12	Schd. 80	Pipe-sweepolet	B31-Reactor Recirc - Loop B Circ Weld
B31	B	B-J	B9.11	5358-5(5)	4	FW-RD-2-B15	12	Schd. 80	Pipe-safe-end	
B31	B	B-J	B9.11	5358-5(4)	4	FW-RD-2-B16	12	Schd. 80	Pipe-safe-end	B31-Reactor Recirc - Loop B Circ Weld
B31	B	B-J	B9.11	5358-5(6)	3	FW-RD-2-B17	12	Schd. 80	Pipe-safe-end	
B31	B	B-J	B9.11	5358-5(3)	4	FW-RD-2-B18	12	Schd. 80	Pipe-safe-end	B31-Reactor Recirc - Loop B Circ Weld
B31	B	B-J	B9.11	5358-5(2)	4	FW-RD-2-B19	12	Schd. 80	Pipe-safe-end	B31-Reactor Recirc - Loop B Circ Weld
B31	B	B-J	B9.11	5358-5(5)	2	SW-RD-2-B4-W1	12	Schd. 80	Pipe-elbow	
B31	B	B-J	B9.11	5358-5(5)	3	SW-RD-2-B4-W2	12	Schd. 80	Elbow-pipe	B31-Reactor Recirc - Loop B Circ Weld
B31	B	B-J	B9.11	5358-5(4)	2	SW-RD-2-B5-W1	12	Schd. 80	Pipe-elbow	
B31	B	B-J	B9.11	5358-5(4)	3	SW-RD-2-B5-W2	12	Schd. 80	Elbow-pipe	B31-Reactor Recirc - Loop B Circ Weld
B31	B	B-J	B9.11	5358-5(3)	2	SW-RD-2-B6-W1	12	Schd. 80	Pipe-elbow	
B31	B	B-J	B9.11	5358-5(3)	3	SW-RD-2-B6-W2	12	Schd. 80	Elbow-pipe	
B31	B	B-J	B9.11	5358-5(2)	2	SW-RD-2-B7-W1	12	Schd. 80	Pipe-elbow	
B31	B	B-J	B9.11	5358-5(2)	3	SW-RD-2-B7-W2	12	Schd. 80	Elbow-pipe	
B31	B	B-J	B9.11	5358-5(6)	1	SW-RD-2-B8-W1	12	Schd. 80	Pipe-elbow	B31-Reactor Recirc - Loop B Circ Weld
B31	B	B-J	B9.11	5358-5(6)	2	SW-RD-2-B8-W2	12	Schd. 80	Elbow-safe-end	B31-Reactor Recirc - Loop B Circ Weld
B31	B	B-J	B9.31	5358-5(1)	2	SW-RD-2-B3-W1	22	Schd. 80	Pipe-sweepolet	
B31	B	B-J	B9.31	5358-5(1)	3	SW-RD-2-B3-W2	22	Schd. 80	Pipe-sweepolet	
B31	B	B-J	B9.31	5358-5(1)	6	SW-RD-2-B3-W3	22	Schd. 80	Pipe-sweepolet	
B31	B	B-J	B9.31	5358-5(1)	7	SW-RD-2-B3-W4	22	Schd. 80	Pipe-sweepolet	B31-Reactor Recirc - Loop B Circ Weld
B31	B	B-J	B9.11	5358-5(1)	4	SW-RD-2-B3-W6	22	Schd. 80	Pipe-cross	B31-Reactor Recirc - Loop B Circ Weld
B31	B	B-J	B9.11	5358-5(1)	5	SW-RD-2-B3-W7	22	Schd. 80	Pipe-cross	B31-Reactor Recirc - Loop B Circ Weld

Table D.B.1 BWR-1 – RR System Loop B Weld List

System ID	Loop	Exam Category	Category Item	Line Number	Weld Order	Component ID	NPS (in)	Wall Thk (in) or Schedule	Configuration	Description
B31	B	B-J	B9.11	5358-5(1)	1	SW-RD-2-B3-W8	22	Schd. 80	Pipe-end-cap	B31-Reactor Recirc - Loop B Circ Weld
B31	B	B-J	B9.11	5358-5(1)	8	SW-RD-2-B3-W9	22	Schd. 80	Pipe-end-cap	
B31	B	B-F	B5.10	5359-5(S)	1	4-303B	28	Schd. 80	Nozzle-safe-end	RRS Nozzle-to-Safe End Butt Welds (N1B)
B31	B	B-J	B9.11	5359-5(D)-5358-5(6)	3	FW-RD-2-B1-W1	28	Schd. 80	Pipe-pipe	B31-Reactor Recirc - Loop B Circ Weld
B31	B	B-J	B9.11	5359-5(D)-5358-5(6)	8	FW-RD-2-B2-W2	28	Schd. 80	Pipe-sweepolet	B31-Reactor Recirc - Loop B Circ Weld - 28 x 4 inch SWOL
B31	B	B-J	B9.11	5359-5(D)-5358-5(6)	1	FW-RD-2-B6	28	Schd. 80	Pipe-pump	RR Pump discharge
B31	B	B-J	B9.11	5359-5(D)-5358-5(6)	4	FW-RD-2-B7	28	Schd. 80	Pipe-valve	B31-Reactor Recirc - Loop B Circ Weld
B31	B	B-J	B9.11	5359-5(D)-5358-5(6)	5	FW-RD-2-B8	28	Schd. 80	Pipe-valve	
B31	B	B-J	B9.11	5359-5(D)-5358-5(6)	11	FW-RD-2-B9	28	Schd. 80	Cross-tee	
B31	B	B-J	B9.11	5359-5(S)	2	FW-RS-2-B1	28	Schd. 80	Pipe-safe-end	RPV Nozzle area
B31	B	B-J	B9.11	5359-5(S)	5	FW-RS-2-B2	28	Schd. 80	Pipe-pipe	B31-Reactor Recirc - Loop B Circ Weld
B31	B	B-J	B9.11	5359-5(S)	9	FW-RS-2-B3	28	Schd. 80	Pipe-valve	
B31	B	B-J	B9.11	5359-5(S)	10	FW-RS-2-B4	28	Schd. 80	Pipe-valve	B31-Reactor Recirc - Loop B Circ Weld
B31	B	B-J	B9.11	5359-5(S)	14	FW-RS-2-B5	28	Schd. 80	Pipe-pump	RR Pump suction
B31	B	B-J	B9.31	5359-5(D)-5358-5(6)	2	SW-RD-2-B1-W1	28	Schd. 80	Pipe-pipe	RPV Nozzle area
B31	B	B-J	B9.11	5359-5(D)-5358-5(6)	6	SW-RD-2-B2-W1	28	Schd. 80	Elbow-pipe	
B31	B	B-J	B9.11	5359-5(D)-5358-5(6)	10	SW-RD-2-B2-W2	28	Schd. 80	Pipe-tee	B31-Reactor Recirc - Loop B Circ Weld
B31	B	B-J	B9.11	5359-5(D)-5358-5(6)	9	SW-RD-2-B2-W2O	28	Schd. 80	Pipe-pipe	B31-Reactor Recirc - Loop B Circ Weld
B31	B	B-J	B9.31	5359-5(D)-5358-5(6)	7	SW-RD-2-B2-W3	28	Schd. 80	Pipe-pipe	
B31	B	B-J	B9.11	5359-5(D)-5358-5(6)	99	SW-RD-2-B3-W5	28	Schd. 80	Cross-reducer	B31-Reactor Recirc - Loop B Circ Weld
B31	B	B-J	B9.11	5359-5(S)	3	SW-RS-2-B1-W1	28	Schd. 80	Elbow-pipe	B31-Reactor Recirc - Loop B Circ Weld
B31	B	B-J	B9.11	5359-5(S)	4	SW-RS-2-B1-W2	28	Schd. 80	Elbow-pipe	
B31	B	B-J	B9.11	5359-5(S)	8	SW-RS-2-B2-W1	28	Schd. 80	Elbow-pipe	
B31	B	B-J	B9.11	5359-5(S)	7	SW-RS-2-B2-W10A	28	Schd. 80	Pipe-tee	B31-Reactor Recirc - Loop B Circ Weld
B31	B	B-J	B9.11	5359-5(S)	6	SW-RS-2-B2-W2	28	Schd. 80	Pipe-tee	

Table D.B.1 BWR-1 – RR System Loop B Weld List

System ID	Loop	Exam Category	Category Item	Line Number	Weld Order	Component ID	NPS (in)	Wall Thk (in) or Schedule	Configuration	Description
B31	B	B-J	B9.11	5359-5(S)	13	SW-RS-2-B3-W1	28	Schd. 80	Elbow-pipe	B31-Reactor Recirc - Loop B Circ Weld
B31	B	B-J	B9.31	5359-5(S)	12	SW-RS-2-B3-W2	28	Schd. 80	Pipe-pipe	B31-Reactor Recirc - Loop B Circ Weld
B31	B	B-J	B9.31	5359-5(S)	11	SW-RS-2-B3-W3	28	Schd. 80	Pipe-pipe	
B31	B	B-J	B9.11	5359-5(S)	11.1	SW-RS-2-B3-W4	28	Schd. 80	Pipe-pipe	
B31	B	B-J	B9.11	5359-5(S)	11.2	SW-RS-2-B3-W5	28	Schd. 80	Pipe-pipe	

Table D.B.2 BWR-2 – FW System Loop B Weld List

System ID	Loop	Exam Category	Category Item	Line Number	Weld Order	Component ID	NPS (in)	Wall Thk (in) or Schedule	Configuration	Description
N21	B	B-J	B9.11	3537-5(4)	2	SW-N21-2336-19WF	12	Schd. 100	Elbow-pipe	
N21	B	B-J	B9.11	3537-5(4)	3	SW-N21-2336-19WG	12	Schd. 100	Elbow-pipe	
N21	B	B-J	B9.11	3537-5(4)	4	FW-N21-2336-19W20	12	Schd. 100	Elbow-pipe	
N21	B	B-J	B9.11	3537-5(4)	7	FW-N21-2336-20WF4	12	Schd. 100	Elbow-pipe	
N21	B	B-J	B9.11	3537-5(4)	8	SW-N21-2336-20WM	12	Schd. 100	Elbow-pipe	
N21	B	B-J	B9.11	3537-5(5)	2	SW-N21-2336-17WB	12	Schd. 100	Elbow-pipe	
N21	B	B-J	B9.11	3537-5(5)	3	SW-N21-2336-17WD	12	Schd. 100	Elbow-pipe	
N21	B	B-J	B9.11	3537-5(5)	4	FW-N21-2336-17W18	12	Schd. 100	Elbow-pipe	
N21	B	B-J	B9.11	3537-5(5)	6	SW-N21-2336-18WP	12	Schd. 100	Elbow-pipe	
N21	B	B-J	B9.11	3537-5(5)	7	SW-N21-2336-18WQ	12	Schd. 100	Elbow-pipe	
N21	B	B-J	B9.11	3537-5(5)	8	FW-N21-2336-18W0	12	Schd. 100	Elbow-pipe	Transition piece
N21	B	B-J	B9.11	3537-5(6)	2	SW-N21-2336-14WB	12	Schd. 100	Elbow-pipe	
N21	B	B-J	B9.11	3537-5(6)	6	SW-N21-2336-15WP	12	Schd. 100	Elbow-pipe	N21-2336-Feedwater Loop A Circ Weld
N21	B	B-J	B9.11	3537-5(6)	7	FW-N21-2336-15WF2	12	Schd. 100	Elbow-pipe	
N21	B	B-J	B9.11	3537-5(5)	1	FW-N21-2336-16W17	12	Schd. 100	Elbow-reducing-tee	
N21	B	B-J	B9.11	3537-5(6)	1	FW-N21-2336-13W14	12	Schd. 100	Elbow-reducing-tee	N21-2336-Feedwater Loop A Circ Weld
N21	B	B-J	B9.11	3537-5(4)	11	3-316C	12	Schd. 100	Nozzle-safe-end	N21-2336-Feedwater Loop A Circ Weld
N21	B	B-J	B9.11	3537-5(5)	10	3-316B	12	Schd. 100	Nozzle-safe-end	N21-2336-Feedwater Loop A Circ Weld
N21	B	B-J	B9.11	3537-5(6)	10	3-316A	12	Schd. 100	Nozzle-safe-end	N21-2336-Feedwater Loop A Circ Weld
N21	B	B-J	B9.11	3537-5(4)	5	FW-N21-2336-20WF2	12	Schd. 100	Pipe-pipe	
N21	B	B-J	B9.11	3537-5(4)	6	FW-N21-2336-20WF3	12	Schd. 100	Pipe-pipe	
N21	B	B-J	B9.11	3537-5(5)	5	FW-N21-2336-18WF1	12	Schd. 100	Pipe-pipe	
N21	B	B-J	B9.11	3537-5(6)	3	FW-N21-2336-14WF1	12	Schd. 100	Pipe-pipe	N21-2336-Feedwater Loop A Circ Weld
N21	B	B-J	B9.11	3537-5(6)	4	FW-N21-2336-14W15	12	Schd. 100	Pipe-pipe	N21-2336-Feedwater Loop A Circ Weld
N21	B	B-J	B9.11	3537-5(6)	5	FW-N21-2336-15WF1	12	Schd. 100	Pipe-pipe	
N21	B	B-J	B9.11	3537-5(4)	1	FW-N21-2336-16W19	12	Schd. 100	Pipe-reducer	N21-2336-Feedwater Loop A Circ Weld
N21	B	B-J	B9.11	3537-5(4)	9	FW-N21-2336-20W0	12	Schd. 100	Pipe-safe-end	

Table D.B.2 BWR-2 – FW System Loop B Weld List

System ID	Loop	Exam Category	Category Item	Line Number	Weld Order	Component ID	NPS (in)	Wall Thk (in) or Schedule	Configuration	Description
N21	B	B-J	B9.11	3537-5(6)	8	FW-N21-2336-15W0	12	Schd. 100	Pipe-safe-end	N21-2336-Feedwater Loop A Circ Weld
N21	B	B-J	B9.11	3537-5(4)	10	N4-C	12	Schd. 100	Safe-end-Safe-end	
N21	B	B-J	B9.11	3537-5(5)	9	N4-B	12	Schd. 100	Safe-end-Safe-end	
N21	B	B-J	B9.11	3537-5(6)	9	N4A	12	Schd. 100	Safe-end-Safe-end	N21-2336-Feedwater Loop A Circ Weld
N21	B	B-J	B9.11	3537-5(2)	3	SW-N21-2336-11WD	20	Schd. 100	Elbow-pipe	
N21	B	B-J	B9.11	3537-5(3)	8	FW-N21-2336-2WB	20	Schd. 100	Elbow-pipe	
N21	B	B-J	B9.11	3537-5(2)	4	SW-N21-2336-11WE	20	Schd. 100	Elbow-pipe	
N21	B	B-J	B9.11	3537-5(3)	12	SW-N21-2336-13WC	20	Schd. 100	Elbow-reducing-tee	N21-2336-Feedwater Loop A Circ Weld
N21	B	B-J	B9.11	3537-5(3)	7	FW-N21-2336-0W02	20	Schd. 100	Elbow-valve	
N21	B	B-J	B9.11	3537-5(3)	11	SW-N21-2336-13WB	20	Schd. 100	Elbow-valve	
N21	B	B-J	B9.11	3537-5(3)	5	SW-X9A-W1	20	Schd. 100	Penetration	Longitudinal weld
N21	B	B-J	B9.11	3537-5(3)	4	FW-N21-2336-11W01	20	Schd. 100	Pipe-penetration	
N21	B	B-J	B9.11	3537-5(2)	2	FW-N21-2336-11WF1	20	Schd. 100	Pipe-pipe	
N21	B	B-J	B9.11	3537-5(2)	5	FW-N21-2336-11WF2	20	Schd. 100	Pipe-pipe	
N21	B	B-J	B9.11	3537-5(2)	6	FW-N21-2336-11WF3	20	Schd. 100	Pipe-pipe	
N21	B	B-J	B9.11	3537-5(3)	14	FW-N21-2336-3AW13	20	Schd. 100	Pipe-pipe	
N21	B	B-J	B9.11	3537-5(3)	15	FW-N21-2336-3W03A	20	Schd. 100	Pipe-pipe	
N21	B	B-J	B9.11	3537-5(3)	16	FW-N21-2336-3W16	20	Schd. 100	Pipe-pipe	
N21	B	B-J	B9.11	3537-5(2)	1	SW-N21-2336-11WB	20	Schd. 100	Pipe-reducer	
N21	B	B-J	B9.11	3537-5(3)	13	SW-N21-2336-13WE	20	Schd. 100	Pipe-reducing-tee	N21-2336-Feedwater Loop A Circ Weld
N21	B	B-J	B9.11	3537-5(3)	17	SW-N21-2336-16WC	20	Schd. 100	Pipe-reducing-tee	
N21	B	B-J	B9.11	3537-5(2)	7	SW-N21-2336-11WN	20	Schd. 100	Pipe-tee	
N21	B	B-J	B9.11	3537-5(3)	3	SW-N21-2336-11WH	20	Schd. 100	Pipe-tee	
N21	B	B-J	B9.11	3537-5(3)	1	FW-N21-2336-1VW11	20	Schd. 100	Pipe-valve	
N21	B	B-J	B9.11	3537-5(3)	6	FW-N21-2336-1VW12	20	Schd. 100	Pipe-valve	
N21	B	B-J	B9.11	3537-5(3)	9	FW-N21-2336-2W0	20	Schd. 100	Pipe-valve	N21-2336-Feedwater Loop A Circ Weld
N21	B	B-J	B9.11	3537-5(3)	10	FW-N21-2336-0W13	20	Schd. 100	Pipe-valve	N21-2336-Feedwater Loop A Circ Weld
N21	B	B-J	B9.11	3537-5(3)	18	SW-N21-2336-16WE	20	Schd. 100	Reducer-reducing-tee	
N21	B	B-J	B9.11	3537-5(3)	2	FW-N21-2336-0W11	20	Schd. 100	Tee-valve	

Table D.B.3 PWR-1 – Reactor Coolant System Hot Leg

Examination Category	Category Item	Component ID	Configuration	NPS (in)	Wall Thk (in)	Weld Material
B-J	B9.11	1-4100A- 8	ELBOW TO PIPE	31.00	2.600	304N/351CF
B-J	B9.11	1-4100A- 9	ELBOW TO PIPE	31.00	2.625	SS
B-J	B9.11	1-4100A- 10	ELBOW TO PIPE	31.00	2.625	SS
B-J	B9.11	1-4100A- 11	PIPE TO ELBOW	31.00	2.625	SS
B-J	B9.11	1-4100A- 12	ELBOW TO PUMP	31.00	2.625	SS
B-J	B9.11	1-4100A- 13	PIPE TO PUMP	27.50	2.375	304N
B-J	B9.11	1-4100A- 14	PIPE TO ELBOW	27.50	2.375	SS
B-J	B9.11	1-4100A- 15	BIMETAL (INCONEL) WELD.ELBOW TO SAFE END	27.50	2.375	SS/INCONEL
B-F	B5.10	1-4100A- 16(DM)	BIMETAL (INCONEL) WELD. R.V. LOOP A INLET NOZZLE TO SAFE END	27.50	2.375	CS/SS

Table D.B.4 PWR-2 – Pressurizer Surge Line

ASME XI Examination Category	Component ID	Description / Configuration	NPS [inch]	Wall Thk [inch]	Weld Material
B-J	1-4100A- 19BC	14" Branch (CGE-1-4500) Connection to 29-inch pipe	14.00	2.350	304N/SA182
B-F	1-4500A- 1(DM)	Bimetal (INCONEL) weld; Pressurizer Surge Line Nozzle to Safe End	14.00	1.406	SS
B-J	1-4500A- 2	Bimetal (INCONEL) Weld: Safe End to Pipe	14.00	1.406	SS
B-J	1-4500A- 3	Pipe-to-Elbow	14.00	1.406	SS
B-J	1-4500A- 4	Pipe-to-Elbow	14.00	1.406	SS
B-J	1-4500A- 5	Pipe-to-Elbow	14.00	1.406	SS
B-J	1-4500A- 6	Pipe-to-Elbow	14.00	1.406	SS
B-J	1-4500A- 7	Pipe-to-Elbow	14.00	1.406	SS
B-J	1-4500A- 8	Pipe-to-Elbow	14.00	1.406	SS
B-J	1-4500A- 9	Pipe-to-Elbow	14.00	1.406	SS
B-J	1-4500A- 10	Pipe-to-Elbow	14.00	1.406	SS
B-J	1-4500A- 11	Pipe-to-Elbow	14.00	1.406	SS
B-J	1-4500A- 12	Pipe-to-Elbow	14.00	1.406	SS
B-J	1-4500A- 13	Pipe-to-Branch Connection	14.00	1.406	SS

Table D.B.5 PWR-5 – HPI/NMU Line

Weld ID	Configuration	Inside Diameter
17-MU-23-73AR	Elbow-valve	2.50
17-MU-23-21-055	Elbow-pipe	2.50
17-MU-23-21-057	Elbow-pipe	2.50
17-MU-23-21-058	Elbow-pipe	2.50
17-MU-23-21-059	Elbow-valve	2.50
17-MU-23-21-059B	Elbow-valve	2.50
17-MU-23-21-061	Pipe-pipe	2.50
17-MU-23-21-062	Elbow-pipe	2.50
17-MU-23-21-063	Elbow-nozzle	2.50

Table D.B.6 BWR-1 Weld Failure Rate Calculation Sheet

Description	Mean (Prior)[25]	Range Factor	Failures (Evidence)	Exposure [Weld-Yr]	Weld Count[26]	Mean	5th	50th	95th	Range Factor	Calc Date
NPS28 E-P Low	7.19E-04	10	0	4,725	5	1.28E-04	1.41E-05	8.78E-05	3.70E-04	5.1	5/13/2003 15:16
NPS28 E-P Medium	7.19E-04	10	0	9,450	10	8.37E-05	1.10E-05	6.09E-05	2.29E-04	4.6	5/13/2003 15:16
NPS28 E-P High	7.19E-04	10	0	14,175	15	6.42E-05	9.28E-06	4.81E-05	1.70E-04	4.3	5/13/2003 15:16
NPS28 P-V Low	2.25E-04	10	0	3,780	4	8.20E-05	6.29E-06	4.84E-05	2.66E-04	6.5	5/13/2003 15:16
NPS28 P-V Medium	2.25E-04	10	0	7,560	8	5.86E-05	5.41E-06	3.75E-05	1.80E-04	5.8	5/13/2003 15:16
NPS28 P-V High	2.25E-04	10	0	11,340	12	4.71E-05	4.84E-06	3.15E-05	1.40E-04	5.4	5/13/2003 15:16
NPS28 N-Se Low	1.50E-04	10	0	1,890	2	8.34E-05	4.89E-06	4.27E-05	2.93E-04	7.7	5/13/2003 15:16
NPS28 N-Se Medium	1.50E-04	10	0	3,780	4	6.49E-05	4.49E-06	3.64E-05	2.18E-04	7.0	5/13/2003 15:16
NPS28 N-Se High	1.50E-04	10	0	5,670	6	5.46E-05	4.20E-06	3.23E-05	1.78E-04	6.5	5/13/2003 15:16
NPS28 P-Se Low	6.74E-04	10	0	1,890	2	2.03E-04	1.74E-05	1.26E-04	6.39E-04	6.1	5/13/2003 15:16
NPS28 P-Se Medium	6.74E-04	10	0	3,780	4	1.41E-04	1.45E-05	9.43E-05	4.19E-04	5.4	5/13/2003 15:16
NPS28 P-Se High	6.74E-04	10	0	5,670	6	1.12E-04	1.28E-05	7.76E-05	3.20E-04	5.0	5/13/2003 15:16
NPS28 P-P Low	4.99E-05	20	0	8,505	9	1.75E-05	3.99E-07	6.56E-06	7.03E-05	13.3	5/13/2003 15:16
NPS28 P-P Medium	4.99E-05	20	0	17,010	18	1.27E-05	3.64E-07	5.45E-06	4.91E-05	11.6	5/13/2003 15:16
NPS28 P-P High	4.99E-05	20	0	25,515	27	1.04E-05	3.39E-07	4.77E-06	3.88E-05	10.7	5/13/2003 15:16
NPS28 P-Pu Low	2.25E-04	10	0	1,890	2	1.09E-04	7.01E-06	5.88E-05	3.73E-04	7.3	5/13/2003 15:16
NPS28 P-Pu Medium	2.25E-04	10	0	3,780	4	8.20E-05	6.29E-06	4.84E-05	2.66E-04	6.5	5/13/2003 15:16
NPS28 P-Pu High	2.25E-04	10	0	5,670	6	6.78E-05	5.79E-06	4.20E-05	2.13E-04	6.1	5/13/2003 15:16
NPS28 P-T Low	1.25E-04	10	0	1,890	2	7.36E-05	4.15E-06	3.68E-05	2.61E-04	7.9	5/13/2003 15:16
NPS28 P-T Medium	1.25E-04	10	0	3,780	4	5.81E-05	3.84E-06	3.18E-05	1.98E-04	7.2	5/13/2003 15:16
NPS28 P-T High	1.25E-04	10	0	5,670	6	4.93E-05	3.61E-06	2.85E-05	1.63E-04	6.7	5/13/2003 15:16
NPS28 P-X Low	7.49E-05	20	0	945	1	4.63E-05	6.74E-07	1.27E-05	1.95E-04	17.0	5/13/2003 15:16
NPS28 P-X Medium	7.49E-05	20	0	1,890	2	3.79E-05	6.51E-07	1.17E-05	1.58E-04	15.6	5/13/2003 15:16
NPS28 P-X High	7.49E-05	20	0	2,835	3	3.29E-05	6.33E-07	1.10E-05	1.36E-04	14.6	5/13/2003 15:16
NPS28 R-X Low	7.49E-05	20	0	945	1	4.63E-05	6.74E-07	1.27E-05	1.95E-04	17.0	5/13/2003 15:16
NPS28 R-X Medium	7.49E-05	20	0	1,890	2	3.79E-05	6.51E-07	1.17E-05	1.58E-04	15.6	5/13/2003 15:16
NPS28 R-X High	7.49E-05	20	0	2,835	3	3.29E-05	6.33E-07	1.10E-05	1.36E-04	14.6	5/13/2003 15:16

[25] From Table 11 (RR NPS28).
[26] Weld count (medium value) is taken from PlantBWelds.xls (see Section 1.3); Table B-1 is an excerpt from this Excel-file.

Table D.B.7 BWR-2 Weld Failure Rate Calculation Sheet

Description	Mean (Prior)	Range Factor	Failures (Evidence)	Exposure (Weld-Yr)	Welds	Mean	5th	50th	95th	Range Factor	Calc Date
NPS12 N-Se Low	1.26E-04	10	0	2,835	3	6.47E-05	4.00E-06	3.42E-05	2.24E-04	7.5	5/13/2003 15:05
NPS12 N-Se Medium	1.26E-04	10	0	5,670	6	4.95E-05	3.63E-06	2.86E-05	1.63E-04	6.7	5/13/2003 15:05
NPS12 N-Se High	1.26E-04	10	0	8,505	9	4.12E-05	3.36E-06	2.50E-05	1.32E-04	6.3	5/13/2003 15:05
NPS12 E-P Low	6.73E-06	20	0	13,230	14	3.92E-06	6.00E-08	1.11E-06	1.64E-05	16.6	5/13/2003 15:05
NPS12 E-P Medium	6.73E-06	20	0	26,460	28	3.15E-06	5.77E-08	1.02E-06	1.31E-05	15.1	5/13/2003 15:05
NPS12 E-P High	6.73E-06	20	0	39,690	42	2.71E-06	5.58E-08	9.51E-07	1.11E-05	14.1	5/13/2003 15:05
NPS12 P-P Low	1.05E-05	15	0	5,670	6	7.48E-06	1.71E-07	2.45E-06	3.01E-05	13.3	5/13/2003 15:05
NPS12 P-P Medium	1.05E-05	15	0	11,340	12	6.33E-06	1.66E-07	2.29E-06	2.53E-05	12.3	5/13/2003 15:05
NPS12 P-P High	1.05E-05	15	0	17,010	18	5.61E-06	1.62E-07	2.17E-06	2.21E-05	11.7	5/13/2003 15:05
NPS12 P-R Low	3.14E-05	15	0	945	1	2.54E-05	5.23E-07	7.64E-06	1.02E-04	13.9	5/13/2003 15:05
NPS12 P-R Medium	3.14E-05	15	0	1,890	2	2.24E-05	5.14E-07	7.34E-06	9.04E-05	13.3	5/13/2003 15:05
NPS12 P-R High	3.14E-05	15	0	2,835	3	2.05E-05	5.06E-07	7.09E-06	8.22E-05	12.7	5/13/2003 15:05
NPS12 P-Se Low	8.80E-06	20	0	4,725	5	6.16E-06	8.05E-08	1.55E-06	2.57E-05	17.9	5/13/2003 15:05
NPS12 P-Se Medium	8.80E-06	20	0	9,450	10	5.22E-06	7.86E-08	1.47E-06	2.19E-05	16.7	5/13/2003 15:05
NPS12 P-Se High	8.80E-06	20	0	14,175	15	4.63E-06	7.71E-08	1.40E-06	1.94E-05	15.9	5/13/2003 15:05
NPS12 E-Rt Low	9.43E-06	20	0	1,890	2	7.63E-06	8.76E-08	1.72E-06	3.12E-05	18.9	5/13/2003 15:05
NPS12 E-Rt Medium	9.43E-06	20	0	3,780	4	6.80E-06	8.65E-08	1.67E-06	2.83E-05	18.1	5/13/2003 15:05
NPS12 E-Rt High	9.43E-06	20	0	5,670	6	6.24E-06	8.56E-08	1.63E-06	2.62E-05	17.5	5/13/2003 15:05

Table D.B.8 PWR-1 Weld Failure Rate Calculation Sheet

Description	Mean (Prior)[27]	Range Factor	Failures (Evidence)	Exposure [Weld-Yr]	Weld Count	Mean	5th	50th	95th	Range Factor	Calc Date
NPS30 E-P Low	3.65E-06	100	0	6,720	20	1.49E-06	7.05E-10	6.85E-08	5.61E-06	89.2	5/2/2003 9:25
NPS30 E-P Medium	3.65E-06	100	0	13,440	40	1.19E-06	6.96E-10	6.65E-08	4.88E-06	83.7	5/2/2003 9:25
NPS30 E-P High	3.65E-06	100	0	20,160	60	1.02E-06	6.89E-10	6.49E-08	4.40E-06	79.9	5/2/2003 9:25
NPS30 P-Pu Low	4.06E-05	100	0	504	2	1.75E-05	7.86E-09	7.67E-07	6.42E-05	90.4	5/2/2003 9:25
NPS30 P-Pu Medium	4.06E-05	100	0	1,008	3	1.40E-05	7.77E-09	7.47E-07	5.66E-05	85.3	5/2/2003 9:25
NPS30 P-Pu High	4.06E-05	100	0	1,512	5	1.22E-05	7.71E-09	7.30E-07	5.14E-05	81.7	5/2/2003 9:25
NPS30 N-Se Low	8.12E-04	100	0	504	2	1.06E-04	1.42E-07	1.19E-05	5.07E-04	59.8	5/2/2003 9:25
NPS30 N-Se Medium	8.12E-04	100	0	1,008	3	7.35E-05	1.33E-07	1.04E-05	3.53E-04	51.4	5/2/2003 9:25
NPS30 N-Se High	8.12E-04	100	0	1,512	5	5.85E-05	1.28E-07	9.41E-06	2.80E-04	46.8	5/2/2003 9:25
NPS30 E-Se Low	8.12E-05	100	0	504	2	2.81E-05	1.55E-08	1.49E-06	1.13E-04	85.3	5/2/2003 9:25
NPS30 E-Se Medium	8.12E-05	100	0	1,008	3	2.18E-05	1.53E-08	1.43E-06	9.49E-05	78.8	5/2/2003 9:25
NPS30 E-Se High	8.12E-05	100	0	1,512	5	1.85E-05	1.51E-08	1.39E-06	8.34E-05	74.3	5/2/2003 9:25
NPS30 E-Pu Low	5.07E-05	100	0	504	2	2.04E-05	9.78E-09	9.51E-07	7.74E-05	89.0	5/2/2003 9:25
NPS30 E-Pu Medium	5.07E-05	100	0	1,008	3	1.62E-05	9.66E-09	9.22E-07	6.72E-05	83.4	5/2/2003 9:25
NPS30 E-Pu High	5.07E-05	100	0	1,512	5	1.40E-05	9.57E-09	8.99E-07	6.04E-05	79.4	5/2/2003 9:25

[27] From Table 12 (RC Hot Leg).

Table D.B.9 PWR-2 Weld Failure Rate Calculation Sheet

Description	Mean (Prior)	Range Factor	Failures (Evidence)	Exposure	Welds Count	Mean	5th	50th	95th	Range Factor	Calc Date
NPS14 E-P Low	6.70E-07	100	0	18,105	5	3.33E-07	1.30E-10	1.28E-08	1.13E-06	93.3	5/2/2003 10:14
NPS14 E-P Medium	6.70E-07	100	0	36,210	10	2.74E-07	1.29E-10	1.26E-08	1.03E-06	89.3	5/2/2003 10:14
NPS14 E-P High	6.70E-07	100	0	54,315	15	2.41E-07	1.29E-10	1.24E-08	9.57E-07	86.3	5/2/2003 10:14
NPS14 N-Se Low	3.75E-05	100	0	1,811	1	1.08E-05	7.10E-09	6.70E-07	4.62E-05	80.7	5/2/2003 10:14
NPS14 N-Se Medium	3.75E-05	100	0	3,621	1	8.18E-06	6.94E-09	6.34E-07	3.72E-05	73.2	5/2/2003 10:14
NPS14 N-Se High	3.75E-05	100	0	5,432	2	6.84E-06	6.81E-09	6.06E-07	3.19E-05	68.4	5/2/2003 10:14
NPS14 P-Se Low	3.35E-06	100	0	1,811	1	1.96E-06	6.55E-10	6.48E-08	6.03E-06	96.0	5/2/2003 10:14
NPS14 P-Se Medium	3.35E-06	100	0	3,621	1	1.67E-06	6.52E-10	6.41E-08	5.67E-06	93.3	5/2/2003 10:14
NPS14 P-Se High	3.35E-06	100	0	5,432	2	1.49E-06	6.49E-10	6.35E-08	5.39E-06	91.1	5/2/2003 10:14
NPS14 B-HL Low	4.69E-05	100	0	1,811	1	1.24E-05	8.82E-09	8.25E-07	5.42E-05	78.4	5/2/2003 10:14
NPS14 B-HL Medium	4.69E-05	100	0	3,621	1	9.28E-06	8.60E-09	7.74E-07	4.28E-05	70.6	5/2/2003 10:14
NPS14 B-HL High	4.69E-05	100	0	5,432	2	7.71E-06	8.42E-09	7.37E-07	3.63E-05	65.7	5/2/2003 10:14
NPS14 B-P Low	3.35E-06	100	0	1,811	1	1.96E-06	6.55E-10	6.48E-08	6.03E-06	96.0	5/2/2003 10:14
NPS14 B-P Medium	3.35E-06	100	0	3,621	1	1.67E-06	6.52E-10	6.41E-08	5.67E-06	93.3	5/2/2003 10:14
NPS14 B-P High	3.35E-06	100	0	5,432	2	1.49E-06	6.49E-10	6.35E-08	5.39E-06	91.1	5/2/2003 10:14

Table D.B.10 PWR-3 Weld Failure Rate Calculation Sheet

Description	Mean (Prior)	Range Factor	Failures (Evidence)	Exposure	Weld Count	Mean	5th	50th	95th	Range Factor	Calc Date
NPS4 E-P Low	1.90E-06	97	0	140	4	1.69E-06	4.05E-10	3.93E-08	3.79E-06	96.7	5/2/2003 14:14
NPS4 E-P Medium	1.90E-06	97	0	280	8	1.61E-06	4.05E-10	3.92E-08	3.77E-06	96.5	5/2/2003 14:14
NPS4 E-P High	1.90E-06	97	0	420	12	1.54E-06	4.05E-10	3.92E-08	3.75E-06	96.3	5/2/2003 14:14
NPS4 E-V Low	1.31E-06	98	0	105	3	1.21E-06	2.69E-10	2.64E-08	2.59E-06	98.2	5/2/2003 14:14
NPS4 E-V Medium	1.31E-06	98	0	210	6	1.16E-06	2.69E-10	2.64E-08	2.59E-06	98.1	5/2/2003 14:14
NPS4 E-V High	1.31E-06	98	0	315	9	1.13E-06	2.69E-10	2.64E-08	2.58E-06	98.0	5/2/2003 14:14
NPS4 P-P Low	6.48E-06	100	0	35	1	5.81E-06	1.28E-09	1.27E-07	1.27E-05	99.6	5/2/2003 14:14
NPS4 P-P Medium	6.48E-06	100	0	70	2	5.52E-06	1.28E-09	1.27E-07	1.26E-05	99.4	5/2/2003 14:14
NPS4 P-P High	6.48E-06	100	0	105	3	5.32E-06	1.28E-09	1.27E-07	1.26E-05	99.2	5/2/2003 14:14
NPS4 E-N Low	9.86E-04	16	0	35	1	7.59E-04	1.37E-05	2.17E-04	3.09E-03	15.0	5/2/2003 14:15
NPS4 E-N Medium	9.86E-04	16	0	70	2	6.61E-04	1.34E-05	2.07E-04	2.70E-03	14.2	5/2/2003 14:15
NPS4 E-N High	9.86E-04	16	0	105	3	5.97E-04	1.32E-05	2.00E-04	2.43E-03	13.6	5/2/2003 14:15

ATTACHMENT C TO APPENDIX D

BASIS FOR LOCA FREQUENCY MODELS

Attached as Table D.C.1 is an Excel spreadsheet used to calculate time-dependent LOCA frequencies. Table D.C.1 includes the parameters input to the BWR-1 Cat1 LOCA frequency calculation.

Table D.C.1 Application of Markov Model to BWR-1 Cat 1 LOCA Frequency

Config. (Count)	INSPECTION CASE			INSPECTION INDEPENDENT PARAMETERS									INSPECTION DEPENDENT PARAMETERS			
	CASE	ISI Inspection Coverage	Leak Inspection Interval	P_{LI}	TSUBLI	MU	PHI	RHO F	RHO L	Lambda	T_{FI}	TSUBR	P_{FI}	P_{FD}	OMEGA	Haz{T}
Cross-to-reducer (2)	1	None	None	0.00	1.50	0.0000	3.34E-03	5.96E-07	2.00E-02	4.40E-04	10.00	0.023	0.000	0.000	0.000	7.62E-06
	2	None	RF	0.90	1.50	0.5910	3.34E-03	5.96E-07	2.00E-02	4.40E-04	10.00	0.023	0.000	0.000	0.000	1.12E-06
	3	None	Wk	0.90	1.92E-02	21.3971	3.34E-03	5.96E-07	2.00E-02	4.40E-04	10.00	0.023	0.000	0.000	0.000	8.01E-08
	4	None	Shift	0.90	9.13E-04	37.9048	3.34E-03	5.96E-07	2.00E-02	4.40E-04	10.00	0.023	0.000	0.000	0.000	6.59E-08
	5	Secondary	None	0.00	1.50	0.0000	3.34E-03	5.96E-07	2.00E-02	4.40E-04	10.00	0.023	1.000	0.500	0.050	5.18E-06
	6	Secondary	RF	0.90	1.50	0.5910	3.34E-03	5.96E-07	2.00E-02	4.40E-04	10.00	0.023	1.000	0.500	0.050	6.66E-07
	7	Secondary	Wk	0.90	1.92E-02	21.3971	3.34E-03	5.96E-07	2.00E-02	4.40E-04	10.00	0.023	1.000	0.500	0.050	4.62E-08
	8	Secondary	Shift	0.90	9.13E-04	37.9048	3.34E-03	5.96E-07	2.00E-02	4.40E-04	10.00	0.023	1.000	0.500	0.050	3.80E-08
	9	Primary	None	0.00	1.50	0.0000	3.34E-03	5.96E-07	2.00E-02	4.40E-04	10.00	0.023	1.000	0.900	0.090	4.02E-06
	10	Primary	RF	0.90	1.50	0.5910	3.34E-03	5.96E-07	2.00E-02	4.40E-04	10.00	0.023	1.000	0.900	0.090	4.74E-07
	11	Primary	Wk	0.90	1.92E-02	21.3971	3.34E-03	5.96E-07	2.00E-02	4.40E-04	10.00	0.023	1.000	0.900	0.090	3.24E-08
	12	Primary	Shift	0.90	9.13E-04	37.9048	3.34E-03	5.96E-07	2.00E-02	4.40E-04	10.00	0.023	1.000	0.900	0.090	2.67E-08
Cross-to-tee (2)	1	None	None	0.00	1.50	0.0000	3.29E-03	5.88E-07	2.00E-02	4.34E-04	10.00	0.023	0.000	0.000	0.000	7.43E-06
	2	None	RF	0.90	1.50	0.5910	3.29E-03	5.88E-07	2.00E-02	4.34E-04	10.00	0.023	0.000	0.000	0.000	1.09E-06
	3	None	Wk	0.90	1.92E-02	21.3971	3.29E-03	5.88E-07	2.00E-02	4.34E-04	10.00	0.023	0.000	0.000	0.000	7.81E-08
	4	None	Shift	0.90	9.13E-04	37.9048	3.29E-03	5.88E-07	2.00E-02	4.34E-04	10.00	0.023	0.000	0.000	0.000	6.42E-08
	5	Secondary	None	0.00	1.50	0.0000	3.29E-03	5.88E-07	2.00E-02	4.34E-04	10.00	0.023	1.000	0.500	0.050	5.05E-06
	6	Secondary	RF	0.90	1.50	0.5910	3.29E-03	5.88E-07	2.00E-02	4.34E-04	10.00	0.023	1.000	0.500	0.050	6.49E-07
	7	Secondary	Wk	0.90	1.92E-02	21.3971	3.29E-03	5.88E-07	2.00E-02	4.34E-04	10.00	0.023	1.000	0.500	0.050	4.50E-08
	8	Secondary	Shift	0.90	9.13E-04	37.9048	3.29E-03	5.88E-07	2.00E-02	4.34E-04	10.00	0.023	1.000	0.500	0.050	3.71E-08
	9	Primary	None	0.00	1.50	0.0000	3.29E-03	5.88E-07	2.00E-02	4.34E-04	10.00	0.023	1.000	0.900	0.090	3.92E-06
	10	Primary	RF	0.90	1.50	0.5910	3.29E-03	5.88E-07	2.00E-02	4.34E-04	10.00	0.023	1.000	0.900	0.090	4.62E-07
	11	Primary	Wk	0.90	1.92E-02	21.3971	3.29E-03	5.88E-07	2.00E-02	4.34E-04	10.00	0.023	1.000	0.900	0.090	3.16E-08
	12	Primary	Shift	0.90	9.13E-04	37.9048	3.29E-03	5.88E-07	2.00E-02	4.34E-04	10.00	0.023	1.000	0.900	0.090	2.60E-08
Elbow-to-pipe (10)	1	None	None	0.00	1.50	0.0000	7.36E-03	1.32E-06	2.00E-02	9.71E-04	10.00	0.023	0.000	0.000	0.000	3.57E-05
	2	None	RF	0.90	1.50	0.5910	7.36E-03	1.32E-06	2.00E-02	9.71E-04	10.00	0.023	0.000	0.000	0.000	5.19E-06
	3	None	Wk	0.90	1.92E-02	21.3971	7.36E-03	1.32E-06	2.00E-02	9.71E-04	10.00	0.023	0.000	0.000	0.000	3.69E-07
	4	None	Shift	0.90	9.13E-04	37.9048	7.36E-03	1.32E-06	2.00E-02	9.71E-04	10.00	0.023	0.000	0.000	0.000	3.04E-07
	5	Secondary	None	0.00	1.50	0.0000	7.36E-03	1.32E-06	2.00E-02	9.71E-04	10.00	0.023	1.000	0.500	0.050	2.45E-05
	6	Secondary	RF	0.90	1.50	0.5910	7.36E-03	1.32E-06	2.00E-02	9.71E-04	10.00	0.023	1.000	0.500	0.050	3.11E-06
	7	Secondary	Wk	0.90	1.92E-02	21.3971	7.36E-03	1.32E-06	2.00E-02	9.71E-04	10.00	0.023	1.000	0.500	0.050	2.15E-07
	8	Secondary	Shift	0.90	9.13E-04	37.9048	7.36E-03	1.32E-06	2.00E-02	9.71E-04	10.00	0.023	1.000	0.500	0.050	1.77E-07
	9	Primary	None	0.00	1.50	0.0000	7.36E-03	1.32F-06	2.00F-02	9.71E-04	10.00	0.023	1.000	0.900	0.090	1.91E-05
	10	Primary	RF	0.90	1.50	0.5910	7.36E-03	1.32E-06	2.00E-02	9.71E-04	10.00	0.023	1.000	0.900	0.090	2.23E-06
	11	Primary	Wk	0.90	1.92E-02	21.3971	7.36E-03	1.32E-06	2.00E-02	9.71E-04	10.00	0.023	1.000	0.900	0.090	1.52E-07
	12	Primary	Shift	0.90	9.13E-04	37.9048	7.36E-03	1.32E-06	2.00E-02	9.71E-04	10.00	0.023	1.000	0.900	0.090	1.25E-07
Nozzle-to-	1	None	None	0.00	1.50	0.0000	5.66E-03	1.01E-06	2.00E-02	7.47E-04	10.00	0.023	0.000	0.000	0.000	2.15E-05

Table D.C.1 Application of Markov Model to BWR-1 Cat 1 LOCA Frequency

Config. (Count)	INSPECTION CASE			INSPECTION INDEPENDENT PARAMETERS									INSPECTION DEPENDENT PARAMETERS			
	CASE	ISI Inspection Coverage	Leak Inspection Interval	P_{LI}	TSUBLI	MU	PHI	RHO F	RHO L	Lambda	T_{FI}	TSUBR	P_{FI}	P_{FD}	OMEGA	Haz(T)
safe-end (2)	2	None	RF	0.90	1.50	0.5910	5.66E-03	1.01E-06	2.00E-02	7.47E-04	10.00	0.023	0.000	0.000	0.000	3.13E-06
	3	None	Wk	0.90	1.92E-02	21.3971	5.66E-03	1.01E-06	2.00E-02	7.47E-04	10.00	0.023	0.000	0.000	0.000	2.23E-07
	4	None	Shift	0.90	9.13E-04	37.9048	5.66E-03	1.01E-06	2.00E-02	7.47E-04	10.00	0.023	0.000	0.000	0.000	1.84E-07
	5	Secondary	None	0.00	1.50	0.0000	5.66E-03	1.01E-06	2.00E-02	7.47E-04	10.00	0.023	1.000	0.500	0.050	1.47E-05
	6	Secondary	RF	0.90	1.50	0.5910	5.66E-03	1.01E-06	2.00E-02	7.47E-04	10.00	0.023	1.000	0.500	0.050	1.87E-06
	7	Secondary	Wk	0.90	1.92E-02	21.3971	5.66E-03	1.01E-06	2.00E-02	7.47E-04	10.00	0.023	1.000	0.500	0.050	1.30E-07
	8	Secondary	Shift	0.90	9.13E-04	37.9048	5.66E-03	1.01E-06	2.00E-02	7.47E-04	10.00	0.023	1.000	0.500	0.050	1.07E-07
	9	Primary	None	0.00	1.50	0.0000	5.66E-03	1.01E-06	2.00E-02	7.47E-04	10.00	0.023	1.000	0.900	0.090	1.14E-05
	10	Primary	RF	0.90	1.50	0.5910	5.66E-03	1.01E-06	2.00E-02	7.47E-04	10.00	0.023	1.000	0.900	0.090	1.34E-06
	11	Primary	Wk	0.90	1.92E-02	21.3971	5.66E-03	1.01E-06	2.00E-02	7.47E-04	10.00	0.023	1.000	0.900	0.090	9.15E-08
	12	Primary	Shift	0.90	9.13E-04	37.9048	5.66E-03	1.01E-06	2.00E-02	7.47E-04	10.00	0.023	1.000	0.900	0.090	7.52E-08
Pipe-to-pipe (18)	1	None	None	0.00	1.50	0.0000	1.11E-03	1.99E-07	2.00E-02	1.47E-04	10.00	0.023	0.000	0.000	0.000	8.70E-07
	2	None	RF	0.90	1.50	0.5910	1.11E-03	1.99E-07	2.00E-02	1.47E-04	10.00	0.023	0.000	0.000	0.000	1.29E-07
	3	None	Wk	0.90	1.92E-02	21.3971	1.11E-03	1.99E-07	2.00E-02	1.47E-04	10.00	0.023	0.000	0.000	0.000	9.23E-09
	4	None	Shift	0.90	9.13E-04	37.9048	1.11E-03	1.99E-07	2.00E-02	1.47E-04	10.00	0.023	0.000	0.000	0.000	7.59E-09
	5	Secondary	None	0.00	1.50	0.0000	1.11E-03	1.99E-07	2.00E-02	1.47E-04	10.00	0.023	1.000	0.500	0.050	5.89E-07
	6	Secondary	RF	0.90	1.50	0.5910	1.11E-03	1.99E-07	2.00E-02	1.47E-04	10.00	0.023	1.000	0.500	0.050	7.62E-08
	7	Secondary	Wk	0.90	1.92E-02	21.3971	1.11E-03	1.99E-07	2.00E-02	1.47E-04	10.00	0.023	1.000	0.500	0.050	5.29E-09
	8	Secondary	Shift	0.90	9.13E-04	37.9048	1.11E-03	1.99E-07	2.00E-02	1.47E-04	10.00	0.023	1.000	0.500	0.050	4.35E-09
	9	Primary	None	0.00	1.50	0.0000	1.11E-03	1.99E-07	2.00E-02	1.47E-04	10.00	0.023	1.000	0.900	0.090	4.55E-07
	10	Primary	RF	0.90	1.50	0.5910	1.11E-03	1.99E-07	2.00E-02	1.47E-04	10.00	0.023	1.000	0.900	0.090	5.40E-08
	11	Primary	Wk	0.90	1.92E-02	21.3971	1.11E-03	1.99E-07	2.00E-02	1.47E-04	10.00	0.023	1.000	0.900	0.090	3.70E-09
	12	Primary	Shift	0.90	9.13E-04	37.9048	1.11E-03	1.99E-07	2.00E-02	1.47E-04	10.00	0.023	1.000	0.900	0.090	3.04E-09
Pipe-to-pump (4)	1	None	None	0.00	1.50	0.0000	7.23E-03	1.29E-06	2.00E-02	9.53E-04	10.00	0.023	0.000	0.000	0.000	3.44E-05
	2	None	RF	0.90	1.50	0.5910	7.23E-03	1.29E-06	2.00E-02	9.53E-04	10.00	0.023	0.000	0.000	0.000	5.00E-06
	3	None	Wk	0.90	1.92E-02	21.3971	7.23E-03	1.29E-06	2.00E-02	9.53E-04	10.00	0.023	0.000	0.000	0.000	3.56E-07
	4	None	Shift	0.90	9.13E-04	37.9048	7.23E-03	1.29E-06	2.00E-02	9.53E-04	10.00	0.023	0.000	0.000	0.000	2.93E-07
	5	Secondary	None	0.00	1.50	0.0000	7.23E-03	1.29E-06	2.00E-02	9.53E-04	10.00	0.023	1.000	0.500	0.050	2.36E-05
	6	Secondary	RF	0.90	1.50	0.5910	7.23E-03	1.29E-06	2.00E-02	9.53E-04	10.00	0.023	1.000	0.500	0.050	3.00E-06
	7	Secondary	Wk	0.90	1.92E-02	21.3971	7.23E-03	1.29E-06	2.00E-02	9.53E-04	10.00	0.023	1.000	0.500	0.050	2.08E-07
	8	Secondary	Shift	0.90	9.13E-04	37.9048	7.23E-03	1.29E-06	2.00E-02	9.53E-04	10.00	0.023	1.000	0.500	0.050	1.71E-07
	9	Primary	None	0.00	1.50	0.0000	7.23E-03	1.29E-06	2.00E-02	9.53E-04	10.00	0.023	1.000	0.900	0.090	1.84E-05
	10	Primary	RF	0.90	1.50	0.5910	7.23E-03	1.29E-06	2.00E-02	9.53E-04	10.00	0.023	1.000	0.900	0.090	2.15E-06
	11	Primary	Wk	0.90	1.92E-02	21.3971	7.23E-03	1.29E-06	2.00E-02	9.53E-04	10.00	0.023	1.000	0.900	0.090	1.47E-07
	12	Primary	Shift	0.90	9.13E-04	37.9048	7.23E-03	1.29E-06	2.00E-02	9.53E-04	10.00	0.023	1.000	0.900	0.090	1.21E-07
Pipe-to-valve (8)	1	None	None	0.00	1.50	0.0000	5.15E-03	9.20E-07	2.00E-02	6.80E-04	10.00	0.023	0.000	0.000	0.000	1.78E-05
	2	None	RF	0.90	1.50	0.5910	5.15E-03	9.20E-07	2.00E-02	6.80E-04	10.00	0.023	0.000	0.000	0.000	2.61E-06
	3	None	Wk	0.90	1.92E-02	21.3971	5.15E-03	9.20E-07	2.00E-02	6.80E-04	10.00	0.023	0.000	0.000	0.000	1.86E-07

Table D.C.1 Application of Markov Model to BWR-1 Cat 1 LOCA Frequency

Config. (Count)	INSPECTION CASE			INSPECTION INDEPENDENT PARAMETERS									INSPECTION DEPENDENT PARAMETERS			
	CASE	ISI Inspection Coverage	Leak Inspection Interval	P_{LI}	TSUBLI	MU	PHI	RHO_F	RHO_L	Lambda	T_{FI}	TSUBR	P_{FI}	P_{FD}	OMEGA	Haz{T}
	4	None	Shift	0.90	9.13E-04	37.9048	5.15E-03	9.20E-07	2.00E-02	6.80E-04	10.00	0.023	0.000	0.000	0.000	1.53E-07
	5	Secondary	None	0.00	1.50	0.0000	5.15E-03	9.20E-07	2.00E-02	6.80E-04	10.00	0.023	1.000	0.500	0.050	1.22E-05
	6	Secondary	RF	0.90	1.50	0.5910	5.15E-03	9.20E-07	2.00E-02	6.80E-04	10.00	0.023	1.000	0.500	0.050	1.56E-06
	7	Secondary	Wk	0.90	1.92E-02	21.3971	5.15E-03	9.20E-07	2.00E-02	6.80E-04	10.00	0.023	1.000	0.500	0.050	1.08E-07
	8	Secondary	Shift	0.90	9.13E-04	37.9048	5.15E-03	9.20E-07	2.00E-02	6.80E-04	10.00	0.023	1.000	0.500	0.050	8.88E-08
	9	Primary	None	0.00	1.50	0.0000	5.15E-03	9.20E-07	2.00E-02	6.80E-04	10.00	0.023	1.000	0.900	0.090	9.46E-06
	10	Primary	RF	0.90	1.50	0.5910	5.15E-03	9.20E-07	2.00E-02	6.80E-04	10.00	0.023	1.000	0.900	0.090	1.11E-06
	11	Primary	Wk	0.90	1.92E-02	21.3971	5.15E-03	9.20E-07	2.00E-02	6.80E-04	10.00	0.023	1.000	0.900	0.090	7.61E-08
	12	Primary	Shift	0.90	9.13E-04	37.9048	5.15E-03	9.20E-07	2.00E-02	6.80E-04	10.00	0.023	1.000	0.900	0.090	6.26E-08
Pipe-to-safe-end (2)	1	None	None	0.00	1.50	0.0000	1.24E-02	2.22E-06	2.00E-02	1.64E-03	10.00	0.023	0.000	0.000	0.000	9.72E-05
	2	None	RF	0.90	1.50	0.5910	1.24E-02	2.22E-06	2.00E-02	1.64E-03	10.00	0.023	0.000	0.000	0.000	1.39E-05
	3	None	Wk	0.90	1.92E-02	21.3971	1.24E-02	2.22E-06	2.00E-02	1.64E-03	10.00	0.023	0.000	0.000	0.000	9.84E-07
	4	None	Shift	0.90	9.13E-04	37.9048	1.24E-02	2.22E-06	2.00E-02	1.64E-03	10.00	0.023	0.000	0.000	0.000	8.10E-07
	5	Secondary	None	0.00	1.50	0.0000	1.24E-02	2.22E-06	2.00E-02	1.64E-03	10.00	0.023	1.000	0.500	0.050	6.72E-05
	6	Secondary	RF	0.90	1.50	0.5910	1.24E-02	2.22E-06	2.00E-02	1.64E-03	10.00	0.023	1.000	0.500	0.050	8.44E-06
	7	Secondary	Wk	0.90	1.92E-02	21.3971	1.24E-02	2.22E-06	2.00E-02	1.64E-03	10.00	0.023	1.000	0.500	0.050	5.83E-07
	8	Secondary	Shift	0.90	9.13E-04	37.9048	1.24E-02	2.22E-06	2.00E-02	1.64E-03	10.00	0.023	1.000	0.500	0.050	4.79E-07
	9	Primary	None	0.00	1.50	0.0000	1.24E-02	2.22E-06	2.00E-02	1.64E-03	10.00	0.023	1.000	0.900	0.090	5.27E-05
	10	Primary	RF	0.90	1.50	0.5910	1.24E-02	2.22E-06	2.00E-02	1.64E-03	10.00	0.023	1.000	0.900	0.090	6.11E-06
	11	Primary	Wk	0.90	1.92E-02	21.3971	1.24E-02	2.22E-06	2.00E-02	1.64E-03	10.00	0.023	1.000	0.900	0.090	4.16E-07
	12	Primary	Shift	0.90	9.13E-04	37.9048	1.24E-02	2.22E-06	2.00E-02	1.64E-03	10.00	0.023	1.000	0.900	0.090	3.42E-07
Pipe-to-tee (6)	1	None	None	0.00	1.50	0.0000	5.00E-03	8.92E-07	2.00E-02	6.59E-04	10.00	0.023	0.000	0.000	0.000	1.68E-05
	2	None	RF	0.90	1.50	0.5910	5.00E-03	8.92E-07	2.00E-02	6.59E-04	10.00	0.023	0.000	0.000	0.000	2.46E-06
	3	None	Wk	0.90	1.92E-02	21.3971	5.00E-03	8.92E-07	2.00E-02	6.59E-04	10.00	0.023	0.000	0.000	0.000	1.75E-07
	4	None	Shift	0.90	9.13E-04	37.9048	5.00E-03	8.92E-07	2.00E-02	6.59E-04	10.00	0.023	0.000	0.000	0.000	1.44E-07
	5	Secondary	None	0.00	1.50	0.0000	5.00E-03	8.92E-07	2.00E-02	6.59E-04	10.00	0.023	1.000	0.500	0.050	1.15E-05
	6	Secondary	RF	0.90	1.50	0.5910	5.00E-03	8.92E-07	2.00E-02	6.59E-04	10.00	0.023	1.000	0.500	0.050	1.47E-06
	7	Secondary	Wk	0.90	1.92E-02	21.3971	5.00E-03	8.92E-07	2.00E-02	6.59E-04	10.00	0.023	1.000	0.500	0.050	1.02E-07
	8	Secondary	Shift	0.90	9.13E-04	37.9048	5.00E-03	8.92E-07	2.00E-02	6.59E-04	10.00	0.023	1.000	0.500	0.050	8.37E-08
	9	Primary	None	0.00	1.50	0.0000	5.00E-03	8.92E-07	2.00E-02	6.59E-04	10.00	0.023	1.000	0.900	0.090	8.91E-06
	10	Primary	RF	0.90	1.50	0.5910	5.00E-03	8.92E-07	2.00E-02	6.59E-04	10.00	0.023	1.000	0.900	0.090	1.05E-06
	11	Primary	Wk	0.90	1.92E-02	21.3971	5.00E-03	8.92E-07	2.00E-02	6.59E-04	10.00	0.023	1.000	0.900	0.090	7.16E-08
	12	Primary	Shift	0.90	9.13E-04	37.9048	5.00E-03	8.92E-07	2.00E-02	6.59E-04	10.00	0.023	1.000	0.900	0.090	5.89E-08
Pipe-to-socket-weld (capped bypass) (2)	1	None	None	0.00	1.50	0.0000	1.11E-03	1.99E-07	2.00E-02	1.47E-04	10.00	0.023	0.000	0.000	0.000	8.70E-07
	2	None	RF	0.90	1.50	0.5910	1.11E-03	1.99E-07	2.00E-02	1.47E-04	10.00	0.023	0.000	0.000	0.000	1.29E-07
	3	None	Wk	0.90	1.92E-02	21.3971	1.11E-03	1.99E-07	2.00E-02	1.47E-04	10.00	0.023	0.000	0.000	0.000	9.23E-09
	4	None	Shift	0.90	9.13E-04	37.9048	1.11E-03	1.99E-07	2.00E-02	1.47E-04	10.00	0.023	0.000	0.000	0.000	7.59E-09
	5	Secondary	None	0.00	1.50	0.0000	1.11E-03	1.99E-07	2.00E-02	1.47E-04	10.00	0.023	1.000	0.500	0.050	5.89E-07

Table D.C.1 Application of Markov Model to BWR-1 Cat 1 LOCA Frequency

Config. (Count)	INSPECTION CASE			INSPECTION INDEPENDENT PARAMETERS									INSPECTION DEPENDENT PARAMETERS			
	CASE	ISI Inspection Coverage	Leak Inspection Interval	P_{LI}	TSUBLI	MU	PHI	RHO_F	RHO_L	Lambda	T_{FI}	TSUBR	P_{FI}	P_{FD}	OMEGA	Haz(T)
	6	Secondary	RF	0.90	1.50	0.5910	1.11E-03	1.99E-07	2.00E-02	1.47E-04	10.00	0.023	1.000	0.500	0.050	7.62E-08
	7	Secondary	Wk	0.90	1.92E-02	21.3971	1.11E-03	1.99E-07	2.00E-02	1.47E-04	10.00	0.023	1.000	0.500	0.050	5.29E-09
	8	Secondary	Shift	0.90	9.13E-04	37.9048	1.11E-03	1.99E-07	2.00E-02	1.47E-04	10.00	0.023	1.000	0.500	0.050	4.35E-09
	9	Primary	None	0.00	1.50	0.0000	1.11E-03	1.99E-07	2.00E-02	1.47E-04	10.00	0.023	1.000	0.900	0.090	4.55E-07
	10	Primary	RF	0.90	1.50	0.5910	1.11E-03	1.99E-07	2.00E-02	1.47E-04	10.00	0.023	1.000	0.900	0.090	5.40E-08
	11	Primary	Wk	0.90	1.92E-02	21.3971	1.11E-03	1.99E-07	2.00E-02	1.47E-04	10.00	0.023	1.000	0.900	0.090	3.70E-09
	12	Primary	Shift	0.90	9.13E-04	37.9048	1.11E-03	1.99E-07	2.00E-02	1.47E-04	10.00	0.023	1.000	0.900	0.090	3.04E-09

ATTACHMENT D TO APPENDIX D

SIGNIFICANT FAILURES OF SAFETY RELATED PIPING

Table D.D.1 is a list of selected significant pipe failures during the period 1970 – 2003. The list includes failures of Code Class 1 and 2 piping systems inside the containment/drywell and auxiliary/reactor building structures of commercial nuclear power plants. The technical information has been extracted from the OPDE database (Attachment A).

Table D.D.1 Selected Historical Pipe Failure Information

Event Date	Plant	Country	Estimated Peak Leak/Flow Rate [gpm]	Description
12/14/02	Brunsbüttel (BWR)	Germany	-- (see Description)	Rupture of reactor head cooling pipe (the rupture occurred in section of pipe that was separated from the RPV through an isolation valve – no RPV steam leakage observed
11/7/02	Hamaoka-1 (BWR)	Japan	>> 50	Rupture of pipe in High Pressure Coolant Injection system; the rupture occurred during a functional system test
7/12/99	Tsuruga-2 (PWR)	Japan	16	Thermal fatigue induced fracture of elbow connected to regenerative heat exchanger
5/12/98	Civaux-1 (PWR)	France	131	Thermal fatigue induced fracture of seam welded elbow in the Residual Heat Removal System
5/27/97	Calvert Cliffs-1 (PWR)	USA	8.0	Fractured pressurizer instrument sensing line; attributed to vibration fatigue
4/21/97	Oconee-2 (PWR)	USA	12.0	Thermal fatigue induced fracture of weld connecting HPI/NMU pipe to RCS (see Base Case PWR-3)
12/21/96	Dampierre-1 (PWR)	France	0.6	Thermal-fatigue induced weld crack in straight section of Safety Injection line to RCS hot leg.
3/8/95	Borssele (PWR)	The Netherlands	65.8	While in hot standby prior to startup a weld fractured on the High Head Safety Injection common discharge header; attributed to vibration-fatigue
2/23/95	Biblis-B (PWR)	Germany	15.8	Thermal fatigue induced fracture of base metal of pipe in Chemical and Volume Control system
3/3/94	Kola-2 (PWR)	Russia	S-LOCA	Soviet-designed PWR of type WWER-440/230; full circumferential fracture of NPS2 makeup pipe while shutting down for maintenance outage. Event resulted in High Pressure Safety Injection system actuation; a beyond-design basis accident.
9/20/92	Dampierre-2 (PWR)	France	3.2	Non-isolable, thermal fatigue induced weld fracture in Safety Injection System.
6/18/88	Tihange-1	Belgium	6.3	Thermal fatigue induced fracture of base

Table D.D.1 Selected Historical Pipe Failure Information

Event Date	Plant	Country	Estimated Peak Leak/Flow Rate [gpm]	Description
	(PWR)			metal of Safety Injection line to RCS hot leg
12/9/87	Farley-2 (PWR)	USA	0.7	Thermal fatigue induced weld fracture in Safety Injection line to RCS cold leg
8/16/87	McGuire-1 (PWR)	USA	39.5	Fracture (80% of circumference) of 1-inch socket weld in drain line off of letdown line inside containment. The weld fracture occurred during startup operations (8% reactor power)
5/31/86	Obrigheim (PWR)	Germany	0.32	Thermal fatigue induced weld fracture in makeup line to RCS.
7/29/85	Sequoyah-2 (PWR)	USA	60.0	Fractured sample line in Chemical and Volume Control system; attributed to vibration-fatigue
8/6/84	McGuire-2 (PWR)	USA	8.0	Water hammer induced fracture of socket weld in letdown line
1/25/83	Maine Yankee (PWR)	USA	100	Fractured main feedwater pipe adjacent to weld joining pipe and steam generator safe end; attributed to severe water hammer.
1/21/82	Crystal River-3 (PWR)	USA	1	140-degree circumferential crack in makeup line near valve-to-safe end weld; attributed to thermal fatigue
2/12/80	Santa Maria de Garona (BWR)	Spain	0.8	IGSCC induced through-wall flaw in Reactor Recirculation nozzle-to-safe end weld
8/29/80	TVO-1 (BWR)	Finland	315	Thermal fatigue induced fracture of tee in Reactor Water Cleanup system. The fracture occurred during the commissioning of this reactor unit.
6/14/78	Duane Arnold (BWR)	USA	3.0	IGSCC induced through-wall flaw in Reactor Recirculation nozzle-to-safe end weld
11/13/73	Indian Point-2 (PWR)	USA	15.8	180-degree circumferential crack of 18-inch feedwater line weld inside containment
4/28/70	H.B. Robinson-2 (PWR)	USA	>> 50[28]	360-degree break in 6-inch branch line between No. 3 steam generator main steam line and safety valve. The failure occurred during the final stages of hot functional testing

[28] At the time of the pipe break the primary system was at 278 C (533 F) and 15.3 MPa (2,225 psi) primary system pressure with a secondary system pressure of 6.2 MPa (900 psi)

APPENDIX E

PIPING BASE CASE RESULTS OF WILLIAM GALYEAN

APPENDIX E

PIPING BASE CASE RESULTS OF WILLIAM GALYEAN

E.1 Summary

In this base case study, LOCA frequencies are calculated using a "top-down" approach. Specifically, a total LOCA frequency is calculated using U.S. commercial nuclear power plant (NPP) operating experience. This total frequency is then allocated to the LOCA size categories, RCS subsystems and components, and degradation mechanisms. This allocation is performed using data on primary system leaks and cracks from both U.S. and foreign PWR and BWR reactors.

E.2 Assumptions and Observations

As with all analyses, there are a number of implicit assumptions associated with this approach. First is that past performance is representative of future performance. The common scenario for the occurrence of a LOCA starts with postulating the existence of a flaw or defect in the primary reactor coolant boundary. This flaw is then subjected to a stress that results in the catastrophic failure of the primary pressure boundary, producing a LOCA. The U.S. LWR operating experience to date consists of approximately 100 reactors with an average age of about 23 years. During this time the RCS of these plants have experience numerous transients and loads, which have produce a wide range of stresses. Whether these plants operate for 40 years (or 60 years with license extensions) this available operating experience represents a significant portion of the average plants lifetime. It is therefore reasonable to assume that the stresses that have already occurred are representative of those that will occur in the future. Similarly, various degradation mechanisms have affected RCS pipe, welds and components. However, when these degradation mechanisms have been detected, mitigation programs have subsequently been implemented (e.g., IGSCC in BWRs). Therefore, the number of flaws and defects in the RCS is likely cyclic over time. As the degradation mechanism manifests itself, the number of defects grows, as the degradation mechanism is addressed and mitigated, the number of defects is reduced. Again, the assumption here is that current 23 years of operating (on average, per reactor) are representative of the remaining operating life.

Another observation is the occurrence of zero LOCAs for both PWRs and BWRs. Although this does not prove that the LOCA frequencies are the same for both designs, it likewise does not support different LOCA frequencies. Therefore, for this analysis, the operating experience data (i.e., zero failures) will be pooled to generate a single LOCA frequency.

This analysis, just as every LOCA frequency estimate performed to date, assumes that the frequency of a LOCA decreases as pipe size increases. This might be attributable to a couple of issues. First, for small diameter pipe, some failure mechanisms exist that don't apply to larger diameter pipe (e.g., compression fitting failures and socket welds). Second, the same flaw in both a small diameter pipe and a large diameter pipe represents a large percentage of the pipe diameter in the small diameter pipe. Third, inspection is probably more thorough in larger diameter pipe so that the chance of a defect going undetected is less in the larger diameter pipe. For all of these reasons (and probably others), the total LOCA frequency is reduced as LOCA size category increases. The somewhat arbitrary scaling factor of ½ order of magnitude (assuming a lognormal probability distribution on LOCA frequency) appears to be reasonably consistent with historical LOCA frequency estimates.

The final premise of this base case analysis is that the relative frequency of precursor data (i.e., leaks and cracks) is an indicator of the relative frequency of LOCA events. In the calculations that follow, the total LOCA frequency is allocated to the different RCS subsystems and components, and the different degradation mechanisms according to the relative frequency of observed leaks and cracks attributable to these subsystems and mechanism. Note that in order to determine the relative frequencies, complete crack and leak data are not needed, only consistent data that has not been biased by the over reporting of one attribute relative to another. Completeness in the data is neither required nor important, only consistency.

E.3 Total LOCA Frequency Estimates

The total LOCA frequency is calculated using U.S. NPP experience of zero Category-1 LOCAs (i.e., greater than 380 lpm [100 gpm]) in 2,647 LWR-years of operation (as of 4/24/2003). A Bayesian update of a non-informative prior-distribution was performed to produce a total LOCA frequency of 1.9E-4 per LWR-year.

Table E.1 Total LOCA Frequency (per LWR-Year) Including Uncertainty, Using a Non-Informative Prior and U.S. LWR Operating Experience

5%	50%	mean	95%
7.4E-07	8.6E-05	1.9E-04	7.3E-04

E.4 LOCA Frequency Allocation by RCS Pipe and Non-Pipe

The total LOCA frequency calculated above is first allocated between pipe and non-pipe passive components using data on primary system leaks and cracks collected from licensee event reports (LERs). These data records were collected, reviewed and categorized specifically for this effort. Since these data will only be used to ascribe a relative frequency between pipe and passive non-pipe components, complete data are not necessary, only data that have been reported consistently. These data and the resultant allocation are summarized in the table below. Steam generator tube ruptures are being assessed separately, and are therefore removed from this allocation.

Table E.2 Allocation of LOCA Frequency Between Pipe and Passive Non-Pipe Components

Reactor Coolant Pressure Boundary failure (cracks or leaks) events 1990-2002 LERs

	LWR	PWR	BWR
Total number of failure events	448	388	60
Number of SG tube failure events	112	112	0
Total minus SG events	336	276	60
Number of pipe failure events	54	24	30
Exclude SG tube events since these can be estimated directly			
Therefore			
Number of non-pipe failure events	282	252	30
fraction of LOCA frequency attributed to			
pipes	0.16	0.09	0.50
non-pipes	0.84	0.91	0.50
total LOCA frequency =	1.9E-04		
LOCA frequency attributable to			
pipes	3.0E-05	1.6E-05	9.4E-05
non-pipes	1.6E-04	1.7E-04	9.4E-05

E.5 LOCA Frequency by Size Category

The total LOCA frequencies calculated above are for Category-1 LOCAs. The simple approach taken here is that the LOCA frequency is reduced by ½ order of magnitude (assuming a lognormal distribution), for each step up in size category. There are a number of reasons for this approach. Between the smallest pipe size categories (i.e., < 2 inches, and > 2 inches) there is a significant difference in the failure mechanisms. For the smallest pipes, the operating experience includes failures of compression fittings and socket welds, which are not used in larger size pipe. Also, a number of studies on crack and leak events indicate a decrease in these precursor frequencies, as pipe diameter increases. Lastly, virtually every estimate of LOCA frequencies ever made has resulted in a reduced frequency for the large LOCA sizes.

E.6 LOCA Frequency by Degradation Mechanism and Subsystem

In addition to the LER data used to allocate the LOCA frequency between pipe and passive non-pipe components, the LOCA frequency was further allocated among the different degradation mechanisms observed and among the different RCS subsystems and components defined for this project. These allocations were based on data collected from both U.S. reactor operating experience (primarily LERs), and from foreign LWR operating experience (SLAP database). One complication to this approach is the IGSCC-related experience in U.S. BWR plants. IGSCC was an issue for BWRs in the early 1980's. Many U.S. BWRs implemented IGSCC mitigation programs in the mid-1980's, which have greatly reduced the occurrence of IGSCC. To avoid unrealistically over weighting the IGSCC mechanism, the

BWR experience was segregated and only the post 1985 experience was used for allocating the relative contribution to LOCA frequency by degradation mechanism.

Lastly, although the guidance for calculating base-case frequencies for this project included estimates for 25 years, 40 years and 60 years, this particular base-case calculation assumed that the frequencies were generally independent of plant life. This is based on the IGSCC experience that demonstrated that although degradation mechanisms are at work that can result in an increase in the LOCA frequency over time, so to are mitigation programs and general performance improvement programs (e.g., more effective inspections), that can result in a decrease in the LOCA frequency. Therefore, overall these competing effects are assumed to cancel each other out for a net zero effect on LOCA frequency. That is, the current LOCA frequency (approximately 25 year life) is assumed to be application for 40 and 60 years as well.

E.7 LOCA Frequency Tables

The following tables display the detailed results of Base Case #1 calculations. The legend of degradation categories is shown in Table E.3. LOCA data are presented on the tables listed below:

PWR pipe	Table E.4
PWR passive non-pipe	Table E.5
BWR pipe	Table E.6
BWR passive non-pipe	Table E.7

Table E.3 Degradation Categories

Deg Mech	DM Description
MA	Material Aging
FDR	Fabrication Defect and Repair
SCC	Stress Corrosion Cracking
LC	Local Corrosion
MF	Mechanical Fatigue
TF	Thermal Fatigue
FS	Flow Sensitive (includes FAC and E/C)
UNK	Unknown

Table E.4 PWR LOCA Frequency (for Pipes) Allocated by System, Degradation Mechanism, and Size Category

SLAP event counts by system (for PWRs)
Only includes RCPB (S-type)

LOCA System	Event Counts	Deg. Mech. Fractional Contribution	Est. LOCA Fraction by System	Min (mm) (in)	Max (mm) (in)	# of Welds Dom. Total	Total LOCA Freq 1.6E-05	Cat-1	Cat-2	Cat-3	Cat-4	Cat-5	Cat-6	
							greater than	100	1500	5000	25000	100000	500000	gpm
								0.4	1.7	3.0	6.8	14.0937	31.5146	dia. (in.)
								1	0.3	0.1	0.03	0.01	0.003	probabilty
CRDM	4.6		0.01	70	100		1.4E-07	1.4E-07	4.1E-08	1.4E-08	0.0	0.0	0.0	
CRDM	***4***	*1*		2.8	3.9	*1*	***1.4E-07***	***1.4E-07***	***4.1E-08***	***1.4E-08***	–	–	–	
UA		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	–	–	–	
MA		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	–	–	–	
LC		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	–	–	–	
FDR		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	–	–	–	
SCC	*4*	*1.00*					*1.4E-07*	*1.4E-07*	*4.1E-08*	*1.4E-08*	–	–	–	
MF		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	–	–	–	
TF		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	–	–	–	
FS		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	–	–	–	
UK		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	–	–	–	
CRDM pipe	5.0		0.01	60	100		1.5E-07	1.5E-07	4.5E-08	1.5E-08	0.0	0.0	0.0	
CRDM pipe	***5***	*1*		2.4	3.9	*1*	***1.5E-07***	***1.5E-07***	***4.5E-08***	***1.5E-08***	–	–	–	
UA		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	–	–	–	
MA		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	–	–	–	
LC		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	–	–	–	
FDR	*1*	*0.20*					*3.0E-08*	*3.0E-08*	*9.0E-09*	*3.0E-09*	–	–	–	
SCC	*2*	*0.40*					*6.0E-08*	*6.0E-08*	*1.8E-08*	*6.0E-09*	–	–	–	
MF		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	–	–	–	
TF		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	–	–	–	
FS		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	–	–	–	
UK	*2*	*0.40*					*6.0E-08*	*6.0E-08*	*1.8E-08*	*6.0E-09*	–	–	–	
CVCS	140.0		0.26	13	250		4.2E-06	4.2E-06	1.3E-06	4.2E-07	1.3E-07	0.0	0.0	
CVCS	***140***	*1*		0.5	9.8	*1*	***4.2E-06***	***4.2E-06***	***1.3E-06***	***4.2E-07***	***1.3E-07***	–	–	
UA		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	–	–	
MA		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	–	–	
LC		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	–	–	
FDR	*12*	*0.09*					*3.6E-07*	*3.6E-07*	*1.1E-07*	*3.6E-08*	*1.1E-08*	–	–	
SCC	*19*	*0.14*					*5.7E-07*	*5.7E-07*	*1.7E-07*	*5.7E-08*	*1.7E-08*	–	–	
MF	*94*	*0.67*					*2.8E-06*	*2.8E-06*	*8.5E-07*	*2.8E-07*	*8.5E-08*	–	–	
TF	*7*	*0.05*					*2.1E-07*	*2.1E-07*	*6.3E-08*	*2.1E-08*	*6.3E-09*	–	–	
FS	*6*	*0.04*					*1.8E-07*	*1.8E-07*	*5.4E-08*	*1.8E-08*	*5.4E-09*	–	–	

SLAP event counts by system (for PWRs)
Only includes RCPB (S-type)

LOCA System	Event Counts	Deg. Mech. Fractional Contribution	Est. LOCA Fraction by System	Min (mm) (in)	Max (mm) (in)	# of Welds Dom. Total	greater than / Total LOCA Freq 1.6E-05	Cat-1	Cat-2	Cat-3	Cat-4	Cat-5	Cat-6	
							greater than	100	1500	5000	25000	100000	500000	gpm
								0.4	1.7	3.0	6.8	14.0937	31.5146	dia. (in.)
								1	0.3	0.1	0.03	0.01	0.003	probabilty
UK	*2*	*0.01*					*6.0E-08*	*6.0E-08*	*1.8E-08*	*6.0E-09*	*1.8E-09*	--	--	
Drain Lines	58.5		0.11	10	80		1.8E-06	1.8E-06	5.3E-07	1.8E-07	0.0	0.0	0.0	
Drain Lines	***46***	***1***		0.4	3.1	***1***	***1.8E-06***	***1.8E-06***	***5.3E-07***	***1.8E-07***	--	--	--	
UA		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	--	--	
MA		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	--	--	
LC		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	--	--	
FDR	*9*	*0.20*					*3.4E-07*	*3.4E-07*	*1.0E-07*	*3.4E-08*	--	--	--	
SCC	*5*	*0.11*					*1.9E-07*	*1.9E-07*	*5.7E-08*	*1.9E-08*	--	--	--	
MF	*27*	*0.59*					*1.0E-06*	*1.0E-06*	*3.1E-07*	*1.0E-07*	--	--	--	
TF	*2*	*0.04*					*7.7E-08*	*7.7E-08*	*2.3E-08*	*7.7E-09*	--	--	--	
FS	*2*	*0.04*					*7.7E-08*	*7.7E-08*	*2.3E-08*	*7.7E-09*	--	--	--	
UK	*1*	*0.02*					*3.8E-08*	*3.8E-08*	*1.1E-08*	*3.8E-09*	--	--	--	
In-Core Instr.	16.6		0.03	10	25		5.0E-07	5.0E-07	0.0	0.0	0.0	0.0	0.0	
In-Core Instr.	***13***	***1***		0.4	1.0	***1***	***5.0E-07***	***5.0E-07***	--	--	--	--	--	
UA		*0.00*					*0.0E+00*	*0.0E+00*	--	--	--	--	--	
MA		*0.00*					*0.0E+00*	*0.0E+00*	--	--	--	--	--	
LC	*4*	*0.31*					*1.5E-07*	*1.5E-07*	--	--	--	--	--	
FDR	*2*	*0.15*					*7.7E-08*	*7.7E-08*	--	--	--	--	--	
SCC	*2*	*0.15*					*7.7E-08*	*7.7E-08*	--	--	--	--	--	
MF	*2*	*0.15*					*7.7E-08*	*7.7E-08*	--	--	--	--	--	
TF		*0.00*					*0.0E+00*	*0.0E+00*	--	--	--	--	--	
FS		*0.00*					*0.0E+00*	*0.0E+00*	--	--	--	--	--	
UK	*3*	*0.23*					*1.2E-07*	*1.2E-07*	--	--	--	--	--	
Instr. lines	151.8		0.28	9	200		4.6E-06	4.6E-06	1.4E-06	4.6E-07	1.4E-07	0.0	0.0	
Instr. Lines	***119***	***1***		0.4	7.9	***1***	***4.6E-06***	***4.6E-06***	***1.4E-06***	***4.6E-07***	***1.4E-07***	--	--	
UA		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	--	
MA	*1*	*0.01*					*3.8E-08*	*3.8E-08*	*1.2E-08*	*3.8E-09*	*1.2E-09*	--	--	
LC	*1*	*0.01*					*3.8E-08*	*3.8E-08*	*1.2E-08*	*3.8E-09*	*1.2E-09*	--	--	
FDR	*11*	*0.09*					*4.2E-07*	*4.2E-07*	*1.3E-07*	*4.2E-08*	*1.3E-08*	--	--	
SCC	*19*	*0.16*					*7.3E-07*	*7.3E-07*	*2.2E-07*	*7.3E-08*	*2.2E-08*	--	--	
MF	*74*	*0.62*					*2.8E-06*	*2.8E-06*	*8.5E-07*	*2.8E-07*	*8.5E-08*	--	--	
TF	*2*	*0.02*					*7.7E-08*	*7.7E-08*	*2.3E-08*	*7.7E-09*	*2.3E-09*	--	--	
FS	*1*	*0.01*					*3.8E-08*	*3.8E-08*	*1.2E-08*	*3.8E-09*	*1.2E-09*	--	--	
UK	*10*	*0.08*					*3.8E-07*	*3.8E-07*	*1.2E-07*	*3.8E-08*	*1.2E-08*	--	--	

SLAP event counts by system (for PWRs)
Only includes RCPB (S-type)

LOCA System	Event Counts	Deg. Mech. Fractional Contribution	Est. LOCA Fraction by System	Min (mm) (in)	Max (mm) (in)	# of Welds Dom. Total	greater than Total LOCA Freq 1.6E-05	Cat-1	Cat-2	Cat-3	Cat-4	Cat-5	Cat-6	
								100	1500	5000	25000	100000	500000	gpm
								0.4	1.7	3.0	6.8	14.0937	31.5146	dia. (in.)
								1	0.3	0.1	0.03	0.01	0.003	probabilty
Pressurizer	6.4		0.01	25	25		1.9E-07	1.9E-07	5.8E-08	0.0	0.0	0.0	0.0	
Pressurizer	*5*	*1*		1.0	1.0	*1*	*1.9E-07*	*1.9E-07*	*5.8E-08*	--	--	--	--	
UA		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	--	--	--	--	
MA		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	--	--	--	--	
LC		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	--	--	--	--	
FDR	*2*	*0.40*					*7.7E-08*	*7.7E-08*	*2.3E-08*	--	--	--	--	
SCC	*2*	*0.40*					*7.7E-08*	*7.7E-08*	*2.3E-08*	--	--	--	--	
MF		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	--	--	--	--	
TF	*1*	*0.20*					*3.8E-08*	*3.8E-08*	*1.2E-08*	--	--	--	--	
FS		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	--	--	--	--	
UK		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	--	--	--	--	
Pzr Spray Lines	6.4		0.01	20	75		1.9E-07	1.9E-07	5.8E-08	1.9E-08	0.0	0.0	0.0	
Pzr Spray Lines	*5*	*1*		0.8	3.0	*1*	*1.9E-07*	*1.9E-07*	*5.8E-08*	*1.9E-08*	--	--	--	
UA		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	--	--	
MA		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	--	--	
LC		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	--	--	
FDR		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	--	--	
SCC	*1*	*0.20*					*3.8E-08*	*3.8E-08*	*1.2E-08*	*3.8E-09*	--	--	--	
MF	*3*	*0.60*					*1.2E-07*	*1.2E-07*	*3.5E-08*	*1.2E-08*	--	--	--	
TF		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	--	--	
FS		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	--	--	
UK	*1*	*0.20*					*3.8E-08*	*3.8E-08*	*1.2E-08*	*3.8E-09*	--	--	--	
RCP cold-leg	1.3		0.002				3.8E-08	3.8E-08	1.2E-08	3.8E-09	1.2E-09	3.8E-10	1.2E-10	
RCP cold-leg	*1*	*1*				*1*	*3.8E-08*	*3.8E-08*	*1.2E-08*	*3.8E-09*	*1.2E-09*	*3.8E-10*	*1.2E-10*	
UA		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	
MA		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	
LC		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	
FDR		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	
SCC	*1*	*1.00*					3.8E-08	3.8E-08	1.2E-08	3.8E-09	1.2E-09	3.8E-10	1.2E-10	
MF		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	
TF		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	
FS		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	
UK		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	
RCS hot-leg	5.1		0.009	25	650		1.5E-07	1.5E-07	4.6E-08	1.5E-08	4.6E-09	1.5E-09	0.0E+00	
RCP hot-leg -	*4.5*	*1*		1.0	25.6	*1*	*1.5E-07*	*1.5E-07*	*4.6E-08*	*1.5E-08*	*4.6E-09*	*1.5E-09*	--	

SLAP event counts by system (for PWRs)
Only includes RCPB (S-type)

LOCA System	Event Counts	Deg. Mech. Fractional Contribution	Est. LOCA Fraction by System	Min (mm) (in)	Max (mm) (in)	# of Welds Dom. Total	Total LOCA Freq 1.6E-05	Cat-1	Cat-2	Cat-3	Cat-4	Cat-5	Cat-6	
							greater than	100	1500	5000	25000	100000	500000	gpm
								0.4	1.7	3.0	6.8	14.0937	31.5146	dia. (in.)
								1	0.3	0.1	0.03	0.01	0.003	probabilty
BC														
UA		*0.00*				40.0	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	–	
MA		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	–	
LC		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	–	
FDR	*2*	*0.44*					*6.8E-08*	*6.8E-08*	*2.1E-08*	*6.8E-09*	*2.1E-09*	*6.8E-10*	–	
SCC	*0.5*	*0.11*					*1.7E-08*	*1.7E-08*	*5.1E-09*	*1.7E-09*	*5.1E-10*	*1.7E-10*	–	
MF	*1*	*0.22*					*3.4E-08*	*3.4E-08*	*1.0E-08*	*3.4E-09*	*1.0E-09*	*3.4E-10*	–	
TF	*1*	*0.22*					*3.4E-08*	*3.4E-08*	*1.0E-08*	*3.4E-09*	*1.0E-09*	*3.4E-10*	–	
FS		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	–	
UK		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	–	
Rx-Head	1.3		0.002				3.8E-08	3.8E-08	1.2E-08	3.8E-09	1.2E-09	3.8E-10	1.2E-10	
Rx-Head	***1***	***1***				***1***	***3.8E-08***	***3.8E-08***	***1.2E-08***	***3.8E-09***	***1.2E-09***	***3.8E-10***	***1.2E-10***	
UA		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	
MA		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	
LC		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	
FDR		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	
SCC		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	
MF		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	
TF		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	
FS		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	
UK	*1*	*1.00*					3.8E-08	3.8E-08	1.2E-08	3.8E-09	1.2E-09	3.8E-10	1.2E-10	
RHR	49.0		0.09	13	500		1.5E-06	1.5E-06	4.4E-07	1.5E-07	4.4E-08	1.5E-08	0.0	
RHR	***49***	***1***		0.5	19.7	***1***	***1.5E-06***	***1.5E-06***	***4.4E-07***	***1.5E-07***	***4.4E-08***	***1.5E-08***	–	
UA		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	–	
MA		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	–	
LC		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	–	
FDR	*9*	*0.18*					*2.7E-07*	*2.7E-07*	*8.1E-08*	*2.7E-08*	*8.1E-09*	*2.7E-09*	–	
SCC	*9*	*0.18*					*2.7E-07*	*2.7E-07*	*8.1E-08*	*2.7E-08*	*8.1E-09*	*2.7E-09*	–	
MF	*23*	*0.47*					*6.9E-07*	*6.9E-07*	*2.1E-07*	*6.9E-08*	*2.1E-08*	*6.9E-09*	–	
TF	*1*	*0.02*					*3.0E-08*	*3.0E-08*	*9.0E-09*	*3.0E-09*	*9.0E-10*	*3.0E-10*	–	
FS	*4*	*0.08*					*1.2E-07*	*1.2E-07*	*3.6E-08*	*1.2E-08*	*3.6E-09*	*1.2E-09*	–	
UK	*3*	*0.06*					*9.0E-08*	*9.0E-08*	*2.7E-08*	*9.0E-09*	*2.7E-09*	*9.0E-10*	–	
SIS Accum	14.3		0.03	15	100		4.3E-07	4.3E-07	1.3E-07	4.3E-08	0.0	0.0	0.0	
SIS Accum	***14***	***1***		0.6	3.9	***1***	***4.3E-07***	***4.3E-07***	***1.3E-07***	***4.3E-08***	–	–	–	
UA		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	–	–	–	

SLAP event counts by system (for PWRs)								Cat-1	Cat-2	Cat-3	Cat-4	Cat-5	Cat-6	
Only includes RCPB (S-type)			Est.			# of	greater than	100	1500	5000	25000	100000	500000	gpm
		Deg. Mech.	LOCA	Min	Max	Welds	Total	0.4	1.7	3.0	6.8	14.0937	31.5146	dia. (in.)
	Event	Fractional	Fraction	(mm)	(mm)	Dom.	LOCA Freq	1	0.3	0.1	0.03	0.01	0.003	probabilty
LOCA System	Counts	Contribution	by System	(in)	(in)	Total	1.6E-05							
MA		0.00					0.0E+00	0.0E+00	0.0E+00	0.0E+00	--	--	--	
LC		0.00					0.0E+00	0.0E+00	0.0E+00	0.0E+00	--	--	--	
FDR	2	0.14					6.1E-08	6.1E-08	1.8E-08	6.1E-09	--	--	--	
SCC	5	0.36					1.5E-07	1.5E-07	4.6E-08	1.5E-08	--	--	--	
MF	6	0.43					1.8E-07	1.8E-07	5.5E-08	1.8E-08	--	--	--	
TF	1	0.07					3.1E-08	3.1E-08	9.2E-09	3.1E-09	--	--	--	
FS		0.00					0.0E+00	0.0E+00	0.0E+00	0.0E+00	--	--	--	
UK		0.00					0.0E+00	0.0E+00	0.0E+00	0.0E+00	--	--	--	
SIS inj	78.0		0.14	9	600		2.3E-06	2.3E-06	7.0E-07	2.3E-07	7.0E-08	2.3E-08	0.0	
SIS Inj	**78**	**1**		**0.4**	**23.6**	**1**	**2.3E-06**	**2.3E-06**	**7.0E-07**	**2.3E-07**	**7.0E-08**	**2.3E-08**	--	
UA		0.00				114	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	--	
MA		0.00					0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	--	
LC		0.00					0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	--	
FDR	5	0.06					1.5E-07	1.5E-07	4.5E-08	1.5E-08	4.5E-09	1.5E-09	--	
SCC	28	0.36					8.4E-07	8.4E-07	2.5E-07	8.4E-08	2.5E-08	8.4E-09	--	
MF	26	0.33					7.8E-07	7.8E-07	2.3E-07	7.8E-08	2.3E-08	7.8E-09	--	
TF	9	0.12					2.7E-07	2.7E-07	8.1E-08	2.7E-08	8.1E-09	2.7E-09	--	
FS	4	0.05					1.2E-07	1.2E-07	3.6E-08	1.2E-08	3.6E-09	1.2E-09	--	
UK	6	0.08					1.8E-07	1.8E-07	5.4E-08	1.8E-08	5.4E-09	1.8E-09	--	
SRV lines	6.4		0.01	50	75		1.9E-07	1.9E-07	5.8E-08	1.9E-08	0.0	0.0	0.0	
SRV lines	**6**	**1**		**2.0**	**3.0**	**1**	**1.9E-07**	**1.9E-07**	**5.8E-08**	**1.9E-08**	--	--	--	
UA		0.00					0.0E+00	0.0E+00	0.0E+00	0.0E+00	--	--	--	
MA		0.00					0.0E+00	0.0E+00	0.0E+00	0.0E+00	--	--	--	
LC		0.00					0.0E+00	0.0E+00	0.0E+00	0.0E+00	--	--	--	
FDR	1	0.17					3.2E-08	3.2E-08	9.6E-09	3.2E-09	--	--	--	
SCC	3	0.50					9.6E-08	9.6E-08	2.9E-08	9.6E-09	--	--	--	
MF	1	0.17					3.2E-08	3.2E-08	9.6E-09	3.2E-09	--	--	--	
TF		0.00					0.0E+00	0.0E+00	0.0E+00	0.0E+00	--	--	--	
FS		0.00					0.0E+00	0.0E+00	0.0E+00	0.0E+00	--	--	--	
UK	1	0.17					3.2E-08	3.2E-08	9.6E-09	3.2E-09	--	--	--	
Surge Line	0.5		0.001				1.5E-08	1.5E-08	4.5E-09	1.5E-09	4.5E-10	0.0E+00	0.0E+00	
Surge Line - BC	**0.5**	**1**			**10**	**1**	**1.5E-08**	**1.5E-08**	**4.5E-09**	**1.5E-09**	**4.5E-10**	--	--	
UA	0.0	0.00				13	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	--	--	
MA	0.0	0.00					0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	--	--	
LC		0.00					0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	--	--	

SLAP event counts by system (for PWRs)								Cat-1	Cat-2	Cat-3	Cat-4	Cat-5	Cat-6	
Only includes RCPB (S-type)			Est.			# of	greater than	100	1500	5000	25000	100000	500000	gpm
		Deg. Mech.	LOCA	Min	Max	Welds	Total	0.4	1.7	3.0	6.8	14.0937	31.5146	dia. (in.)
	Event	Fractional	Fraction	(mm)	(mm)	Dom.	LOCA Freq	1	0.3	0.1	0.03	0.01	0.003	probabilty
LOCA System	**Counts**	**Contribution**	**by System**	**(in)**	**(in)**	**Total**	**1.6E-05**							
FDR		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	--	
SCC	*0.3*	*0.50*					*7.5E-09*	*7.5E-09*	*2.3E-09*	*7.5E-10*	*2.3E-10*	--	--	
MF		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	--	
TF	*0.3*	*0.50*					*7.5E-09*	*7.5E-09*	*2.3E-09*	*7.5E-10*	*2.3E-10*	--	--	
FS		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	--	
UK		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	--	
total	545		1				1.64E-05	1.64E-05	4.78E-06	1.57E-06	3.86E-07	4.06E-08	2.31E-10	

Table E.5 PWR Passive, Non-Pipe Component LOCA Frequency

PWR Non-Pipe LOCA Contributors												
Total non-pipe event count =				252	PWR							
		252			Non-Pipe	*Cat-1*	*Cat-2*	*Cat-3*	*Cat-4*	*Cat-5*	*Cat-6*	
Non-Pipe					LOCA	*100*	*1500*	*5000*	*25000*	*100000*	*500000*	*gpm*
LOCA	Degradation Mechanism			Non-Pipe	Freq	*0.5*	*1.8*	*3.3*	*7.3*	*18.4*	*41.2*	*dia. (in.)*
System		Counts	Fract Contr	LOCA Fract	1.7E-04	*1*	*0.3*	*0.1*	*0.03*	*0.01*	*0.003*	*probabilty*
LIV		**3**	**1.0**	**0.01**	**2.05E-06**	**2.05E-06**	**6.16E-07**	**2.05E-07**	**6.16E-08**	**2.05E-08**	**6.16E-09**	
	FDR		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
	FS		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
	LC		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
	MA	2	0.67		1.37E-06	1.37E-06	4.10E-07	1.37E-07	4.10E-08	1.37E-08	4.10E-09	
	MF		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
	SCC	1	0.33		6.84E-07	6.84E-07	2.05E-07	6.84E-08	2.05E-08	6.84E-09	2.05E-09	
	TF		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
	UNK		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
PIV		**1**	**1.0**	**0.00**	**6.84E-07**	**6.84E-07**	**2.05E-07**	**6.84E-08**	**2.05E-08**	**6.84E-09**	**2.05E-09**	
	FDR		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
	FS		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
	LC		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
	MA	1	1.00		2.05E-06	2.05E-06	6.16E-07	2.05E-07	6.16E-08	2.05E-08	6.16E-09	
	MF		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
	SCC		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
	TF		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
	UNK		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
Pzr		**65**	**1.0**	**0.26**	**4.45E-05**	**4.45E-05**	**1.33E-05**	**4.45E-06**	**1.33E-06**	**4.45E-07**	**1.33E-07**	
	FDR	4	0.06		1.26E-07	1.26E-07	3.79E-08	1.26E-08	3.79E-09	1.26E-09	3.79E-10	
	FS		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
	LC	1	0.02		3.16E-08	3.16E-08	9.47E-09	3.16E-09	9.47E-10	3.16E-10	9.47E-11	
	MA	1	0.02		3.16E-08	3.16E-08	9.47E-09	3.16E-09	9.47E-10	3.16E-10	9.47E-11	
	MF		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
	SCC	59	0.91		1.86E-06	1.86E-06	5.59E-07	1.86E-07	5.59E-08	1.86E-08	5.59E-09	
	TF		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	

PWR Non-Pipe LOCA Contributors												
Total non-pipe event count = 252				252	PWR							
					Non-Pipe	*Cat-1*	*Cat-2*	*Cat-3*	*Cat-4*	*Cat-5*	*Cat-6*	
Non-Pipe					LOCA	*100*	*1500*	*5000*	*25000*	*100000*	*500000*	*gpm*
LOCA		Degradation Mechanism		Non-Pipe	Freq	*0.5*	*1.8*	*3.3*	*7.3*	*18.4*	*41.2*	*dia. (in.)*
System		Counts	Fract Contr	LOCA Fract	1.7E-04	*1*	*0.3*	*0.1*	*0.03*	*0.01*	*0.003*	*probabilty*
	UNK		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
RCP		**4**	**1.0**	**0.02**	**2.74E-06**	**2.74E-06**	**8.21E-07**	**2.74E-07**	**8.21E-08**	**2.74E-08**	**8.21E-09**	
	FDR	3	0.75		1.54E-06	1.54E-06	4.62E-07	1.54E-07	4.62E-08	1.54E-08	4.62E-09	
	FS		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
	LC		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
	MA		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
	MF		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
	SCC		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
	TF		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
	UNK	1	0.25		5.13E-07	5.13E-07	1.54E-07	5.13E-08	1.54E-08	5.13E-09	1.54E-09	
RPV		**173**	**1.00**	**0.69**	**1.18E-04**	**1.18E-04**	**3.55E-05**	**1.18E-05**	**3.55E-06**	**1.18E-06**	**3.55E-07**	
	FDR	6	0.03		7.12E-08	7.12E-08	2.14E-08	7.12E-09	2.14E-09	7.12E-10	2.14E-10	
	FS		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
	LC	7	0.04		8.30E-08	8.30E-08	2.49E-08	8.30E-09	2.49E-09	8.30E-10	2.49E-10	
	MA		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
	MF		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
	SCC	153	0.88		1.82E-06	1.82E-06	5.45E-07	1.82E-07	5.45E-08	1.82E-08	5.45E-09	
	TF		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
	UNK	7	0.04		8.30E-08	8.30E-08	2.49E-08	8.30E-09	2.49E-09	8.30E-10	2.49E-10	
SG		**6**	**1.00**	**0.02**	**4.10E-06**	**4.10E-06**	**1.23E-06**	**4.10E-07**	**1.23E-07**	**4.10E-08**	**1.23E-08**	
	FDR		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
	FS	1	0.17		3.42E-07	3.42E-07	1.03E-07	3.42E-08	1.03E-08	3.42E-09	1.03E-09	
	LC		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
	MA		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
	MF	1	0.17		3.42E-07	3.42E-07	1.03E-07	3.42E-08	1.03E-08	3.42E-09	1.03E-09	
	SCC	2	0.33		6.84E-07	6.84E-07	2.05E-07	6.84E-08	2.05E-08	6.84E-09	2.05E-09	
	TF		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
	UNK	2	0.33		6.84E-07	6.84E-07	2.05E-07	6.84E-08	2.05E-08	6.84E-09	2.05E-09	

Table E.6 BWR LOCA Frequency (for Pipes) Allocated by System, Degradation Mechanism, and Size Category

SLAP event counts by system (for BWRs)								*Cat-1*	*Cat-2*	*Cat-3*	*Cat-4*	*Cat-5*	*Cat-6*	
Only includes RCPB (S-type)			Est.	Pipe Sizes		# of	*greater than*	*100*	*1500*	*5000*	*25000*	*100000*	*500000*	*gpm*
		Deg. Mech.	LOCA	Min	Max	Welds	Total	*0.5*	*1.8*	*3.3*	*7.3*	*18.4*	*41.2*	*dia. (in.)*
	Event	Fractional	Fraction	(mm)	(mm)	Dom.	LOCA Freq	*1*	*0.3*	*0.1*	*0.03*	*0.01*	*0.003*	*probabilty*
LOCA System	Counts	Contribution	by System	(in)	(in)	Total	9.4E-05							
CRD Piping	16		0.09	10	550		8.0E-06	8.0E-06	2.4E-06	8.0E-07	2.4E-07	8.0E-08	0.0	
CRD Piping	***16***	***1***		0.4	21.7	1	***8.0E-06***	***8.0E-06***	***2.4E-06***	***8.0E-07***	***2.4E-07***	***8.0E-08***	--	
UA		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	
MA		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	
LC	*0*	*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	
FDR	*3*	*0.19*					*1.5E-06*	*1.5E-06*	*4.5E-07*	*1.5E-07*	*4.5E-08*	*1.5E-08*	--	
SCC	*12*	*0.75*					*6.0E-06*	*6.0E-06*	*1.8E-06*	*6.0E-07*	*1.8E-07*	*6.0E-08*	--	
MF	*0*	*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	
TF	*0*	*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	
FS		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	
UK	*1*	*0.06*					*5.0E-07*	*5.0E-07*	*1.5E-07*	*5.0E-08*	*1.5E-08*	*5.0E-09*	--	
Drain Lines	11		0.06	19	250		5.5E-06	5.5E-06	1.7E-06	5.5E-07	1.7E-07	0.0	0.0	
Drain Lines	***11***	***1***		0.7	9.8	1	***5.5E-06***	***5.5E-06***	***1.7E-06***	***5.5E-07***	***1.7E-07***	--	--	
UA		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	--	
MA		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	--	
LC		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	--	
FDR	*0*	*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	--	
SCC	*2*	*0.18*					*1.0E-06*	*1.0E-06*	*3.0E-07*	*1.0E-07*	*3.0E-08*	--	--	
MF	*8*	*0.73*					*4.0E-06*	*4.0E-06*	*1.2E-06*	*4.0E-07*	*1.2E-07*	--	--	
TF	*0*	*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	--	
FS	*1*	*0.09*					*5.0E-07*	*5.0E-07*	*1.5E-07*	*5.0E-08*	*1.5E-08*	--	--	
UK		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	--	
Feedwater	8		0.04	8	300		4.0E-06	4.0E-06	1.2E-06	4.0E-07	1.2E-07	0.0	0.0	
Feedwater - BC	***8***	***1***		0.3	11.8	1	***4.0E-06***	***4.0E-06***	***1.2E-06***	***4.0E-07***	***1.2E-07***	--	--	

SLAP event counts by system (for BWRs)								Cat-1	Cat-2	Cat-3	Cat-4	Cat-5	Cat-6	
Only includes RCPB (S-type)			Est.	Pipe Sizes		# of	greater than	100	1500	5000	25000	100000	500000	gpm
		Deg. Mech.	LOCA	Min	Max	Welds	Total	0.5	1.8	3.3	7.3	18.4	41.2	dia. (in.)
	Event	Fractional	Fraction	(mm)	(mm)	Dom.	LOCA Freq	1	0.3	0.1	0.03	0.01	0.003	probabilty
LOCA System	Counts	Contribution	by System	(in)	(in)	Total	9.4E-05							
UA		0.00				123	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	--	--	
MA		0.00					0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	--	--	
LC		0.00					0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	--	--	
FDR	0	0.00					0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	--	--	
SCC	0	0.00					0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	--	--	
MF	0	0.00					0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	--	--	
TF	7	0.88					3.5E-06	3.5E-06	1.1E-06	3.5E-07	1.1E-07	--	--	
FS	1	0.13					5.0E-07	5.0E-07	1.5E-07	5.0E-08	1.5E-08	--	--	
UK		0.00					0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	--	--	
ECCS	20		0.11	10	500		1.0E-05	1.0E-05	3.0E-06	1.0E-06	3.0E-07	1.0E-07	0.0	
ECCS	20	1		0.4	19.7	1	1.0E-05	1.0E-05	3.0E-06	1.0E-06	3.0E-07	1.0E-07	--	
UA		0.00				60	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	--	
MA		0.00					0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	--	
LC	1	0.05					5.0E-07	5.0E-07	1.5E-07	5.0E-08	1.5E-08	5.0E-09	--	
FDR	0	0.00					0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	--	
SCC	9	0.45					4.5E-06	4.5E-06	1.4E-06	4.5E-07	1.4E-07	4.5E-08	--	
MF	5	0.25					2.5E-06	2.5E-06	7.5E-07	2.5E-07	7.5E-08	2.5E-08	--	
TF	2	0.10					1.0E-06	1.0E-06	3.0E-07	1.0E-07	3.0E-08	1.0E-08	--	
FS	3	0.15					1.5E-06	1.5E-06	4.5E-07	1.5E-07	4.5E-08	1.5E-08	--	
UK	0	0.00					0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	--	
Inst	20		0.11	10	150		1.0E-05	1.0E-05	3.0E-06	1.0E-06	0.0	0.0	0.0	
Inst	20	1		0.4	5.9	1	1.0E-05	1.0E-05	3.0E-06	1.0E-06	--	--	--	
UA		0.00					0.0E+00	0.0E+00	0.0E+00	0.0E+00	--	--	--	
MA		0.00					0.0E+00	0.0E+00	0.0E+00	0.0E+00	--	--	--	
LC		0.00					0.0E+00	0.0E+00	0.0E+00	0.0E+00	--	--	--	
FDR	5	0.25					2.5E-06	2.5E-06	7.5E-07	2.5E-07	--	--	--	
SCC	4	0.20					2.0E-06	2.0E-06	6.0E-07	2.0E-07	--	--	--	
MF	11	0.55					5.5E-06	5.5E-06	1.7E-06	5.5E-07	--	--	--	
TF	0	0.00					0.0E+00	0.0E+00	0.0E+00	0.0E+00	--	--	--	
FS		0.00					0.0E+00	0.0E+00	0.0E+00	0.0E+00	--	--	--	
UK	0	0.00					0.0E+00	0.0E+00	0.0E+00	0.0E+00	--	--	--	
RCIC	2		0.01	20	150		1.0E-06	1.0E-06	3.0E-07	1.0E-07	0.0	0.0	0.0	
RCIC	2	1		0.8	5.9	1	1.0E-06	1.0E-06	3.0E-07	1.0E-07	--	--	--	

SLAP event counts by system (for BWRs) Only includes RCPB (S-type)			Est.	Pipe Sizes		# of	greater than	Cat-1 100	Cat-2 1500	Cat-3 5000	Cat-4 25000	Cat-5 100000	Cat-6 500000	gpm
		Deg. Mech.	LOCA	Min	Max	Welds	Total	0.5	1.8	3.3	7.3	18.4	41.2	dia. (in.)
	Event	Fractional	Fraction	(mm)	(mm)	Dom.	LOCA Freq	1	0.3	0.1	0.03	0.01	0.003	probabilty
LOCA System	Counts	Contribution	by System	(in)	(in)	Total	9.4E-05							
UA		*0.00*				16	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	--	--	
MA		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	--	--	
LC		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	--	--	
FDR		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	--	--	
SCC		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	--	--	
MF	*2*	*1.00*					*1.0E-06*	*1.0E-06*	*3.0E-07*	*1.0E-07*	--	--	--	
TF	*0*	*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	--	--	
FS	*0*	*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	--	--	
UK		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	--	--	
Recirc	48		0.26	20	700		2.4E-05	2.4E-05	7.2E-06	2.4E-06	7.2E-07	2.4E-07	0.0E+00	
Recirc - ave	***48***			0.8	27.6	1	***2.4E-05***	***2.4E-05***	***7.2E-06***	***2.4E-06***	***7.2E-07***	***2.4E-07***	0.0	
Recirc - old (BC)	*45*	***1***				121	***5.3E-05***	***5.3E-05***	***1.6E-05***	***5.3E-06***	***1.6E-06***	***5.3E-07***	--	
UA		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	
MA		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	
LC		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	
FDR	*1*	*0.022*					*1.2E-06*	*1.2E-06*	*3.5E-07*	*1.2E-07*	*3.5E-08*	*1.2E-08*	--	
SCC	*43*	*0.96*					*5.0E-05*	*5.0E-05*	*1.5E-05*	*5.0E-06*	*1.5E-06*	*5.0E-07*	--	
MF	*1*	*0.02*					*1.2E-06*	*1.2E-06*	*3.5E-07*	*1.2E-07*	*3.5E-08*	*1.2E-08*	--	
TF		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	
FS		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	
UK		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	
Recirc - new	*3*	***1***					***2.4E-06***	***2.4E-06***	***7.2E-07***	***2.4E-07***	***7.2E-08***	***2.4E-08***	0.0	
UA		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	
MA		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	
LC		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	
FDR	*2*	*0.67*					*1.6E-06*	*1.6E-06*	*4.8E-07*	*1.6E-07*	*4.8E-08*	*1.6E-08*	--	
SCC	*1*	*0.33*					*8.0E-07*	*8.0E-07*	*2.4E-07*	*8.0E-08*	*2.4E-08*	*8.0E-09*	--	
MF	*0*	*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	
TF		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	
FS		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	
UK		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	

SLAP event counts by system (for BWRs)
Only includes RCPB (S-type)

LOCA System	Event Counts	Deg. Mech. Fractional Contribution	Est. LOCA Fraction by System	Pipe Sizes Min (mm) (in)	Pipe Sizes Max (mm) (in)	# of Welds Dom. Total	greater than Total LOCA Freq 9.4E-05	*Cat-1*	*Cat-2*	*Cat-3*	*Cat-4*	*Cat-5*	*Cat-6*	
							greater than	*100*	*1500*	*5000*	*25000*	*100000*	*500000*	*gpm*
								0.5	*1.8*	*3.3*	*7.3*	*18.4*	*41.2*	*dia. (in.)*
								1	*0.3*	*0.1*	*0.03*	*0.01*	*0.003*	*probabilty*
RHR	39		0.21	19	600		2.0E-05	2.0E-05	5.9E-06	2.0E-06	5.9E-07	2.0E-07	0.0	
RHR	*39*	*1*		0.7	23.6	1	***2.0E-05***	***2.0E-05***	***5.9E-06***	***2.0E-06***	***5.9E-07***	***2.0E-07***	--	
UA		*0.00*				74	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	
MA		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	
LC		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	
FDR	*2*	*0.05*					*1.0E-06*	*1.0E-06*	*3.0E-07*	*1.0E-07*	*3.0E-08*	*1.0E-08*	--	
SCC	*33*	*0.85*					*1.7E-05*	*1.7E-05*	*5.0E-06*	*1.7E-06*	*5.0E-07*	*1.7E-07*	--	
MF	*2*	*0.05*					*1.0E-06*	*1.0E-06*	*3.0E-07*	*1.0E-07*	*3.0E-08*	*1.0E-08*	--	
TF	*2*	*0.05*					*1.0E-06*	*1.0E-06*	*3.0E-07*	*1.0E-07*	*3.0E-08*	*1.0E-08*	--	
FS		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	
UK		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	
RWCU	17		0.09	10	400		8.5E-06	8.5E-06	2.6E-06	8.5E-07	2.6E-07	0.0	0.0	
RWCU	*17*	*1*		0.4	15.7	1	***8.5E-06***	***8.5E-06***	***2.6E-06***	***8.5E-07***	***2.6E-07***	--	--	
UA		*0.00*				72	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	--	
MA		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	--	
LC	*0*	*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	--	
FDR	*1*	*0.06*					*5.0E-07*	*5.0E-07*	*1.5E-07*	*5.0E-08*	*1.5E-08*	--	--	
SCC	*6*	*0.35*					*3.0E-06*	*3.0E-06*	*9.0E-07*	*3.0E-07*	*9.0E-08*	--	--	
MF	*3*	*0.18*					*1.5E-06*	*1.5E-06*	*4.5E-07*	*1.5E-07*	*4.5E-08*	--	--	
TF	*6*	*0.35*					*3.0E-06*	*3.0E-06*	*9.0E-07*	*3.0E-07*	*9.0E-08*	--	--	
FS	*0*	*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	--	
UK	*1*	*0.06*					*5.0E-07*	*5.0E-07*	*1.5E-07*	*5.0E-08*	*1.5E-08*	--	--	
SLC	3		0.02	20	75		1.5E-06	1.5E-06	4.5E-07	0.0	0.0	0.0	0.0	
SLC	3	*1*		0.8	3.0	1	***1.5E-06***	***1.5E-06***	***4.5E-07***	--	--	--	--	
UA		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	--	--	--	--	
MA		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	--	--	--	--	
LC		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	--	--	--	--	
FDR		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	--	--	--	--	
SCC	3	*1.00*					*1.5E-06*	*1.5E-06*	*4.5E-07*	--	--	--	--	
MF		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	--	--	--	--	
TF		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	--	--	--	--	
FS		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	--	--	--	--	
UK		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	--	--	--	--	

SLAP event counts by system (for BWRs) Only includes RCPB (S-type)								Cat-1	Cat-2	Cat-3	Cat-4	Cat-5	Cat-6	
			Est.	Pipe Sizes		# of	greater than	100	1500	5000	25000	100000	500000	gpm
		Deg. Mech.	LOCA	Min	Max	Welds	Total	0.5	1.8	3.3	7.3	18.4	41.2	dia. (in.)
	Event	Fractional	Fraction	(mm)	(mm)	Dom.	LOCA Freq	1	0.3	0.1	0.03	0.01	0.003	probabilty
LOCA System	Counts	Contribution	by System	(in)	(in)	Total	9.4E-05							
SRV Lines	3		0.02	15	25		1.5E-06	1.5E-06	0.0	0.0	0.0	0.0	0.0	
SRV Lines	*3*	*1*		0.6	1.0	1	***1.5E-06***	***1.5E-06***	--	--	--	--	--	
UA		*0.00*					*0.0E+00*	*0.0E+00*	--	--	--	--	--	
MA		*0.00*					*0.0E+00*	*0.0E+00*	--	--	--	--	--	
LC		*0.00*					*0.0E+00*	*0.0E+00*	--	--	--	--	--	
FDR		*0.00*					*0.0E+00*	*0.0E+00*	--	--	--	--	--	
SCC	*2*	*0.67*					*1.0E-06*	*1.0E-06*	--	--	--	--	--	
MF	*1*	*0.33*					*5.0E-07*	*5.0E-07*	--	--	--	--	--	
TF	*0*	*0.00*					*0.0E+00*	*0.0E+00*	--	--	--	--	--	
FS		*0.00*					*0.0E+00*	*0.0E+00*	--	--	--	--	--	
UK		*0.00*					*0.0E+00*	*0.0E+00*	--	--	--	--	--	
Steam Lines	1		0.01	20	100		5.0E-07	5.0E-07	1.5E-07	5.0E-08	0.0	0.0	0.0	
Steam Lines	*1*	*1*		0.8	28.0	1	***5.0E-07***	***5.0E-07***	***1.5E-07***	***5.0E-08***	***1.5E-08***	***5.0E-09***	--	
UA		*0.00*				113.0	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	
MA		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	
LC		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	
FDR		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	
SCC	*0*	*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	
MF	*1*	*1.00*					*5.0E-07*	*5.0E-07*	*1.5E-07*	*5.0E-08*	*1.5E-08*	*5.0E-09*	--	
TF		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	
FS		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	
UK		*0.00*					*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	*0.0E+00*	--	
	188		1.00				9.4E-05	9.4E-05	2.8E-05	9.1E-06	2.4E-06	6.2E-07	0.0E+00	

Table E.7 BWR Passive Non-Pipe LOCA Frequency

BWR Non-Pipe LOCA Contributors											
Total non-pipe event count =			30								
				Non-Pipe	*Cat-1*	*Cat-2*	*Cat-3*	*Cat-4*	*Cat-5*	*Cat-6*	
Non-Pipe				LOCA	*100*	*1500*	*5000*	*25000*	*100000*	*500000*	*gpm*
LOCA	Degradation Mechanism		Non-Pipe	Freq	*0.5*	*1.8*	*3.3*	*7.3*	*18.4*	*41.2*	*dia. (in.)*
System	Counts	Fract Contr	LOCA Fract	9.4E-05	*1*	*0.3*	*0.1*	*0.03*	*0.01*	*0.003*	*probabilty*
LIV	**1**	**1**	**0.03**	**3.15E-06**	**3.15E-06**	**9.44E-07**	**3.15E-07**	**9.44E-08**	**3.15E-08**	**0.00E+00**	
FDR		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00		
FS		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00		
LC		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00		
MA	1	1.00		3.15E-06	3.15E-06	9.44E-07	3.15E-07	9.44E-08	3.15E-08		
MF		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00		
SCC		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00		
TF		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00		
UNK		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00		
RecP	**5**	**1**	**0.17**	**1.57E-05**	**1.57E-05**	**4.72E-06**	**1.57E-06**	**4.72E-07**	**1.57E-07**	**0.00E+00**	
FDR	3	0.60		1.89E-06	1.89E-06	5.66E-07	1.89E-07	5.66E-08	1.89E-08		
FS		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00		
LC		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00		
MA	2	0.40		1.26E-06	1.26E-06	3.78E-07	1.26E-07	3.78E-08	1.26E-08		
MF		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00		
SCC		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00		
TF		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00		
UNK		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00		
RPV	**24**	**1**	**0.80**	**7.55E-05**	**7.55E-05**	**2.27E-05**	**7.55E-06**	**2.27E-06**	**7.55E-07**	**2.27E-07**	
FDR	1	0.04		1.31E-07	1.31E-07	3.93E-08	1.31E-08	3.93E-09	1.31E-09	3.93E-10	
FS		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
LC		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
MA		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
MF		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
SCC	23	0.96		3.02E-06	3.02E-06	9.05E-07	3.02E-07	9.05E-08	3.02E-08	9.05E-09	

BWR Non-Pipe LOCA Contributors

Total non-pipe event count = 30

Non-Pipe LOCA System	Degradation Mechanism Counts	Fract Contr	Non-Pipe LOCA Fract	Non-Pipe LOCA Freq	*Cat-1*	*Cat-2*	*Cat-3*	*Cat-4*	*Cat-5*	*Cat-6*	
					100	*1500*	*5000*	*25000*	*100000*	*500000*	*gpm*
					0.5	*1.8*	*3.3*	*7.3*	*18.4*	*41.2*	*dia. (in.)*
				9.4E-05	*1*	*0.3*	*0.1*	*0.03*	*0.01*	*0.003*	*probabilty*
TF		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
UNK		0.00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	

APPENDIX F

PIPING BASE CASE RESULTS OF DAVID HARRIS

APPENDIX F

PIPING BASE CAES RESULTS OF DAVID HARRIS

Probabilistic Fracture Mechanics Analyses Performed in Support of LOCA Frequency Re-evaluation Effort

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F.1 Introduction

The purpose of this document is to report the procedures used and the results obtained in probabilistic fracture mechanics analyses of the base case systems considered in the LOCA Re-evaluation effort performed by use of expert elicitation by the Nuclear Regulatory Commission in the period February 2003 – March 2004. The base case systems, which were defined in the kick-off meeting of the expert panel that was held in Rockville, Maryland in February 2003, consisted of the following:

Pressurized Water Reactor

- hot leg (cast austenitic stainless steel)
- surge line (austenitic stainless steel)
- HPI makeup nozzle (austenitic stainless steel)

Boiling Water Reactor

- recirculation line (austenitic stainless steel)
- feedwater (carbon steel)

These were identified as key systems that could serve as benchmarks for use by members of the expert panel in their estimation of LOCA frequencies.

Piping isometrics of the base case systems and other systems identified in the kick-off meeting as important to estimations of flow rate probabilities were included in the FTP site that was set up for the use of panel members. Times in this appendix are in reactor-years (1 calendar year ~ 0.8 reactor years).

F.2 Software

The following discussion provides only a brief review of the PRAISE software. The references cited give the details. The results reported here were generated by use of the PRAISE software, which was developed with NRC support over a period of some 20 years. PRAISE is based on deterministic fracture mechanics, with some of the inputs considered as random variables. This allows the statistical distribution of lifetime to be computed, rather than a single deterministic failure time. The probability of failure (leaks of various sizes) is obtained from the computed lifetime distribution.

Several versions of PRAISE were employed, depending on the nature of the problem. The original version of PRAISE [F.1] considers fatigue crack growth from crack-like weld defects introduced during fabrication. Semi-elliptical interior surface cracks are considered, usually circumferentially oriented. The initial crack size and fatigue crack growth properties are the major random variables, and Monte Carlo simulation is used to generate numerical results. Stratified sampling of crack depth and aspect ratio is employed to allow very small probabilities to be obtained without excessive computer time. Figure F.1 is a schematic representation of the probabilistic fracture mechanics procedures used in the original version of PRAISE. The cumulative probability of a flow (leak) rate exceeding a specified size is generated by PRAISE as a function of time. If the stress history is specified in reactor-years, then the PRAISE results are also in reactor years.

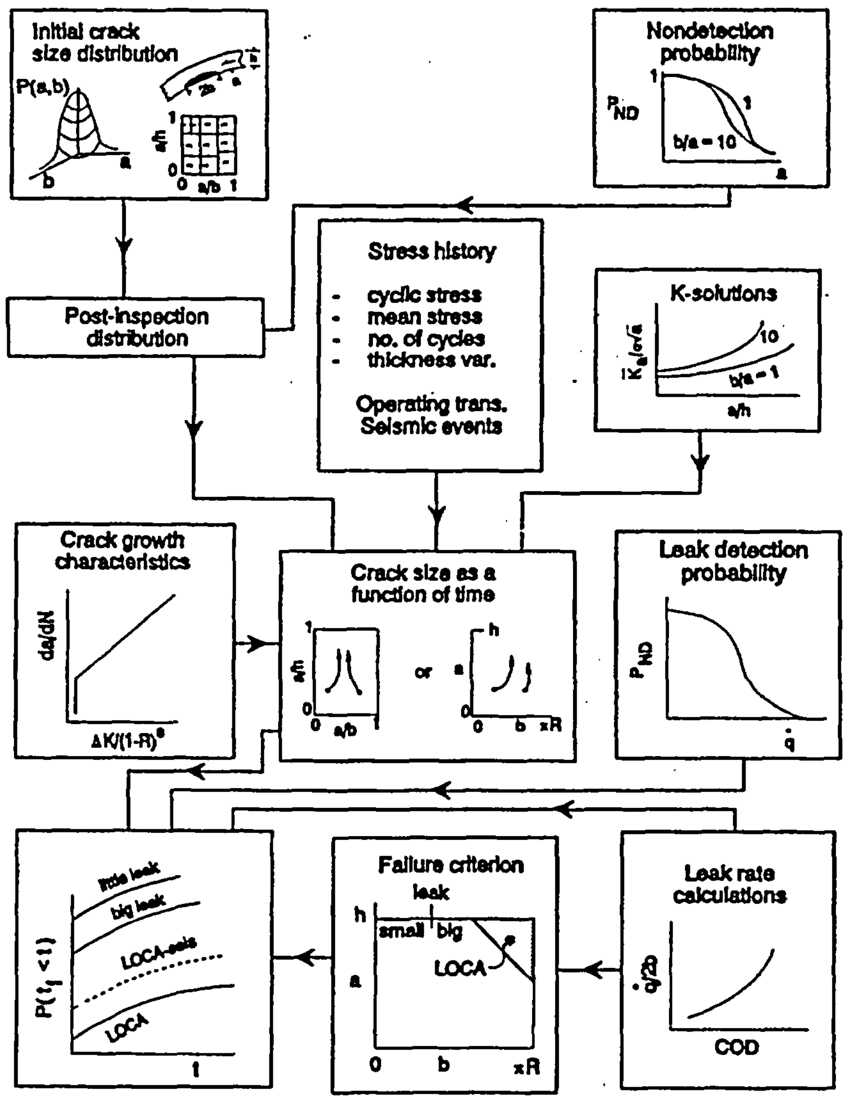

Figure F.1 Overview of PRAISE Methodology for Probabilistic Analysis of Fatigue Crack Growth

A later version of PRAISE was used the analyze initiation and growth of stress corrosion cracks [F.2]. Both the fatigue crack growth and stress corrosion initiation and growth capabilities are included in pcPRAISE [F.3] and also in WinPRAISE [F.4], which is a Windows version that is much easier to use than the earlier DOS versions. WinPRAISE gives the same results as pcPRAISE for the same problem with the same inputs.

In some cases, the cyclic stresses were such that fatigue crack initiation is the expected dominant degradation mechanism, and it was necessary to use a later version of PRAISE. This version was developed and used in Reference F.5, with additional capabilities described in Reference F.6 (such as the ability to consider detailed circumferential variations of the stresses). The fatigue crack initiation analyses employed probabilistic strain-life relations developed by Argonne National Laboratory [F.7, F.8]. Once a fatigue crack initiates, the original growth analysis capabilities in PRAISE are used for the crack growth portion of the lifetime. The depth of an initiated fatigue crack is taken to be 3.0 mm (0.12 inches), in accordance with the ANL correlations, with a random surface length. In some instances, modifications to the source code of PRAISE were made to provide results of particular use for the problem at hand. These instances are discussed in the specific base case problem where they were employed.

In cases where inspection was considered, the nondetection probability was represented by the expression

$$P_{ND}(a) = \varepsilon + \frac{1}{2}(1+\varepsilon)\,\mathrm{erfc}\left[\upsilon \ln \frac{a}{a^*}\right] \qquad \text{[F.1]}$$

In this expression, ε is the probability of not detecting a crack no matter how deep it is, a^* is the crack depth having about a 50% chance of being detected, and ν controls the slope of the P_{ND} curve. "Good" and "outstanding" detection capabilities were considered [F.9], with the parameters given in Table F.1.

Table F.1 Parameters in Non-Detection Relation

		Ferritic	Austenitic	
		Fatigue Cracks	Fatigue Cracks	SCC Cracks
outstanding				
	a^*	0.05h	0.05h	0.05h
	ν	1.6	1.6	1.6
	ε	0.005	0.005	0.005
good				
	a^*	0.15h	0.15h	0.4h
	ν	1.6	1.6	1.6
	ε	0.02	0.02	0.10

The operating and hydro pressures of Table F.2 were considered.

Table F.2 Operating and Hydro Pressures, psi

	Operating Pressure	Hydro Pressure
PWR	2250	3125
BWR	1250	1560

A leak detection capability of 19 lpm (5 gpm) was considered, which means that any through-wall crack with a leak rate greater than 19 lpm (5 gpm) was immediately detected and removed from service.

F.3 Post-Processing of PRAISE Results

PRAISE analyses are performed on a piping location-by-location basis to provide the cumulative failure probability as a function of time, whereas the desired end result is the system failure frequency within various time frames. The time increments of interest are now (0-25 years), the near future (25-40 years) and the more distant future (40-60 years). If the reactor transients analyzed are per reactor-year (rather than calendar year), then the times considered are also per reactor-year. As an average, one calendar year corresponds to 0.8 reactor years, but no adjustments for this are made in this appendix. The system failure frequency is obtained from the failure frequency for individual locations by analyzing the most highly stressed location and multiplying by the number of such highly stressed locations in the system. The failure frequency for a flow rate exceeding $\dot{q}$ for a given time increment from t_1 to t_2 is obtained from the following relation

$$p_{\dot{q}}(t_2) = \frac{P_{\dot{q}}(t_2) - P_{\dot{q}}(t_1)}{t_2 - t_1} \qquad \text{[F.2]}$$

The values of $P_{\dot{q}}(t)$ are the output from PRAISE for the dominant location(s) in the system. The failure frequency for the system is then obtained by multiplying by the number of locations in the system that have the high stresses of the dominant location.

F.3.1 Material Properties

Numerous material properties enter into a PRAISE analysis, many of which are described in the references cited above. A Ramberg-Osgood stress-strain curve is used in the computation of the applied value of the J-integral, as represented by the following relation

$$\varepsilon = \frac{\sigma}{E} + \left(\frac{\sigma}{D}\right)^n \qquad \text{[F.3]}$$

Crack instability is governed by exceedance of a critical net section stress and/or a J-integral based tearing instability using a bilinear tearing resistance curve.

A set of default material tensile and fracture properties are provided in WinPRAISE [F.4], which are summarized in Table F.3. Unless otherwise stated, these properties are used in the base case analyses.

Table F.3 Summary of Default Material Properties

Material		Low Alloy	Carbon	Type 304	Type 316
Flow Stress (normal)					
	Mean, ksi	50.	52.	44.9	44.9
	Standard Deviation	2.1	2.2	1.9	1.9
Tearing Instability Data					
	D, ksi	146	154	106	106
	n	5	4	5	5
	J_{Ic} kips/in	0.6	1.5	5	5
	T_{mat}	60	45	300	300
	E, ksi	27300	27300	25800	25800
	v	0.3	0.3	0.3	0.3
	Yield Strength, ksi	30.8	28.3	19.4	19.4
	Tensile Strength ksi	70.0	60.0	59.3	61.6
Fatigue Crack Growth Properties - (random)		based on ASME Code		see Reference F.1 or F.4	

The properties of Table F.3 are generally somewhat conservative and are representative of undegraded materials. In some instances, degraded material properties are considered, as discussed at the particular component involved.

In the case of fatigue of initial cracks, the distributions of the initial crack depth and aspect ratio are probably the most important random variables. Unless otherwise stated, the depth distribution is taken from Reference F.10, which is the default in WinPRAISE and is also included in Reference F.9.

F.3.2 Hot Leg

The main coolant piping system is one of the base case systems. The failure probability of the large piping of this system is dominated by the hot leg to pressure vessel weld, because this location is at the highest temperature and sees the highest stress.

F.3.2.1 Dimensions and Welds - From the piping isometrics made available to the panel members, the hot leg has a 29 inch inner diameter, and a thickness of 63.5 mm (2.5 inch) (OD=34 inches), fabricated from SA-376 (which is an austenitic stainless steel). The example plant has two coolant loops. There are several welds in the hot leg, including shop and field welds and safe ends. There is a safe end and field weld at the pressure vessel.

F.3.2.2 Stresses and Cycles - Table 1-2, page 9 of Reference F.1 summarizes the deadweight, pressure and restraint of thermal expansion stresses for the 14 field welds in one loop of the large primary piping in the plant considered in that report. Joint 1 is the hot leg to pressure vessel joint, and it has the highest stresses. The seismic stresses are included in Table 1-3, page 10 [F.1]. They are generally quite low. The postulated list of transients is provided in Table 4-1, page 152 [F.1], and is the transients occurring over the 40 year design life, which corresponds to reactor years. The list contains 11 types of transients. There is sufficient information in Reference F.1 to consider all of these transients, but only the heat-up cool-down transient will be considered in this case, because it is the dominant transient contributing to fatigue crack growth [F.1]. The heat-up cool-down transient was postulated to occur 200 times in 40 years (5/year). This is excessive, and 3/year is used herein.

The seismic stresses are given in terms of the maximum load controlled stress (deadweight + pressure + max seismic), and the summary of the stress history needed for fatigue crack growth analysis is also provided. This summary, denoted as S, is the sum of the cyclic stresses as follows

$$S^4 = \sum_{seismicstresshistory} \sigma_{\max}^2 \Delta\sigma^2 \qquad [F.4]$$

This is what controls the amount of fatigue crack growth during a seismic event for a fourth power crack growth law that includes R-ratio effects (See Reference F.1). Table F.4 summarizes the suggested stress history for the hot leg to pressure vessel joint.

Table F.4 Summary of Stress History for Hot Leg to Pressure Vessel Joint

deadweight stress = 2.08 ksi
pressure stress = 6.49 ksi (axial)
restraint of thermal expansion stress = 6.50 ksi
3 times per year

	max σ_{LC} ksi	S^4 ksi^4	$\Delta\sigma$, ksi
OBE	8.76	521.6	1.27
SSE	9.06	2958.3	1.96
3SSE	10.26	63430	4.22
5SSE	10.62	162000	5.33

The right-hand column in the above table is the cyclic stress if the seismic event contains 200 stress cycles all of the same amplitude, with a low minimum load. This column is derived from the value of S^4 and Equation F.4, with $\sigma_{max}=\Delta\sigma$, and is included just to provide an idea of the size of the seismic stresses. They are not large.

Residual stresses, when considered, are taken to be the default values for large lines, as reported in Reference F.2.

F.3.2.3 Results - WinPRAISE runs were made for the hot leg to pressure vessel weld using the above stresses and default material properties. Table F.5 summarizes the results. "Good" inspections at 0, 20 and 40 years were considered. These results are the cumulative leak probability. The left hand column gives the leak rate in gallons per minute. The next column in gives the time (25, 40 and 60 years), and the probabilities are directly from the PRAISE output for these times. For the Monte Carlo simulation, the crack size plane ($a/h - a/b$) was divided into 20 by 20 strata, with a maximum of 2000 trials drawn from each stratum. Sampling from a given stratum was stopped when 20 failures occurred in that stratum. Sampling began at the corner of the a/h-a/b plane corresponding to long deep cracks (1,0), and continued to shorter, then shallower cracks until no failures occurred in a stratum within 2000 trials. Sampling was then stopped. This procedure is referred to as automated stratification, and is a feature unique to WinPRAISE [F.4]. Earlier versions of the PRAISE software require the user to define each stratum and the sampling from each.

Table F.5 Cumulative Probability PRAISE Results for Hot Leg-Pressure Vessel Weld for Fatigue Crack Growth from Pre-Existing Defects

OD=34.0 inches, h=2.50 inches, cast austenitic, no σ_{DL}, times in reactor years

		Base			No hydro			Aging		
Hydro		yes			no			no		
Insp		good			good			good		
t_{insp}		0,20,40			0,20,40			0,20,40		
Aging		no			no			yes		
J_{IC}		5			5			1.5		
dJ/da		23.44			23.44			15		
		no EQ	SSE	5SSE	no EQ	SSE	5SSE	no EQ	SSE	5SSE
>0	25	1.20x10^{-18}	2.34x10^{-16}	2.45x10^{-16}	6.61x10^{-15}	7.04x10^{-15}	7.08x10^{-15}	1.43x10^{-14}	1.47x10^{-14}	1.47x10^{-14}
	40	1.29x10^{-18}	2.35x10^{-16}	2.45x10^{-16}	6.61x10^{-15}	7.04x10^{-15}	7.08x10^{-15}	1.43x10^{-14}	1.47x10^{-14}	1.47x10^{-14}
	60	1.29x10^{-18}	6.01x10^{-18}	6.19x10^{-18}	6.61x10^{-15}	6.62x10^{-15}	6.62x10^{-15}	1.43x10^{-14}	1.44x10^{-14}	1.44x10^{-14}
		HLA0			HLB0			HLC0		
>100	25	2.44x10^{-19}	2.53x10^{-18}	3.89x10^{-18}	2.93x10^{-17}	3.18x10^{-17}	3.22x10^{-17}	5.11x10^{-17}	5.52x10^{-17}	5.63x10^{-17}
	40	2.55x10^{-19}	2.60x10^{-18}	3.97x10^{-18}	2.94x10^{-17}	3.19x10^{-17}	3.22x10^{-17}	5.11x10^{-17}	5.53x10^{-17}	5.63x10^{-17}
	60	2.56x10^{-19}	3.04x10^{-19}	3.33x10^{-18}	2.94x10^{-17}	2.94x10^{-17}	2.94710^{-17}	2.12x10^{-17}	5.12x10^{-17}	5.13x10^{-17}
		HLA1			HLB1			HLC1		
>1500	25	1.20x10^{-20}	6.48x10^{-19}	1.31x10^{-18}	1.31x10^{-18}	1.99x10^{-18}	2.64x10^{-18}	2.72x10^{-18}	4.33x10^{-18}	5.99x10^{-18}
	40	1.26x10^{-20}	6.62x10^{-19}	1.32x10^{-18}	1.31x10^{-18}	2.00x10^{-18}	2.65x10^{-18}	2.72x10^{-18}	4.36x10^{-18}	5.97x10^{-18}
	60	1.27x10^{-20}	2.61x10^{-20}	3.93x10^{-20}	1.31x10^{-18}	1.33x10^{-18}	1.34x10^{-18}	2.72x10^{-18}	2.75x10^{-18}	2.79x10^{-18}
		HLA2			HLB2			HLC2		
>5000	25	1.19x10^{-20}	6.48x10^{-19}	1.31x10^{-18}	1.31x10^{-18}	1.99x10^{-18}	2.64x10^{-18}	2.72x10^{-18}	4.33x10^{-18}	5.99x10^{-18}
	40	1.26x10^{-20}	6.62x10^{-19}	1.32x10^{-18}	1.31x10^{-18}	2.00x10^{-18}	2.65x10^{-18}	2.72x10^{-18}	4.36x10^{-18}	5.97x10^{-18}
	60	1.27x10^{-20}	2.61x10^{-20}	3.93x10^{-20}	1.31x10^{-18}	1.33x10^{-18}	1.34x10^{-18}	2.72x10^{-18}	2.75x10^{-18}	2.79x10^{-18}
		HLA3			HLB3			HLC3		
>500000	25	1.20x10^{-20}	6.48x10^{-19}	1.31x10^{-18}	1.31x10^{-18}	1.99x10^{-18}	2.64x10^{-18}	2.72x10^{-18}	4.33x10^{-18}	5.99x10^{-18}
	40	1.26x10^{-20}	6.62x10^{-19}	1.32x10^{-18}	1.31x10^{-18}	2.00x10^{-18}	2.65x10^{-18}	2.72x10^{-18}	4.36x10^{-18}	5.97x10^{-18}
	60	1.27x10^{-20}	2.61x10^{-20}	3.93x10^{-20}	1.31x10^{-18}	1.33x10^{-18}	1.34x10^{-18}	2.72x10^{-18}	2.75x10^{-18}	2.79x10^{-18}
		HLA4			HLB4			HLC4		

noticeable effect of hydro
noticeable effect of seismic when hydro test is performed, less effect when no hydro
aging has about x2 effect
>1500 gpm same as DEPB

Runs were made with and without a hydro test, and it is seen that hydro testing has a noticeable effect. Moderate material degradation is considered, with the values of J_{Ic} and $(dJ/da)_{matl}$ identified in the table. The failure probabilities are all very small, even the leak probabilities. The influence or seismic events is seen to be quite small.

Table F.5 provides the base case results for the hot leg. Additional runs were made to study the following variables:

- The effects of applying a load-controlled overload stress at a specified time were studied. This is called a design-limiting stress, and represents an overload event, such as water hammer or a seismic event even larger than the 5 SSE already considered for this component.

- The effects of the fatigue crack growth relation employed were studied. The fatigue crack growth relation in PRAISE for austenitic stainless steel is based on information available during the original software development. More recent crack growth relations have been suggested [F.11]. For the simple stress history in this case, it is possible to run PRAISE with a crack growth relation that is equivalent to the more recent relation.

- PWSCC crack initiation and growth has been identified in the control drive mechanisms (CRDM) in PWRs. This occurs in the Alloy 600 weldment. This alloy is also used in the safe end of the pressure vessel to main coolant piping welds, so is present in the hot leg to pressure vessel weld under consideration. In order to model the initiation and growth of PWSCC cracks, the initiation kinetics were assumed to be the same as for Type 316NG stainless steel as currently in PRAISE [F.2], but the crack growth kinetics were changed to be representative of Alloy 600. Based on information in Reference F.12, the crack growth kinetics is represented by the relation

$$\frac{da}{dt} = CK^m \qquad \text{[F.5]}$$

where m equals 1.16, and C is lognormally distributed with a median value that depends on the temperature and material (weld, base metal, etc.). The median value of C for a weld at 315 C (600°F) is 7.86×10^{-7}, when crack growth rates are in inches/hour and K is in ksi-in$^{1/2}$. Combining the within-heat and heat-to-heat variation in C, the second parameter of the lognormal distribution is 1.193 (standard deviation of $\ln C = 1.193$).

- The effects of more severe material degradation were studied, with the values of the degraded toughness given along with the results. Since PRAISE can not consider time-dependent material properties, the degraded material properties are present even in new pipe. The values of the degraded properties are from Reference F.13.

The results of these additional runs are summarized in Tables F.6 and F.7.

Table F.6 Cumulative PRAISE Results Additional Runs for Hot Leg Pressure Vessel Weld

		From Table F.5	Ref F.11 da/dN	σ_{DL} @ t-1	PWSCC Growth no σ_{res}	PWSCC Growth σ_{res}	PWSCC Initiation σ_{res}
Hydro		yes	yes	yes	yes	yes	--
Insp		good	good	good	good	good	good
t_{insp}		0,20,40	0,20,40	0,20,40	0,20,40	0,20,40	20,40
Aging		no	no	no	no	no	no
J_{IC}		5	5	5	5	5	5
dJ/da		23.44	23.44	23.44	23.44	23.44	23.44
		no EQ	no EQ	no EQ	no EQ	no EQ	no EQ
>0	25	1.20x10^{-18}	2.20x10^{-18}	2.38x10^{-16}	0.923	0.916	0.001
	40	1.29x10^{-18}	2.42x10^{-18}	--	0.926	0.918	0.020
	60	1.29x10^{-18}	2.43x10^{-18}	--	0.926	0.919	0.068
			HLD0	HLE0			
>100	25	2.44x10^{-19}	2.61x10^{-19}	2.38x10^{-16}	7.97x10^{-7}	2.16x10^{-7}	1.0x10^{-5}
	40	2.55x10^{-19}	2.71x10^{-19}	--	7.97x10^{-7}	2.16x10^{-7}	2.69x10^{-4}
	60	2.56x10^{-19}	2.72x10^{-19}	--	7.97x10^{-7}	2.16x10^{-7}	1.78x10^{-3}
			HLD1	HLE1			
>1500	25	1.20x10^{-20}	1.63x10^{-20}	6.41x10^{-20}	9.68x10^{-10}	2.78x10^{-11}	$<10^{-4}$
	40	1.26x10^{-20}	1.65x10^{-20}	--	9.68x10^{-10}	2.78x10^{-11}	1.0x10^{-4}
	60	1.27x10^{-20}	1.66x10^{-20}	--	9.68x10^{-10}	2.78x10^{-11}	4.85x10^{-4}
			HLD2	HLE2			
>5000	25	1.20x10^{-20}	1.63x10^{-20}	6.41x10^{-20}	2.78x10^{-11}	4.66x10^{-11}	$<10^{-5}$
	40	1.26x10^{-20}	1.65x10^{-20}	--	2.78x10^{-11}	4.66x10^{-11}	9.0x10^{-5}
	60	1.27x10^{-20}	1.66x10^{-20}	--	2.78x10^{-11}	4.66x10^{-11}	3.77x10^{-4}
			HLD3	HLE3			
break	25	1.20x10^{-20}	1.63x10^{-20}	6.41x10^{-20}	2.19x10^{-14}	2.59x10^{-13}	$<10^{-5}$
	40	1.26x10^{-20}	1.65x10^{-20}	--	2.19x10^{-14}	2.59x10^{-13}	9.0x10^{-5}
	60	1.27x10^{-20}	1.66x10^{-20}	--	2.19x10^{-14}	2.59x10^{-13}	3.77x10^{-4}
			HLD4	HLE4	DEPB	DEPB	

The design limiting stress was 40.0 MPa (4.49 ksi). In the case of PWSCC growth, initial fabrication defects were considered with the default depth distribution discussed above. Both initiation and growth were considered for the column identified as PWSCC initiation. The higher large leak rates for the initiation relative to the PWSCC growth are due to the possibility of multiple initiation sites, whereas the growth considers only one initial crack.

Table F.7 Additional Hot Leg Pressure Vessel Runs Considering Material Aging

OD=34 inches	Good Inspection at 0, 20 ,40	Updated da/dN
t=2.50 inches	3 HU-CD per year	No Hydro Unless Specified
σ_{dw}=2.08 ksi		Type 304 Stainless
σ_{te}=6.50 ksi	Degraded Properties Used for All Times	

		Base	no Hydro	A	B	C	D	E
J_{Ic} kips/in		5		1.11	0.67	1.72	0.75	0.20
dJ/da ksi		23.44		13.4	8.0	22.6	6.5	0.05
σ_{ys} ksi		19.4		29.2				
σ_{ult} ksi		--		76.7				
σ_{flo} ksi		44.9		53.0				
D ksi		106		104.5				
N		5		4.84				
>0	25	$1.20x10^{-18}$	$6.61x10^{-15}$	$1.34x10^{-14}$	$1.96x10^{-14}$	$9.73x10^{-15}$	$2.07x10^{-14}$	$4.02x10^{-13}$
	40	$1.29x10^{-18}$	$6.61x10^{-15}$	$1.34x10^{-18}$	$1.96x10^{-14}$	$9.73x10^{-15}$	$2.07x10^{-14}$	$4.02x10^{-13}$
	60	$1.29x10^{-18}$	$6.61x10^{-15}$	$1.34x10^{-18}$	$1.96x10^{-14}$	$9.73x10^{-15}$	$2.07x10^{-14}$	$4.02x10^{-13}$
>100	25	$2.44x10^{-19}$	$2.93x10^{-17}$	$5.26x10^{-18}$	$6.76x10^{-17}$	--	--	$2.81x10^{-14}$
	40	$2.55x10^{-19}$	$2.94x10^{-17}$	$5.27x10^{-18}$	$6.77x10^{-17}$	--	--	$2.81x10^{-14}$
	60	$2.56x10^{-19}$	$2.94x10^{-17}$	$5.27x10^{-18}$	$6.77x10^{-17}$	--	--	$2.81x10^{-14}$
break	25	$1.20x10^{-20}$	$1.31x10^{-18}$	$2.67x10^{-18}$	$4.30x10^{-18}$	$1.48x10^{-18}$	$5.31x10^{-18}$	$2.81x10^{-14}$
	40	$1.26x10^{-20}$	$1.31x10^{-18}$	$2.67x10^{-18}$	$4.31x10^{-18}$	$1.48x10^{-18}$	$5.31x10^{-18}$	$2.81x10^{-14}$
	60	$1.27x10^{-20}$	$1.31x10^{-18}$	$2.67x10^{-18}$	$4.31x10^{-18}$	$1.48x10^{-18}$	$5.31x10^{-18}$	$2.81x10^{-14}$
		earlier base case, default WinPRAISE properties, with hydro	no hydro	unaged weld metal J-T CF8M tensile	mult J-T by 0.6	all CF8M	more sensitive aged	extremely sensitive aged

The biggest effect in Table F.7 is not having a hydro test. This assumption is necessary, because when degraded material properties are used, everything that fails does so during the hydro test.

"Extremely sensitive aged" material properties are needed before degradation has a large effect.

F.3.3 Surge Line

The surge line is one of the base case systems.

F.3.3.1 Dimensions and Welds - From the piping isometric available to the panel members, the surge line is a 14 inch line (14 inch outer diameter) with a thickness of 35.7 mm (1.406 inches). The material is SA376 Type 304, which is an austenitic stainless steel. There are some 13 welds in the line.

F.3.3.2 Stresses and Cycles - The stresses at the surge line elbow are provided in Reference F.5, which is evidently the highest stressed location in the line. These stresses include seismic events and are given in Table F.8. The stress amplitude is contained in this table, which is one-half the stress range (peak-to-peak value).

Table F.8 Summary of Stress Cycles for Surge Line Elbow (Stress Amplitudes with Seismic Stresses)

Load Pair	Amplitude (ksi)	Number/ 40 yr	Load Pair	Amplitude (ksi)	Number/ 40 yr
HYDRO-EXTREME	190.17	6	9D-LEAK TEST	52.20	50
8A-OBE	163.18	14	8G-LEAK TEST	52.20	65
9B-OBE	162.06	14	8G-UPSET3	51.00	30
8B-HYDRO	138.05	4	8G-12	50.96	90
8B-OBE	127.94	10	8G-16	50.93	90
9A-OBE	127.04	14	8E-8G	50.92	13
8C-OBE	64.76	68	8E-OBE	43.38	77
9F-OBE	64.17	68	9H-OBE	42.79	500
8F-18	63.40	68	8H-13	39.82	90
9C-11	63.38	68	8H-OBE	37.43	203
8D-OBE	54.02	72	8H-UPSET4	35.42	40
9G-OBE	53.42	400	8H-9E	33.94	90
8G-18	52.38	22	2A-8H	33.94	77
9D-11	52.35	22	3A-10A	33.10	4120
8G-17	52.35	90	6-10A	33.10	200
9D-LEAK TEST	52.20	50	3B-10A	33.10	4120
8G-LEAK TEST	52.20	65	7-10A	33.10	4580
8G-UPSET3	51.00	30	2B-SLUG1	32.87	100
8G-12	50.96	90	2B-SLUG2	32.87	500
			4B-10A	29.90	17040

To estimate the influence of seismic events, it is necessary to also have the stress history without such events. It is not possible to remove seismic events knowing only the information in the above table. This information was provided in Reference F.14 and is summarized in Table F.9.

Table F.9 Summary of Stress Cycles for Surge Line Elbow
(Stress Amplitudes without Seismic Stresses)

Load Pair	Amplitude (ksi)	Number/ 40 yr	Load Pair	Amplitude (ksi)	Number/ 40 yr
HYDRO-EXTREME	190.17	6	8G-16	50.93	90
9B-HYDRO	149.86	4	8G-9H	50.92	128
8A-UPSET 4	140.42	14	2A-8E	40.10	90
9B-UPSET4	139.43	10	8H-9H	40.09	100
8B-UPSET4	105.89	14	9H-10A	40.09	272
9A-UPSET4	105.13	2	9E-13	39.82	90
9A-LEAK	103.86	12	3A-10A	33.10	4120
8F-18	63.40	68	6-10A	33.10	200
9C-11	63.38	68	3B-10A	33.10	4120
9F-LEAK	63.37	68	7-10A	33.10	4580
8C-LEAK	63.37	35	2B-SLUG1	32.87	100
2A-8C	62.30	33	2B-SLUG2	32.87	500
8G-18	52.38	22	5-10A	29.90	9400
8G-17	52.35	90	4A-10A	29.90	17040
9D-11	52.35	22	4B-10A	29.90	17040
2A-8D	51.20	72	2B-10A	20.60	14400
8H-9G	51.18	400	2A-10A	20.60	14805
8G-UPSET3	51.00	30	10A-UPSET1	20.59	70
9D-12	50.96	50	10A-UPSET5	20.59	30
8G-12	50.96	40	10A-UPSET6	20.59	5
			10A-UPSET2	20.59	95
			1B-10A	20.59	1533
			1B-10B	20.00	87710

The cyclic stress amplitudes of Tables F.8 and F.9 provide the information for the initiation analysis, but additional information is required for the growth portion of the analysis. The spatial gradient (primarily radial) is required. Also, when analyzing the stability of a through-wall crack, the steady normal operating stress is needed. This stress is considered to be the sum of the pressure, deadweight and restraint of thermal expansion stresses. The values of these latter two are given in Reference F.5 as

$$\sigma_{dw} = 0 \qquad \sigma_{te} = 102.6 \text{ MP (14.88 ksi)}.$$

Many of the high stress contributors in Tables 8 and 9 are from rapid excursions of the coolant temperature. The largest stress amplitude (half the peak-to-peak) is 1.31 MPa (190 ksi), so the stresses are large (but localized). These are the stresses at the peak stress location, which is not at weld. The spatial stress gradients (both along the surface and into the pipe wall) are required for a thorough analysis. The radial gradient (into the pipe wall) can be estimated by the procedure given in Section 5.3 of Reference F.5, i.e.,

The following specific rules were applied to assign stress to the uniform and gradient categories:

- Cyclic stresses associated with seismic loads were treated as 100 percent uniform stress.
- Cyclic stresses greater than 310 MPa (45 ksi) were treated as having a uniform component of 310 MPa (45 ksi), and the remainder were assigned to the gradient category.
- For those transients with more than 1000 cycles over a 40 year life, it was assumed that 50% of the stress was uniform stress and 50% a through-wall gradient stress. In addition, for these transients, the uniform stress component was not permitted to exceed 69 MPa (10 ksi).

The gradient stress mentioned above is assumed to vary through the thickness as

$$\sigma(\xi) = \sigma_o\left(1 - 3\xi + \frac{3}{2}\xi^2\right) \qquad [F.6]$$

In this equation, σ_o is the stress at the inner wall of the pipe, $\xi = x/h$, x is the distance into the pipe wall from the inner surface, and h is the wall thickness. The stresses and cycles are high enough that fatigue crack initiation is important, which has been considered in Reference F.5, which shows a probability of 0.981 of a leak in 40 years for this component. The LOCA probabilities will be less. The use of the gradient along the surface will reduce this.

A refined stress analysis was available as part of the efforts reported in Reference F.6. These stresses included details of the variation of the stress in the circumferential direction, and are referred to as the "refined stresses".

F.3.3.3 Results - PRAISE runs were made using the versions that can treat fatigue crack initiation. No inspections were considered. Since crack initiation is considered, there will be no effect of a pre-service inspection. The results are summarized in Table F.10.

Table F.10 Cumulative PRAISE Results for the Surge Line Elbow

Condition		Ref. F.5	Table F.8 Stresses	Table F.9 Stresses	Refined Stresses
Seismic		yes	yes	no	yes
σ_{DL}		no	no	no	no
>0	25		0.372	0.233	
	40	0.982	0.772	0.587	$8x10^{-7}$
	60	0.998	0.968	0.882	$3.3x10^{-5}$
					CENC4H1
>100	25	--	$1.6x10^{-5}$	$7.5x10^{-6}$	$<10^{-7}$
	40	--	$3.11x10^{-4}$	$7.1x10^{-5}$	$<10^{-7}$
	60		$1.33x10^{-3}$	$2.51x10^{-4}$	$<10^{-7}$
		--	CENC4D01	CENC4A3	20 hrs
>1500	25	--	$<10^{-7}$	$<10^{-7}$	
	40	--	$<10^{-7}$	$<10^{-7}$	
	60	--	$2.0x10^{-7}$	$1.0x10^{-7}$	
			CENC4D15	CENC4A4	
			axisymmetric seismic	axisymmetric nonseismic	strain rates and bivariate stresses

It is seen that the seismic stresses do not have a large effect, roughly a factor of 3. The use of the refined stresses greatly reduces the calculated failure probabilities. The computer run for 380 lpm (100 gpm) took about 20 hours and resulted in no failures in 10^7 trials. The runs for > 5,700 lpm (1,500 gpm) with the stresses from Tables F.8 and F.9 had 2 and 1 failures in 10^7 trials, respectively, and these runs each took many hours. Hence, it is evident that the Monte Carlo simulation with multiple fatigue crack initiation sites does not allow definition of the small probabilities of large leaks in the surge line elbow, and an alternate procedure was developed. Stratified sampling is not used for fatigue crack initiation.

F.3.3.4 Alternate Procedure - In cases where the dominant degradation mechanism is fatigue crack initiation with subsequent growth, PRAISE currently has no way of generating low probability results other than conventional Monte Carlo simulation. This is the dominant mechanism for three of the base line components; the surge line elbow, the HPI make up nozzle and the BWR feedwater line elbow. Excessive computer time is needed to generate probabilities of various size leaks for these components, with some runs taking 4 days on a 3 GHz pc, with no leaks of even 380 lpm (100 gpm). An alternate procedure is needed to estimate leak probabilities for the large leaks of interest, and such a procedure is described below.

As part of a standard analysis, the PRAISE software computes the crack opening area and leak rate as functions of the length of through-wall cracks. Hence, this information is readily available, and can be used to determine the length of a through-wall crack needed to produce a given leak rate, such as 380 lpm (100 gpm), 5,700 lpm (1,500 gpm), etc. The probability of having a leak of a given magnitude is then the probability of having a through-wall crack exceeding that length. The half-crack length, b, is considered, which is a function of the desired leak rate, $\dot{q}$. Hence, $b(\dot{q})$ can be considered as known.

The probability of a double-ended-pipe-break (DEPB) is also of interest. In the cases of interest here, the critical net section stress failure criterion is used. For a through-wall crack, the value of b for a DEPB is given by the expression

$$\frac{b_{DEPB}}{\pi R_I} = 1 - \frac{\sigma_{LC}}{\sigma_{flo}} \qquad [F.7]$$

where R_I is the inside radius, σ_{flo} is the flow stress (average of yield and ultimate) and σ_{LC} is the load controlled stress, which is equal to the pressure plus deadweight stress.

The version of PRAISE that performs Monte Carlo simulation of fatigue crack initiation and growth commonly provides information on the probability of having any leak and a leak exceeding a given magnitude. In order to have a nonzero number for the latter, a leak exceeding that magnitude must occur during the simulation. The problem is that this often does not occur within a number of trials that can be reasonably performed. In order to overcome this, PRAISE was modified to print out the length of any crack resulting in a leak and the time at which it first became through-wall. This was then used to estimate the size distribution of through-wall cracks as a function of time. The complementary cumulative distribution, denoted as $P_b(>b)$, is concentrated upon. Then the probability of a leak greater than $\dot{q}$ is given by

$$P_{LK}(>\dot{q}) = P_b[>b(\dot{q})] \qquad [F.8]$$

Table F.11 summarizes the information from a PRAISE run using the stresses from Table F.9 for the crack opening area (A) and leak rate ($\dot{q}$) for a given half-length of a through-wall crack (b).

Table F.11 Half Crack Lengths and Areas for a Given Leak Rate (Surge Line Elbow, Table F.9 Stresses)

Q, gpm	b, inches	$\frac{b}{\pi R_I}$	A, in^2	$\frac{A}{A_{pipe}}$
100	5.981	0.445	0.936	0.010
1500	10.379	0.591	10.028	0.143
5000	11.791	0.671	46.762	0.476
DEPB	15.95	0.907	--	--

A table of lengths of through-wall cracks was generated from the modified version of PRAISE using 10^4 trials using the stresses of Table F.9 (no seismic). In this run, there were 2,162 leaks within 25 years, 5,932 within 40 years and 8,890 within 60 years. Dividing these numbers by 10^4 provides leak probabilities that are nearly the same as obtained from the Monte Carlo simulation with 10^7 trials. Figure F.2 is the complementary cumulative distribution of leaking crack sizes for the three times of interest. The upper curve is for 60 years, because there is a higher probability of encountering a longer crack at this longer time. The lines in this figure are least squares curve fits, which are discussed later.

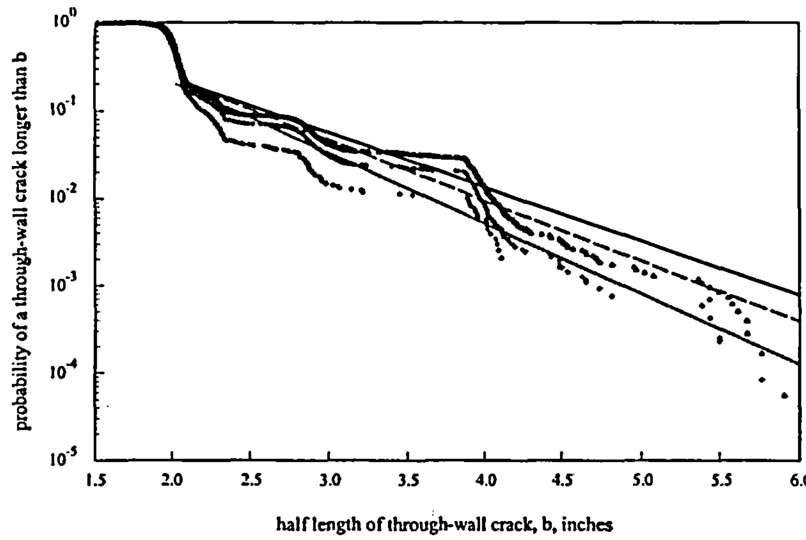

Figure F.2 Complementary Cumulative Distribution of Half-Crack Length of Through-Wall Cracks in Surge Line Elbow at 25, 40 and 60 Years

Figure F.2 shows changes in slope, and the "lumpiness" of the distribution is readily apparent. This "lumpiness" is representative of a multi-modal probability density function of crack length, which is most likely due to the fatigue crack initiation sites being taken as 2 inches in length. That is, each two-inch segment around the circumference is taken as an independent initiation site. The surface length of an initiated crack is also a random variable. Once a crack initiates, it grows, and can link with neighboring cracks. This growth and linking can lead to sudden increases in crack length (by linking) and evidently is responsible for the multi-modal nature of the probability density function of crack length. The multi-modal nature of the probability density function is clearly shown in Figure F.3.

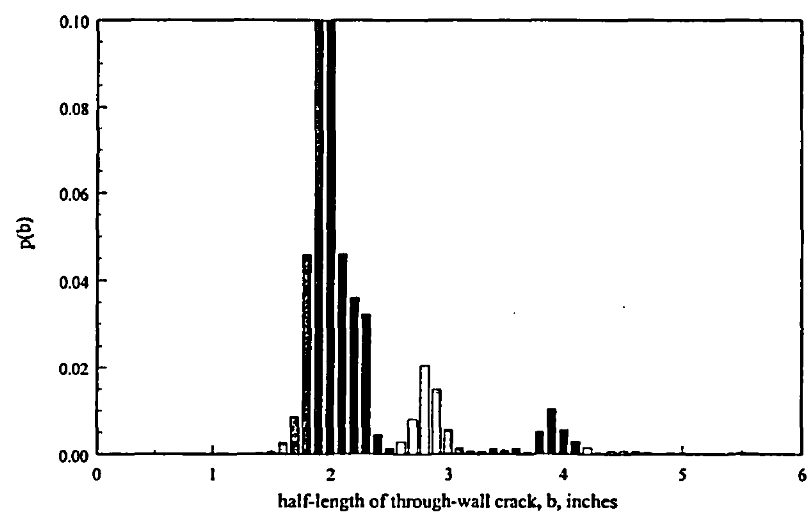

Figure F.3 Histogram of the Half-Length of Through-Wall Cracks at 60 Years for the Surge Line Elbow

Pleasing curve fits to the lines in Figure F.2 are not possible. A good fit could perhaps be obtained by assuming that the histogram of Figure F.3 consists of a sum of lognormals with medians to match the location of the modes and relative weights adjusted to match the relative heights of their modes. This is believed to be unwarranted, since the multi-modal nature of the density function is an artifact of the modeling assumption of a 50 mm (2 inch) long initiation site. It is better to just smooth out the cumulative distribution, and this was accomplished by a linear least squares curve fit to the cumulatives on log-linear scales. Since it is desired to represent the curve at long cracks, the fit was performed only for cracks corresponding to a probability less than 0.2. This eliminates the numerous cracks at higher probabilities that would skew the curve fit if included in the least squares calculations. The lines in Figure F.2 are the curve fits obtained in this manner.

The assumed functional form was

$$P(>b) = 0.2e^{-C(b-b_{0.2})} \qquad \text{[F.9]}$$

The values of C and $b_{0.2}$ depend on the time. Once they are evaluated, the leak probabilities of a given size are obtained using the crack sizes in Table F.11. Table F.12 summarizes the results.

Table F.12 Summary of Results for Surge Line Elbow (Table F.9 Stresses)

Time, years	25	40	60
Number of cracks	2162	5932	8890
Cracks above 0.2	1730	4744	7109
$b_{0.2}$	2.072	2.089	2.108
C	1.876	1.597	1.425
P(>5.981)	1.31×10^{-4}	4.00×10^{-4}	8.02×10^{-4}
P(>10.379)	3.41×10^{-8}	3.56×10^{-7}	1.52×10^{-6}
P(>11.791)	2.42×10^{-9}	3.73×10^{-8}	2.04×10^{-7}
P(>15.95)	9.86×10^{-13}	4.86×10^{-11}	5.43×10^{-10}

Table F.13 summarizes the results along with corresponding ones obtained directly from the Monte Carlo simulation. The conventional Monte Carlo simulation used 10^6 trials for 380 lpm (100 gpm) and 10^7 trials for 5,700 lpm (1,500 gpm).

Table F.13 Cumulative PRAISE Results for the Surge Line Elbow as Obtained from the Alternate Procedure and Directly from Monte Carlo Simulation (Table F.9 Stresses)

		Direct Monte Carlo	Alternate Procedure
>0	25	0.233	0.216
	40	0.587	0.593
	60	0.882	0.889
>100	25	$7.5x10^{-6}$	$1.31x10^{-4}$
	40	$7.1x10^{-5}$	$4.00x10^{-4}$
	60	$2.51x10^{-4}$	$8.02x10^{-4}$
>1500	25	$<10^{-7}$	$3.41x10^{-8}$
	40	$<10^{-7}$	$3.56x10^{-7}$
	60	$1.0X10^{-7}$	$1.52x10^{-6}$
>5000	25	--	$2.42x10^{-9}$
	40	--	$3.73x10^{-8}$
	60	--	$2.04x10^{-7}$
DEPB	25	--	$9.86x10^{-13}$
	40	--	$4.86x10^{-11}$
	60	--	$5.43x10^{-10}$

Table F.13 shows that the alternate procedure is able to greatly extend the leak rates whose probabilities can be estimated. In cases where direct comparisons are possible, the alternate procedure gives higher leak probabilities. The direct Monte Carlo for 5,700 lpm (1500 gpm) employed 10^7 trials and took 36 hours of computer time. The alternate procedure used 10^4 trials, so took about 2 minutes. Even in this era of fast cheap computer time, it would still be prohibitive to use direct Monte Carlo to generate the results obtained by the alternate procedure. It would take 10^{10} trials to produce the DEPB results in the above table. This translates to 36,000 hours of computer time, or about 4 years.

F.3.4 HPI Makeup Nozzle

An HPI/makeup nozzle safe end from a B&W plant type was selected as one of the base case systems.

F.3.4.1 Dimensions and Welds - This type of component was considered in Reference F. 5, which identifies the component as 2 ½ inch schedule 160 pipe fabricated from Type 304 austenitic stainless steel. The location considered in Reference F.5 is in the safe end at the nozzle, which has a thickness of 11.1 mm (0.4375 inches) and a mean radius of 32.5 mm (1.28 inches) at the location of high stresses.

F.3.4.2 Stresses and Cycles - As shown in Reference F.5, the cyclic stress history is dominated by two types of transients, with the amplitudes and frequencies shown in Table F.14.

Table F.14 Stress History for HPI/Make Up Nozzle from NUREG/CR-6674 [F.5]

Name	Stress Amplitude ksi	Number in 40 years
HPI actuation A/B	221.24	33
Test Null	169.31	7

The deadweight and restraint of thermal expansion stresses for this location under normal operation that were used in Reference F.5 are

$$\sigma_{dw}=0$$
$$\sigma_{te}= 63.1 \text{ MPa } (9.16 \text{ ksi})$$

As discussed above, these stresses were composed of 310 MPa (45 ksi) uniform and the remainder the generic gradient of Equation F.6. These stresses are believed to be very conservative and are for the thermal sleeve being intact.

F.3.4.3 Results - The version of PRAISE that considers fatigue crack initiation was run for the HPI/make up nozzle. The stresses of Table F.14 were taken to be axisymmetric. Due to the small line size, only 4 initiation sites around the circumference were considered. Table F.15 summarizes the results.

Table F.15 Cumulative Probability PRAISE Results for HPI/Make Up Nozzle (Intact Thermal Sleeve)

Condition		From Reference F.5	Here
>0	25		1.004×10^{-5}
	40	0.00210	6.08×10^{-4}
	60	0.0309	1.04×10^{-2}
			Inel4a2
>100	25	--	4.5×10^{-8}
	40	--	4.9×10^{-7}
	60	--	1.79×10^{-5}
			Inel4a1
>1500	25	--	2.0×10^{-8}
	40	--	2.10×10^{-7}
	60	--	4.56×10^{-6}
			Inel4a2

Table F.15 shows a cumulative leak probability of 10^{-5} in 25 years, which is quite low. However, leaks in this component have been observed in service, in which case the thermal sleeve in the component was failed. The results of Table F.15 use the stresses for an intact sleeve, and the stresses will be altered if the sleeve fails. A failed thermal sleeve is now considered.

F.3.4.4 Failed Thermal Sleeve - There is a thermal sleeve at the HPI nozzle, and the results in Table F.15 are for the case of the thermal sleeve not failing. The thermal sleeve has been observed to fail in service, which changes the stresses in the component.

In order to model the failure of the thermal sleeve, the following steps were taken:

1. Once the thermal sleeve fails, assume that a crack of the "initiation size" immediately appears. This size is a depth of 3.0 mm (0.12 inches). The WinPRAISE default distribution of the aspect ratio is used, as in other components.

2. A WinPRAISE run with this initial crack is performed, with the stresses that were present before the crack initiated (Table F.14), plus a uniform cyclic stress cycling each hour of sufficient amplitude to result in a high leak probability at not long times. This defines the uniform stress.

3. Use WinPRAISE to compute the leak frequencies for larger leak rates.

This procedure provides the results shown in Table F.16.

Table F.16 Cumulative PRAISE Results for HPI/Make Up Nozzle with Failed Thermal Sleeve and Additional Uniform Cyclic Stress, σ_u

		Intact Sleeve	With Initial Crack and Original Stresses, σ_u=0	With Initial Crack and σ_u = 8 ksi	With Initial Crack and σ_u = 12 ksi	With Initial Crack and σ_u = 25 ksi
>0	5	--	$5.67\text{x}10^{-5}$	$<10^{-2}$	0.047	0.18
	25	$1.004\text{x}10^{-5}$	$3.69\text{x}10^{-3}$	0.032	0.14	0.727
	40	$6.08\text{x}10^{-4}$	$1.26\text{x}10^{-2}$	0.129	0.33	0.909
	60	$1.04\text{x}10^{-2}$	$2.98\text{x}10^{-2}$	0.161	0.47	0.909
>100	25	$4.5\text{x}10^{-8}$	$6.49\text{x}10^{-4}$			$<10^{-5}$
	40	$4.9\text{x}10^{-7}$	$2.68\text{x}10^{-3}$			$<10^{-5}$
	60	$1.79\text{x}10^{-5}$	$5.31\text{x}10^{-3}$			$<10^{-5}$
>1500	25	$2.0\text{x}10^{-8}$	--			
	40	$2.10\text{x}10^{-7}$	--			
	60	$4.56\text{x}10^{-6}$	--			
break	25	--	$6.49\text{x}10^{-4}$			
	40	--	$2.68\text{x}10^{-3}$			
	60	--	$5.31\text{x}10^{-3}$			

Table F.16 shows that a uniform stress of some 170 MPa (25 ksi) is needed to result in an appreciable leak probability within 25 years. However, the frequency of larger leak rates is actually reduced by imposing the uniform stress that is necessary to produce the high leak probabilities seen in service. This uniform stress grows cracks to leaks, so that the larger leak rate frequencies are reduced. The least favorable condition for larger leaks is a failed thermal sleeve with the original stresses (σ_u = 0).

F.3.5 Recirculation Line – 12 inch

The recirculation line is one of the base case systems for a BWR. This system has developed leaks in the past due to intergranular stress corrosion cracking (IGSCC). The 12 inch line has some of the highest stresses, so is considered here. The recirculation system also has 28 inch lines, which can contribute to larger flow rate failures than possible from a 12 inch line. Hence, the 28 inch line is also considered in subsequent sections.

F.3.5.1 Dimensions and Welds - The layout of the recirculation system is given in isometrics made available to panel members. There are two recirculation loops, which are very similar to one another. There are 121 welds in this system, including field welds, shop welds and safe ends. The piping is fabricated from A-358 Class 1 Type 304, and the piping is of diameters 12, 22 and 28 inches – all schedule 80.

F.3.5.2 Stresses and Cycles - IGSCC will be the dominant degradation mechanism. Hence, time at stress is of major concern, and the number of stress cycles is of secondary importance. Estimated stresses at the highest stressed locations for the two pipe sizes of interest are given in Table F.17.

Table F.17 Stress Information for Two Recirculation Joints

OD, inch	Thickness, inch	σ_{NO}, ksi	Seismic σ, ksi
12.75	0.687	20.41	20.41
28	1.201	9.48	10.60

The normal operating stress (σ_{NO}) is the sum of the pressure stress, deadweight stress and restraint of thermal expansion stress. A value of 14 MPa (2 ksi) for the deadweight stress is assumed. The seismic stress is the normal operating stress plus the seismic-induced stress. Note that the seismic stresses are small in this case. The magnitude of the seismic event is unknown.

The time at stress is important for this case, with the cycling frequency being of less importance. Consistent with what is used for the PWR, the cycling is considered to be composed of heat up and cool down at 3 per year. The parameters related to stress corrosion cracking are summarized in Table F.18.

Table F.18 Stress Corrosion Cracking Parameters

Oxygen at startup (PPM) = 8.0
Oxygen at steady state (PPM) = 0.20
Heat up (100-550F) time (hrs) = 5.00
Coolant conductivity (μs/cm) = 0.20
Degree of sensitization (C/cm^2) = 7.04

Residual stresses will be important, and the default residual stress distributions in pcPRAISE, which are documented in Reference F.2, are used when no remedial treatments are performed. In order to include remedial treatments that have been performed in service, a weld overlay at 20 years will be considered. This alters the thickness, crack growth kinetics (post-treatment analyses use Type 316NG crack growth defaults in PRAISE) and residual stresses. The axisymmetric through-wall residual stress distribution of Figure F.4 is employed. This figure is from Reference F.15. PRAISE can not treat the actual gradient, so the linear approximation in this figure is used. The linear gradient employed underestimates the beneficial effect of the weld overlay.

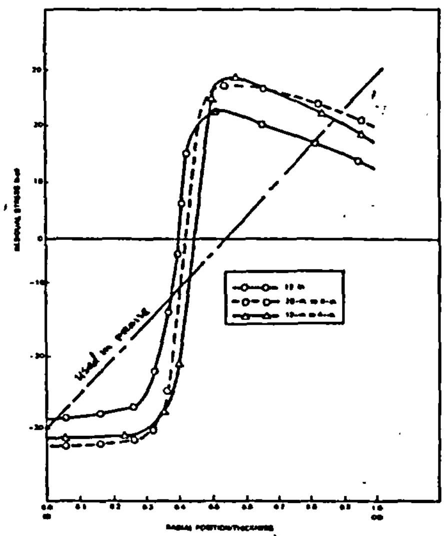

Figure F.4 Through-Wall Residual Axial Stress Distribution from Weld Overlay [F.15]

F.3.5.3 Results - Table F.19 summarizes the results obtained for the 12 inch weld in the recirculation system.

Table F.19 Cumulative Probability PRAISE Results for the 12 inch Recirculation Line Weld, with and without Weld Overlay at 20 Years (σ_{uo} = 141 MPa [20.41 ksi])

OD=12.75 inches, h=0.687 inches, wrought austenitic, stress corrosion crack initiation and growth

		Base	Overlay at 20 years	Overlay & σ_{DL} @ 39 years
>0	25	0.3674	0.2967	0.2968
	40	0.5986	0.3803	0.3872
	60	0.7435	0.4241	0.4253
>100	25	0.1682	0.1427	0.1429
	40	0.2452	0.1622	0.1632
	60	0.2872	0.1693	0.1708
>1500	25	0.1529	0.1066	0.1078
	40	0.2193	0.1250	0.1276
	60	0.2534	0.1312	0.1343
break	25	0.1529	0.0490	0.0502
	40	0.2193	0.0674	0.0700
	60	0.2535	0.0736	0.0767

5000 trials 304 full residual stress

σ_{dw}=2.0 ksi σ_{te}=13.32 ksi σ_{DL}=11.67ksi 3 HU-CD/yr p=1125 psi

The beneficial effect of the weld overlay at 20 years is not readily apparent from the results in Table F.19; such benefits are shown more clearly in Figure F.5, which provides a plot of the cumulative probability of a leak exceeding 380 lpm (100 gpm) as a function of time.

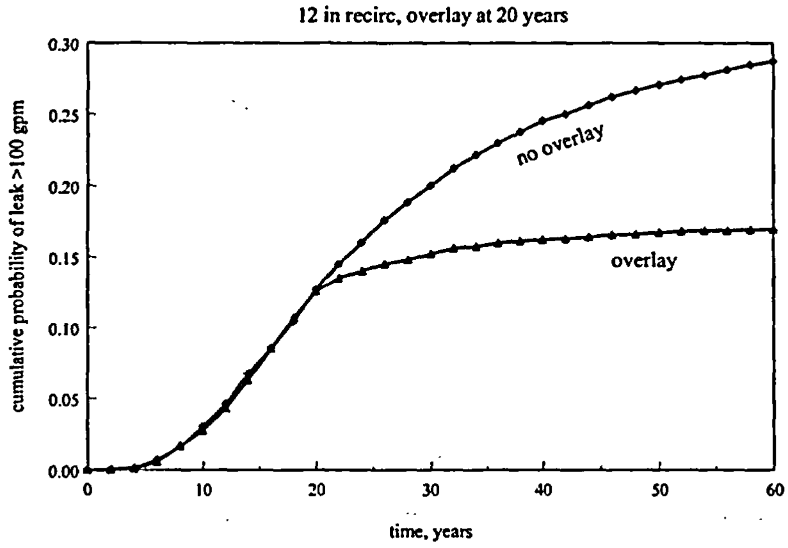

Figure F.5 Cumulative Probability of a Leak Exceeding 100 gpm as Functions of Time for the 12 inch Recirculation Line Weld with and without Weld Overlay at 20 Years

The slopes of the lines in Figure F.5 are the leak frequencies, and the slope at 40 years with no overlay is about 7 times that with overlay.

F.3.5.4 Summary of Observations from Service - Leak frequencies due to IGSCC in recirculation lines were estimated from service experience and reported in Reference F.16. Figure F.6 is Figure F.12 from that reference. With some exceptions, the results in Figure F.6 are between 10^{-4} and 10^{-3} per weld-year. The results are for times up to 15 years and do not include remedial actions. No strong dependencies on time or line size are apparent, but the smaller diameter lines appear to have a somewhat higher failure frequency.

Table F.20, which is from Charts 2 and 3 of Reference F.17, summarizes the depth distribution of observed cracks per weld-year for various pipe sizes in recirculation lines in BWRs. The remedial action of Reference F.17 is considered to consist of a weld overlay at 20 years. Observed crack sizes without remedial action, as reported in Reference F.16, are shown in Figure F.7.

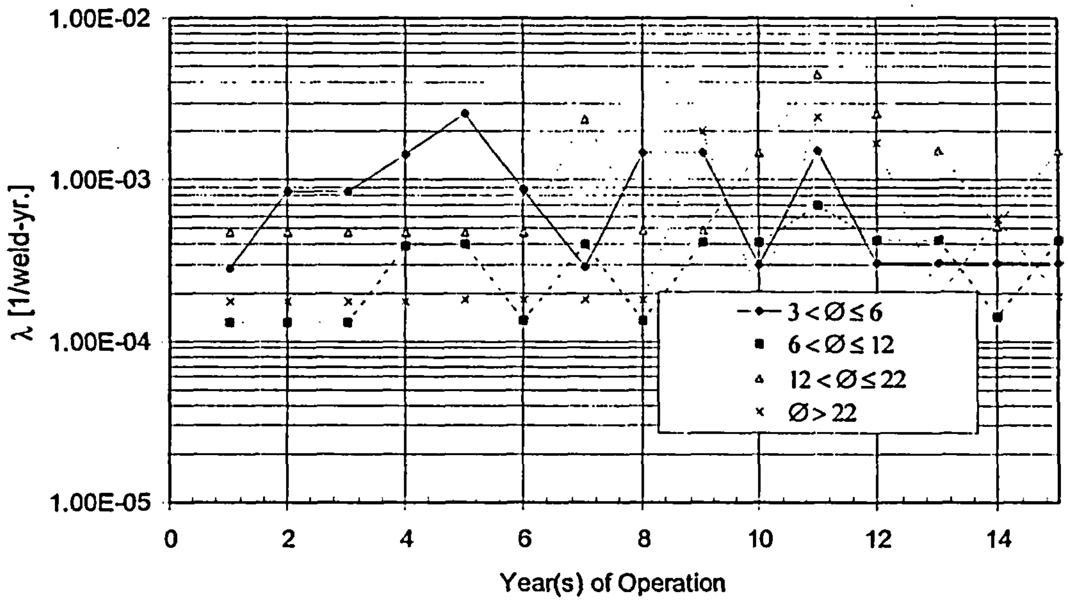

Figure F.6 Leak Frequencies as a Function of Time and Pipe Size (from Reference F.16)

Table F.20 Observed Crack Depth Frequencies in Various Line Sizes in Recirculation Lines as Percentages of the Wall Thickness (from Reference F.17)

No Remedial Action

Size	> 10%	> 20%	> 30%	> 40%	> 50%	> 60%	> 70%	> 80%	> 90%
NPS12	2.06E-03	1.62E-03	7.28E-04	3.64E-04	2.00E-04	1.46E-04	1.09E-04	7.28E-05	3.64E-05
NPS22	1.63E-03	1.11E-03	6.48E-04	3.21E-04	1.90E-04	1.24E-04	9.81E-05	6.54E-05	3.27E-05
NPS28	2.12E-03	1.50E-03	1.04E-03	5.99E-04	2.57E-04	1.84E-04	6.12E-05	3.67E-05	1.22E-05

With Remedial Action

Size	> 10%	> 20%	> 30%	> 40%	> 50%	> 60%	> 70%	> 80%	> 90%
NPS12	1.95E-04	1.60E-04	1.04E-04	8.31E-05	6.73E-05	4.61E-05	2.78E-05	1.90E-05	1.03E-05
NPS22	3.29E-04	2.74E-04	1.70E-04	1.32E-04	1.01E-04	8.62E-05	4.43E-05	2.95E-05	1.48E-05
NPS28	3.95E-04	2.84E-04	1.77E-04	9.15E-05	6.66E-05	3.82E-05	2.08E-05	1.24E-05	3.95E-06

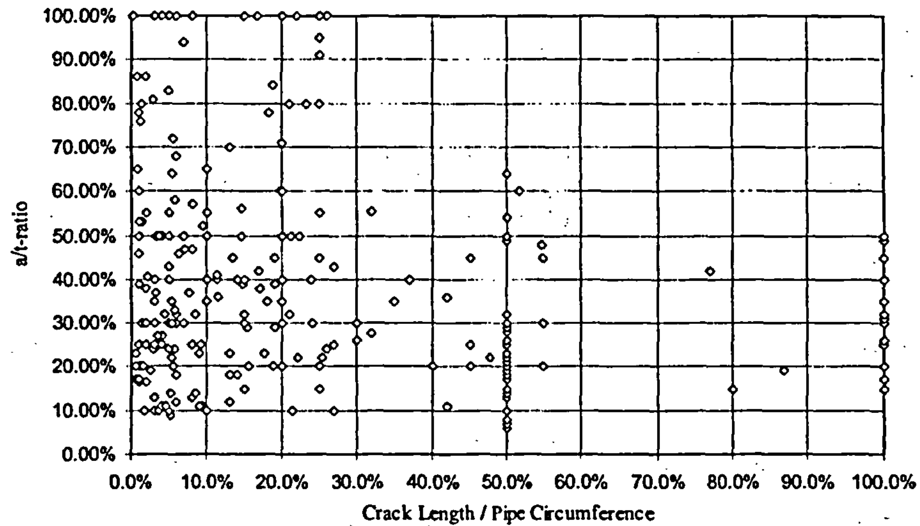

Figure F.7 Observed Crack Sizes as Reported in Reference F.16

F.3.5.5 Comparisons with PRAISE - The normal operating stress in Table F.17 of 20.41 ksi is for the highest stressed joint in the 12 inch recirculation line, whereas the observations are for all joints, including lower stressed locations. In order to generate PRAISE results that would be more representative of the population, runs were made with various stresses. Table F.21 summarizes the results.

Table F.21 Cumulative PRAISE Results for a 12 inch Recirculation Line Weld for Various Normal Operating Stresses (Remedial Action at 20 Years)

		Cumulative			
	Mean σ_{no}	10	12	15	20
	COV	0.0	0.3	0.0	0.0
	Mean σ_{te}	3.32	5.32	8.32	13.32
>0	25	1.42×10^{-3}	1.54×10^{-2}	9.36×10^{-2}	0.2967
	40	1.46×10^{-3}	1.89×10^{-2}	0.1473	0.3803
	60	1.46×10^{-3}	2.08×10^{-2}	0.1781	0.4241
>100	25	4.90×10^{-4}	7.59×10^{-3}	3.90×10^{-2}	0.1427
	40	4.90×10^{-4}	8.86×10^{-3}	5.35×10^{-2}	0.1622
	60	4.90×10^{-4}	9.48×10^{-3}	6.11×10^{-2}	0.1693
>5000	25	3.50×10^{-4}	5.80×10^{-3}	3.19×10^{-2}	0.1066
	40	3.50×10^{-4}	7.06×10^{-3}	4.53×10^{-2}	0.1250
	60	3.50×10^{-4}	7.684×10^{-3}	5.27×10^{-2}	0.1312
DEP	25	1.00×10^{-4}	2.70×10^{-3}	2.12×10^{-2}	0.0490
	40	1.00×10^{-4}	2.96×10^{-3}	3.46×10^{-2}	0.0674
	60	1.00×10^{-4}	4.58×10^{-3}	4.20×10^{-2}	0.0736

The results for 138 MPa (20 ksi) correspond to those in Table F.18. The normal operating stress was taken to be deterministic, except for the case of 83 MPa (12 ksi), in which case the normal operating stress is normally distributed with a mean of 83 MPa (12 ksi) and a standard deviation of 0.3x(5.32+2.00) = 15.2 MPa (2.20 ksi).

The cumulative results from Table F.21 can be compared with the observed frequencies in Table F.20 by converting the cumulative results to a frequency by dividing by the time increment involved. In the current case, the increase in the cumulative following the remedial action is relatively small, as seen from Figure F.5. Hence, the cumulative results at 25 years from Table F.21 should be divided by 20 to provide frequencies for comparison purposes. This provides the results in Table F.22.

Table F.22 Estimated Leak Frequencies Prior to Remedial Action, from Table F.21

Mean σ_{no}	10	12	15	20
COV σ_{no}	0.0	0.3	0.0	0.0
Mean σ_{te}	3.32	5.32	8.32	13.32
Frequency	7.11×10^{-5}	7.69×10^{-4}	4.68×10^{-3}	1.48×10^{-2}

This table shows that the mean normal operating stress of 83 MPa (12 ksi) with some variance provides the best agreement with the results of Figure F.6. This is the case that will be used for benchmarking against observed cracks.

The following steps were followed in order to provide PRAISE results for comparison with observations of part-through cracks:

1. The WinPRAISE software was modified to print out the sizes of cracks present at each time step in the analysis. The depth and length of the deepest crack and the longest crack at that time step are printed into a file, along with the number of cracks present at that time. This file contains at most a number of lines equal to the number of Monte Carlo trials times the number of time steps (which can be a lot of lines).

2. The WinPRAISE file from step 1 is then processed to provide another file that includes only the sizes of part-through cracks present at the time of interest (25 years in this case). (Cracks of zero depth, leaks and other times are eliminated.)

3. The crack size file from step 2 is then loaded into a histogram, which provides the number of cracks present at 25 years that fall within a certain depth range.

4. Since Reference F.17 reports detected cracks, the detection probability (Equation F.1) must be accounted for. This is accomplished by multiplying the number of cracks in each bin by the detection probability for a crack of depth equal to the midpoint of the bin. This provides the number of detected cracks in this bin. The contents of each bin are then divided by the number of trials times the time (25 years) to provide the crack sizes per weld year.

5. The histogram is then converted to a complementary cumulative form, which is then directly comparable to results from Reference F.17.

Figure F.8 presents the crack size results for the benchmark case. Once again, not many deep cracks are observed. A pattern is observed in Figure F.8 which shows a preponderance of cracks below about 2.5 mm (0.1 inches). This pattern is due to cracks growing to a depth of 2.5 mm (0.1 inch) and then slowing down or arresting, which is most likely due to the transitioning from growth of "initiating cracks" to "fracture mechanics cracks" that occurs in the PRAISE modeling of initiation and growth. The transitioning criteria are discussed on page 42 of Reference F.2, and one of the criteria is "If the depth of the crack is greater than 2.5 mm (0.1 inch), its growth will always be by fracture mechanics velocity".

Figure F.9 is a plot of the predicted complementary cumulative number of observed cracks for the benchmark case, along with a comparison with reported observations. The outstanding inspection parameters of Table F.1 were employed. In this figure, Reference F.17 results from Chart 1 (prior) and Chart 2 (posterior) are both shown, since the analysis mixed with and without remedial action (weld overlay at 20 years). The analysis results fall midway between the two results, except for shallow cracks.

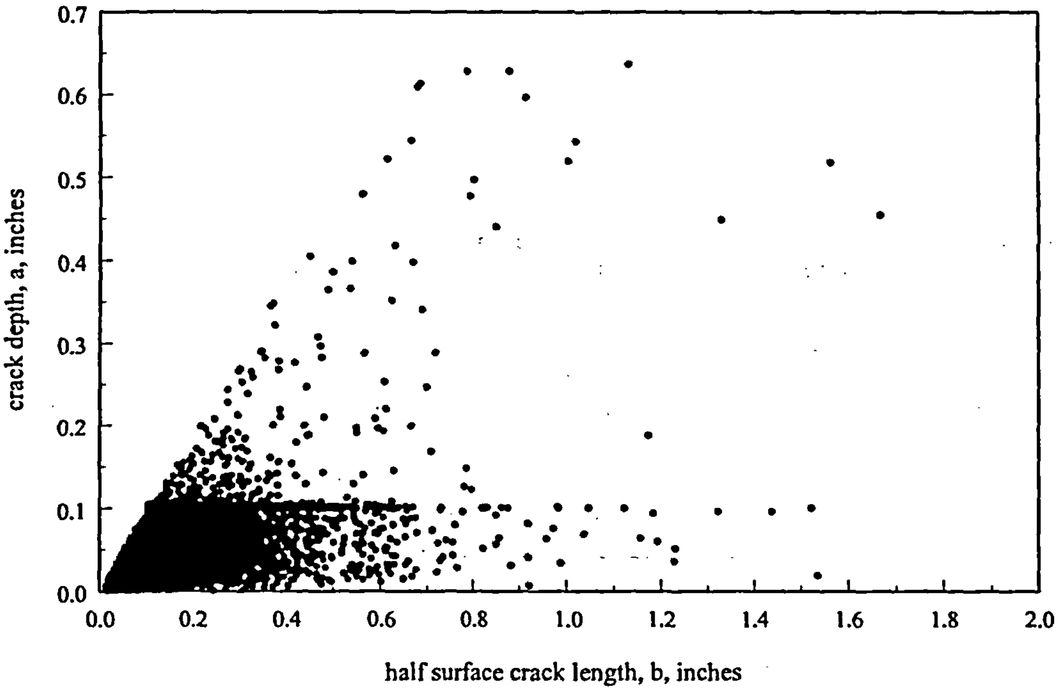

Figure F.8 Crack Sizes After 25 Years for the Benchmark Case (Mean σ_{NO} = 12 ksi) with Weld Overlay at 20 Years

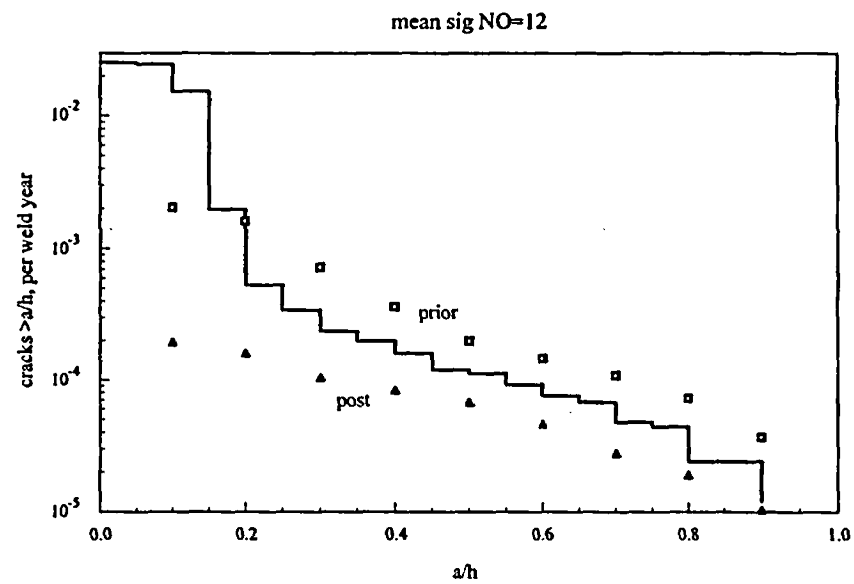

Figure F.9 Comparison of Results for the Benchmark Case Predicted for Outstanding Inspection Quality with Reported Prior and Post Observations [F.17]

The agreement shown in Figure F.9 is felt to be quite good, and indicates that the PRAISE model best fits the observed crack depths when the mean stress of 83 MPa (12 ksi) is used. Figure F.10 shows that the stress has an important effect, because the agreement is not so good when a stress of 103 MPa (15 ksi) is employed.

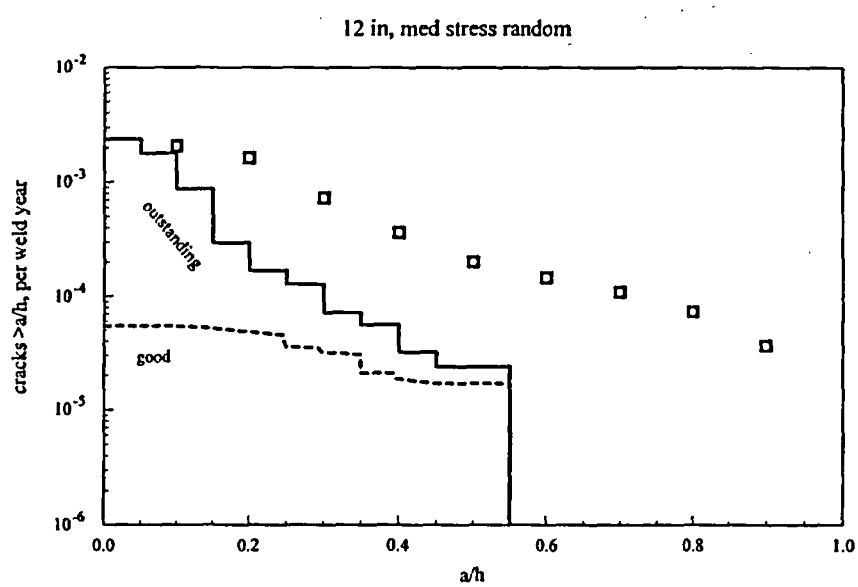

Figure F.10 Comparison of Results for the High-Stress Case Predicted for Two Inspection Qualities with Reported Observations (the Square Data Points are the Observations [F.17], no Weld Overlay)

The results of Figure F.7 are for observed (detected) cracks, whereas Figure F.8 has not had the nondetection probability applied. A direct comparison is therefore not possible, and the two figures are plotted on much different scales. However, the PRAISE predictions contain a much greater proportion of short and shallow cracks than the observations. This could be somewhat affected by the nondetection probabilities, but the differences would not be removed by applying the nondetection probabilities to the cracks predicted by PRAISE.

The question immediately arises regarding the results to be used in the estimation of the recirculation system reliability; the results of Table F.20 for the highly stressed joint or the results of Table F.21 for the benchmarked stress of 83 MPa (12 ksi). Interestingly, the system and average weld leak frequencies are nearly the same whether the 83 MPa (12 ksi) or 140 MPa (20 ksi) weld is used, because the number of joints involved also depends on the stress. Table F.23 summarizes this comparison. This table is in terms of the leak frequency per year, which is obtained from the cumulative results given above by use of Equation F.2.

The system frequency and average frequency per weld are nearly the same for both cases.

Table F.23 System and Average Weld Leak Frequencies for Two Cases of the 12 inch Recirculation Line

Mean σ_{NO}, ksi	12	20
Mean σ_{te}, ksi	5.32	13.32
COV	0.3	0
Std Dev of σ_{te}, ksi	1.6	0
	per weld joint	
0 - 25 years	6.15×10^{-4}	1.19×10^{-2}
25 – 40 years	2.36×10^{-4}	5.57×10^{-3}
40 – 60 years	9.25×10^{-4}	2.19×10^{-3}
Number dominant. joints	49	2
Number in system	49	49
	System (times number dom. joints)	
0-25	3.01×10^{-2}	2.38×10^{-2}
25-40	1.16×10^{-2}	1.11×10^{-2}
40-60	4.53×10^{-3}	4.38×10^{-3}
	Average per joint ($\div 49$)	
0-25	6.15×10^{-4}	4.86×10^{-4}
25-40	2.36×10^{-4}	2.24×10^{-4}
40-60	9.25×10^{-4}	8.94×10^{-5}

Of the two cases in Table F.23, the case of a mean stress of 83 MPa (12 ksi) and coefficient of variation of 0.3 (on $\sigma_{te} + \sigma_{dw}$) is more representative of the population of joints as a whole, so is preferred for comparisons with observations of part-through cracks.

F.3.6 Recirculation Line– 28 inch

Stresses and dimensions are given in the corresponding sections for the 12 inch line. IGSCC crack initiation and growth are the dominant degradation mechanisms. Table F.24 summarizes the results for this weld.

Table F.24 Cumulative PRAISE Results for the Weld in the 28 inch Recirculation Line

OD=28 inches t = 1.201 inches

	Time	Probability
>0	25	$6.23x10^{-3}$
	40	$1.02x10^{-2}$
	60-	$1.46x10^{-2}$
>100	25	$6x10^{-4}$
	40	$8x10^{-4}$
	60-	$8x10^{-4}$
>1500	25	$6.66x10^{-5}$
	40	$6.66x10^{-5}$
	60-	$1.25x10^{-4}$
>5000	25	$6.00x10^{-5}$
	40	$6.87x10^{-5}$
	60-	$9.79x10^{-5}$
break	25	$3.3x10^{-5}$
	40	$3.3x10^{-5}$
	60-	$6.7x10^{-5}$

σ_{dw}=2.0 ksi
σ_{te}=1.75 ksi
P = 1,125 psi
Type 304 full
residual stress
3 HU-CD/yr

F.3.7 Feedwater Elbow

The feedwater elbow is one of the base case systems. This system is subject to flow accelerated corrosion (FAC), which can be a serious degradation mechanism if left unchecked. PRAISE can not model FAC, but some analyses are provided for fatigue crack initiation and growth.

F.3.7.1 Dimensions and Welds - The layout of the feedwater system is given in the piping isometrics made available to the panel members. There are some 123 welds in the two loops of the feedwater systems, all but 6 of them in 12 and 20 inch piping. The 12 inch lines are schedule 100 (17.4 mm [0.687 inches] thick) and the 20 inch lines are schedule 80 (32.5 mm [1.281 inches] thick). The material is A-333 Grade 6 (which is a carbon steel).

F.3.7.2 Stresses and Cycles - The feedwater line elbow is considered in Reference F.5, so this is evidently the high stress point in the system. Note that there are at least 6 such elbows in a feedwater system. (There are many more elbows, but they are likely to not be so highly stressed). The degradation mechanism is fatigue and flow accelerated corrosion (FAC). Stresses do not contribute to FAC, so are not needed for this mechanism. For fatigue, there are a considerable number of cycles of high stress amplitude. They are available from Reference F.5. Table F.25, which (except for the column of temperatures) is page A.25 of Reference F.5, summarizes the stresses. These stresses are "decomposed" according to the procedure discussed above for the surge line. The analysis reported in Reference F.5

used a temperature of 590°F (310°C), as indicated in the text at the top of Table F.25. However, Table 5-123 of Reference F.18 provides the temperatures for these transients, and it is suggested that these temperatures be used, because their use is more realistic and less conservative. They are included as the right-hand column of Table F.25. The temperature influences the strain-life curve, and has a noticeable effect on the computed failure probabilities because of its influence on the initiation probabilities.

The values of the deadweight and restraint of thermal expansion under normal operation that Reference F.5 uses for this location are

$$\sigma_{dw} = 0$$
$$\sigma_{te} = 115 \text{ MPa } (16.68 \text{ ksi}).$$

The stress history in Table F.25 most likely contains seismic events. It is not possible to eliminate them from the list using information currently available, but their influence on the calculated failure probabilities is expected to be minimal.

Table F.25 Summary of Stress Cycles for Feedwater Line Elbow (from Page A.25 of NUREG/CR-6674 [F.5])

```
NAME OF PLANT          =    GE-NEW
NAME OF COMPONENT      =    FEEDWATER LINE ELBOW
NUM OF LOAD PAIRS      =    28
MATERIAL               =    LAS
WALL THICK (INCH)      =    1.000
INNER DIAMETER         =    12.000
AIR/WATER              =    WATER
TEMPERATURE(F)         =    590.000
SULFUR(WHT%)           =    .015
DISOL O2 (PPM)         =    .100
STR RATE (%/SEC)       =    0.00100
USEAGE(DETERM.)        =    3.68800
P-INITIATION@40        =    1.59E-01
P-INITIATION@60        =    3.65E-01
      P-TWC @40        =    1.01E-03
      P-TWC @60        =    1.46E-02
```

LOAD PAIR	AMP(KSI)	NUM/40 YR	EDOT(%/S)	USEAGE	TEMP, °C
HIGH 18/LOW 21	106.040	5.0	.117000	.025000	200
HIGH 18/LOW 21	103.960	5.0	.114000	.024000	200
HIGH 18/LOW 21	102.610	5.0	.113000	.024000	200
HIGH 14/LOW 17	91.590	8.0	.001000	.123000	200
HIGH 8/LOW 17	89.400	10.0	.095000	.037000	200
HIGH 3/LOW 16	88.270	5.0	.094000	.018000	200
HIGH 8/HIGH 7	83.760	126.0	.041000	.519000	200
HIGH 7/HIGH 7	81.430	10.0	.086000	.033000	215
HIGH 7/LOW 13	67.930	97.0	.001000	.740000	200
HIGH 7/LOW 13	66.710	14.0	.001000	.101000	200
HIGH 7/LOW 15	61.290	6.0	.001000	.035000	200
HIGH 7/LOW 15	61.160	64.0	.001000	.451000	212
HIGH 8/LOW 12	55.500	92.0	.001000	.391000	200
HIGH 3/LOW 12	46.630	88.0	.001000	.254000	215
HIGH 7/LOW 22	42.880	15.0	.001000	.029000	212
HIGH 3/HIGH 7	39.440	212.0	.001000	.315000	215
HIGH 3/HIGH 7	38.130	69.0	.001000	.104000	224
HIGH 3/LOW 20	36.800	11.0	.001000	.014000	224
HIGH 4/LOW 20	34.320	60.0	.001000	.053000	215
LOW 11/LOW 20	32.950	203.0	.001000	.122000	200
HIGH 7/LOW 11	32.530	360.0	.001000	.203000	200
HIGH 6/LOW 11	29.770	222.0	.025000	.035000	200
HIGH 2/HIGH 19	26.090	30.0	.028000	.003000	212
HIGH 5/HIGH 19	26.040	81.0	.028000	.007000	200
HIGH 5/HIGH 9	21.640	96.0	.001000	.012000	212
HIGH 1/HIGH 11	20.560	40.0	.001000	.003000	200
LOW 10/LOW 11	14.180	30.0	.001000	.001000	200
HIGH 5/LOW 11	11.220	11515.0	.001000	.008000	200

F.3.7.3 Results - PRAISE runs for this component were made using the version that can treat fatigue crack initiation with details of the circumferential variation of the stresses. The feedwater system is relatively more likely to experience water hammer, so the influence of an overload event with a stress of $0.42\sigma_{flo} = 128$ MPa (18.5 ksi) above that normally present was considered. This stress is denoted as σ_{DL}, and results were generated for one cycle of this stress at 24, 39, or 59 years. The results are summarized in Table F.26, which includes the effects of σ_{DL} (columns D & F).

Table F.26 Cumulative PRAISE Results for Feedwater Line Elbow

		A	B	C	D	E	F	G
Stresses		Ref. F.5	Table F.25	Table F.25	Table F.25	Table F.25	Table F.25	80% of Table F.25
Failure Criterion		σ_{flow}	σ_{flow}	σ_{flow}	σ_{flow}	σ_{flow} & J-T	σ_{flow} & J-T	σ_{flow}
σ_{DL}		no	no	no	σ_{DL}@(t-1)	no	σ_{DL}@(t-1)	no
>0	25	--	--	$<10^{-8}$	2.5×10^{-8}	1.0×10^{-7}	3.10×10^{-6}	$<10^{-7}$
	40	0.001	2×10^{-6}	5.69×10^{-6}	7.19×10^{-6}	1.54×10^{-5}	1.43×10^{-4}	$<10^{-7}$
	60	0.0146	1.8×10^{-4}	2.57×10^{-4}	2.59×10^{-4}	$\sim5\times10^{-4}$	2.9×10^{-3}	4.6×10^{-7}
			Ref F.6 Table 4-8	10^8 trials				GEN6TWA4
>100	25	--	--	$<10^{-8}$	1.5×10^{-6}*	$<10^{-7}$	1.70×10^{-6}*	
	40	--	--	$<10^{-8}$	1.5×10^{-6}*	$<10^{-7}$	1.70×10^{-6}*	
	60	--	--	$<10^{-8}$	1.50×10^{-6}*	--	2.1×10^{-6} *	
				GENC6TW4				
>1500	25	--	--					$<10^{-7}$
	40	--	--					$<10^{-7}$
	60	--	--					$<10^{-7}$
			axi-symmetric actual T					reduced stresses

* also a break

Case A is directly from Reference F.5, and Case B is directly from Table 4-8 of Reference F.6. Case C is Case B rerun with 10^8 trials. Cases D-G are variations of C with different failure criteria, overloads and reduction of stresses. The results for various failure criteria (critical net section stress only or critical net section stress and tearing instability) show that consideration of tearing instability noticeably increases the computed failure probability (compare, for instance, cases C&E). Consideration of an overload event also has a noticeable effect (E&F). The use of lower stresses markedly reduces the computed failure probabilities (G & C). In the case of an overload event, the probability of a 100 gpm failure is the same as a complete pipe break.

F.3.7.4 Alternate Procedure - The results of Table F.26 show that the probability of a large leak was obtainable from the Monte Carlo procedure only when a large overload occurred. When this did not occur, there were no leaks of even 380 lpm (100 gpm) in 10^7 or 10^8 trials. In order to obtain estimates for the larger leak probabilities, the alternate procedure discussed for the surge line was also applied to Case C of Table F.26 for the feedwater elbow.

As before, the crack length for a given leak rate, $b(\dot{q})$, was obtained from a pcPRAISE run, along with the half-crack length of any cracks that become through-wall. Figure F.11 provides a plot of the leak rate as a function of b for the feedwater elbow.

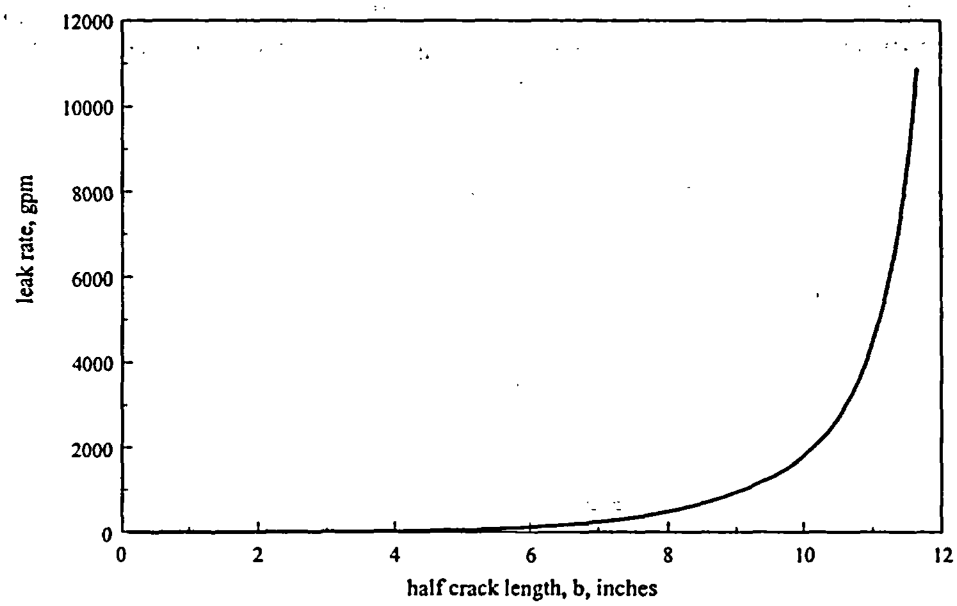

Figure F.11 Leak Rate as a Function of Half Crack Length for Feedwater Elbow Base Case C

The results in Table F.27 are obtained from this figure and the corresponding pcPRAISE results. This table also includes the portion of the circumference that is cracked and the proportion of the crack opening area to the flow area of the pipe. It is seen that the opening area of the crack is nearly equal to the flow area of the pipe when the leak rate is 19,000 lpm (5,000 gpm). The value of b for a complete pipe break, as obtained from Equation E.7 is also included. Table F.29 defines $b(\dot{q})$.

Table F.27 Half Crack Lengths and Areas for a Given Leak Rate (Feedwater Elbow Base Case C)

$\dot{q}$, gpm	b, inches	$\frac{b}{\pi R_I}$	A, in^2	$\frac{A}{A_{pipe}}$
100	5.737	0.32	1.837	0.02
1500	9.743	0.55	27.554	0.27
5000	11.095	0.62	90.877	0.91
DEGB	15.925	0.89	--	--

As before, the next step is to estimate the probability of having a through-wall crack exceeding a given length as a function of time. The modified version of pcPRAISE was used to generate a table of values of b and the time at which the leak first occurred. A run was made with 10^7 trials, with 2,607 cracks becoming through-wall within 60 years. This corresponds to a leak probability of 2.607×10^{-4} at 60 years, which agrees closely with the leak probability obtained earlier. Of these 2,607 cracks, none appeared before 25 years, and 64 occurred between 25 and 40 years. The statistical distribution of these 64 cracks at 40 years provides the probability of having a through-wall crack greater than a given length within 40 years. Extrapolation is required to obtain results for the crack lengths included in Table F.27. Figure F.12 shows the complementary cumulative distribution of b at 40 years, along with the curve fit of Equation F.10.

$$P(>b) = e^{-5.34(b-1)} \qquad \text{(40 years)} \qquad \text{[F.10]}$$

Note that the plot starts at a half-crack length of 25 mm (1 inch), and that the data are closely approximated by a straight line on log-linear scales.

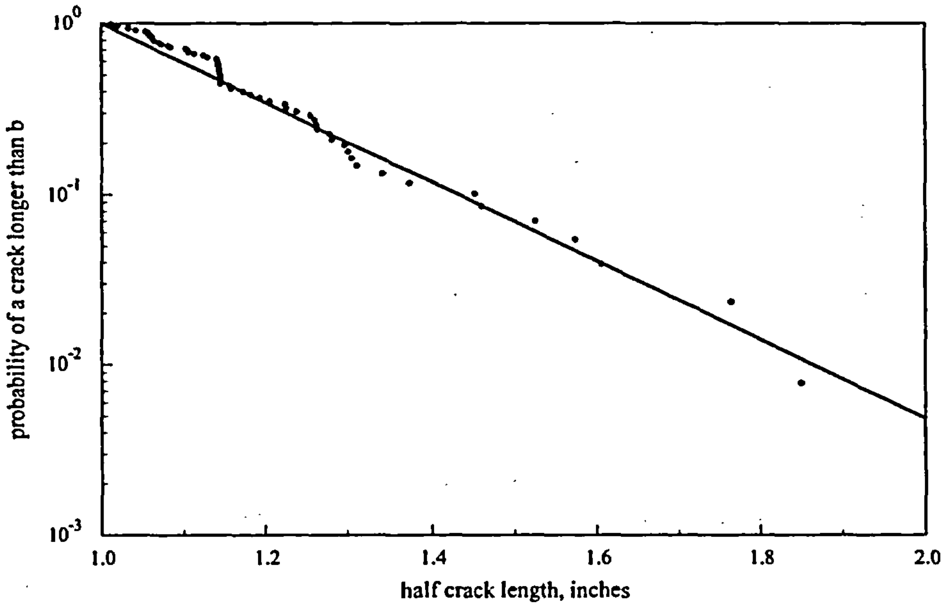

Figure F.12 Complementary Cumulative Distribution of Half-Crack Length of Through-Wall Cracks in Feedwater Elbow within 40 Years, Along with Fit

Figure F.13 provides a similar plot for the 2,607 through-wall cracks that occurred within 60 years. Equation F.11 is the fit of the distribution at 60 years within the range of interest.

$$P(>b) = 0.0274e^{-2.25(b-1)} \qquad \text{(60 years)} \qquad \text{[F.11]}$$

Note that in this case the data appear bilinear and are not well approximated by a straight line on log-linear scales. To represent the data at the longer crack lengths of interest, a straight line was assumed beyond a crack length of 50 mm (2 inches). This corresponds to a probability below about 0.003.

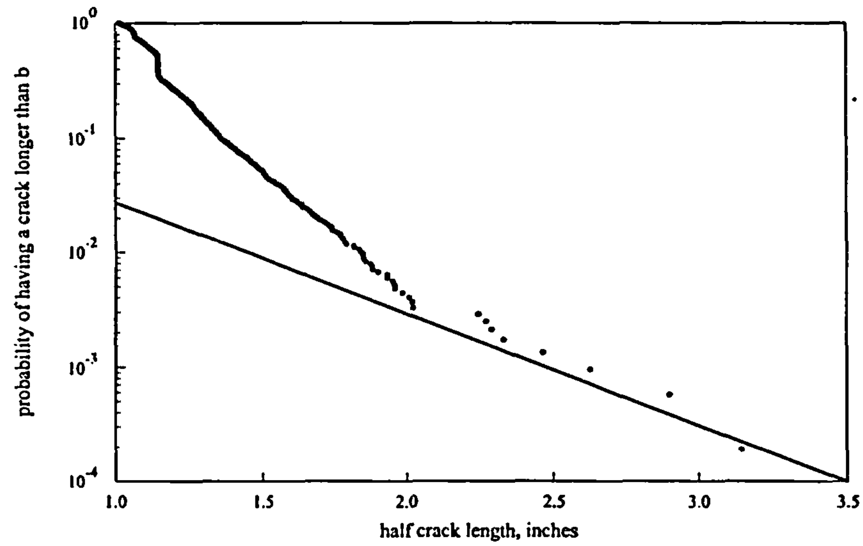

Figure F.13 Complementary Cumulative Distribution of Half-Crack Length of Through-Wall Cracks in Feedwater Elbow within 60 Years, Along with Fit

The probability of a leak exceeding a given size within 40 and 60 years is then obtained by taking using the value of *b* for a given leak rate from Table F.27 in conjunction with Equations F.10 and F.11, respectively. Table F.28 summarizes the results.

Table F.28 Cumulative Results for Feedwater Elbow Case C

	time years	$P(>\dot{q})$
>0	25	$<10^{-8}$
	40	5.69×10^{-6}
	60	2.57×10^{-4}
>100	25	--
	40	1.03×10^{-11}
	60	6.44×10^{-7} *
>1500	25	--
	40	5.29×10^{-21}
	60	7.84×10^{-11}
>5k	25	--
	40	3.88×10^{-24}
	60	3.74×10^{-12}
DEPB	25	--
	40	2.44×10^{-35}
	60	7.14×10^{-17}

* direct Monte Carlo gave $<10^{-8}$

The leak (>0) results in Table F.28 came directly from the Monte Carlo simulation. With 10^8 trials, no leaks exceeding 380 lpm (100 gpm) were obtained. Hence, the Monte Carlo simulation predicts $<10^{-8}$ probability of a leak exceeding 380 lpm (100 gpm) within 60 years. The alternative procedure gave a corresponding value of 6.44×10^{-7}. This suggests that the alternative procedure overestimates the probability of a given leak, as was also the case for the surge line elbow.

F.4 Selection of Reference Cases and Extension to System Frequencies

The earlier sections of this document contain many sets of results for each base case component. The multiple cases were generated primarily as a series of sensitivity studies. For these results are to be useful in the LOCA elicitation, a reference case must be selected for each component as being representative for that component. This section briefly discusses which case for each component is suggested as the reference case, and system leak frequencies are presented for each reference case.

The joint frequency is calculated from the cumulative results reported above by use of Equation F.2. The system frequencies are then obtained by multiplying by the number of highly stressed joints in the system (this approximation works because the failure probabilities are generally small).

Each component is discussed, with a summary table provided after all components are discussed.

F.4.1 Hot Leg Pressure Vessel

As shown in Tables F.5 – F.7, the large leak (>100 gpm and larger) probabilities for this component varied considerably, depending on the crack growth mechanism (cycle-dependent fatigue or time-dependent stress corrosion cracking [PWSCC]), and whether crack initiation or growth from pre-existing defects was considered. The fatigue crack growth results (Table F.5) were very low ($\sim 10^{-18}$), and the PWSCC crack initiation results (Table F.6) were quite large ($\sim 10^{-5}$). Since it is expected that this component will totally dominate the very large ($>10^5$gpm) leak category, the selection of the reference case is critical for very large leak estimates. The PWSCC with fabrication defects has intermediate failure probability results ($\sim 10^{-10}$), and is recommended as the reference case. The case without residual stresses is selected. Table F.6 shows that residual stresses do not have a large influence. The time dependency of the large leak cumulative probability is very small, which suggests that the leak frequency is very small. For estimation purposes, the leak frequencies are estimated by taking the value of the cumulative at 60 years, dividing it by 60, and assuming the value to be applicable independently of time. This will overestimate the leak frequency at long time and underestimate it at short time.

For extension to system failure frequency, it is assumed that there are three comparably stressed joints in the large main coolant piping.

F.4.2 Surge Line Elbow

The surge line elbow result identified as "axisymmetric nonseismic" in Table F.12 is suggested as the reference case. Table F.13 summarizes the cumulative results for the larger flow rates, which were obtained by the alternative procedure.

Two of these highly stresses elbows are considered to be present in the surge line system

F.4.3 HPI Makeup Nozzle

Probability analyses were performed with and without failure of the thermal sleeve, which has been observed to fail in service. The least favorable large leak probabilities were for a failed thermal sleeve, which immediately resulting in fatigue crack initiation, but with the same stresses as before. This is suggested as the reference case, with the column labeled $\sigma_u = 0$ in Table F.16 being the results of interest.

Three such locations are considered to be present in the system.

F.4.4 Recirculation Line – 12 inch

Analyses were performed for this component for a range of applied stresses, with predictions compared to field experience of leaks and observed part-through cracks. Analyses were performed for no remedial action, and for a weld overlay at 20 years. The weld overlay at 20 years is considered to be the most realistic. Comparisons with experience led to an estimate of stresses that were considerably below the peak value used in the original analysis. However, when compensated for the number of weld joints involved, the system leak frequencies were nearly the same whether 49 joints with a random stress (mean σ_{NO} = 83 MPa [12 ksi]) or 2 joints with a high stress (σ_{NO} = 140 MPa [20 ksi]) were considered (see Table F.25). The case of weld overlay at 20 years with the high stress is recommended as the reference case. Table F.19 contains the cumulative results.

Two of the highly stresses joints are considered to be present in the recirculation system.

F.4.5 Recirculation Line – 28 inch

The recirculation line with no remedial action and a high stress representing the dominant joints was the only case considered, and is summarized in Table F.24.

Two such joints are considered to be present in the system.

F.4.6 Feedwater Elbow

Case C in Table F.26 is suggested as the reference case. Results for > 380 lpm (100gpm) and larger were generated by the alternative procedure, and are summarized in Table F.28.

Four such locations were considered to be present in the system.

F.4.7 Summary Table

Table F.29 provides an overall summary of the leak flow rate frequencies for the reference cases of the base case systems.

Table F.29 Summary of Results for Reference Systems

			Hot Leg	Surge Line	HPI	Recirculation 12	Recirculation 28	Feedwater
		OD, in	34	14	3.44	12.75	28	12.75
		t, in	2.5	1.406	0.4375	0.687	1.201	0.687
		A, in^2	661	98.3	5.167	102	515	102
		Q_{max}	423	63	3.6	38	193	38
		matl	cast SS	SS	SS	SS	SS	CS
		Degr Mech	PWSCC growth	fatigue init&gro	fatigue	SCC init&gro	SCC init&gro	fatigue init&gro
		Table	F.6	F.12	F.16	F.21	F.26	F.26, F.28
		Case	PWSCC no σ_{res}	Table F.9 stresses	failed slv σ_u=0	overlay @ 20 yrs		C
		Insp	0,20,40	none	none	0,20,40	0,20,40	none
dominant joint freq	>0	0-25	--	$9.3x10^{-3}$	$1.48x10^{-4}$	$1.19x10^{-2}$	$2.5x10^{-4}$	$<4x10^{-10}$
		25-40	--	0.024	$5.94x10^{-4}$	$5.57x10^{-3}$	$2.6x10^{-4}$	$3.8x10^{-7}$
		40-60	--	0.015	$8.60x10^{-4}$	$2.19x10^{-3}$	$2.2x10^{-4}$	$1.3x10^{-5}$
	>0.1	0-25	$1.33x10^{-8}$	$3.0x10^{-7}$	$2.60x10^{-5}$	$5.71x10^{-3}$	$2.4x10^{-5}$	--
		25-40	$1.33x10^{-8}$	$4.2x10^{-6}$	$1.35x10^{-4}$	$1.30x10^{-3}$	$1.3x10^{-5}$	$6.9x10^{-13}$
		40-60	$1.33x10^{-8}$	$9.0x10^{-6}$	$1.32x10^{-4}$	$3.55x10^{-4}$	$<5x10^{-6}$	$3.2x10^{-8}$
	>1.5	0-25	$1.6x10^{-11}$	$1.4x10^{-9}$	$2.60x10^{-5}$	$4.26x10^{-3}$	$2.7x10^{-6}$	--
		25-40	$1.6x10^{-11}$	$2.2x10^{-8}$	$1.35x10^{-4}$	$1.23x10^{-3}$	--	$3.5x10^{-22}$
		40-60	$1.6x10^{-11}$	$5.8x10^{-8}$	$1.32x10^{-4}$	$3.10x10^{-4}$	$3.0x10^{-6}$	$3.9x10^{-12}$
	>5	0-25	$4.6x10^{-13}$	$9.7x10^{-11}$		$3x10^{-3}$	$2.4x10^{-6}$	--
		25-40	$4.6x10^{-13}$	$2.3x10^{-9}$		$1.23x10^{-3}$	$5.8x10^{-7}$	$2.6x10^{-25}$
		40-60	$4.6x10^{-13}$	$8.3x10^{-9}$		$3.10x10^{-4}$	$1.5x10^{-6}$	$1.9x10^{-13}$
	>25	0-25	$4.6x10^{-13}$	$3.9x10^{-14}$		$1.96x10^{-3}$	$1.3x10^{-6}$	--
		25-40	$4.6x10^{-13}$	$3.2x10^{-12}$		$1.23x10^{-3}$	$\sim2x10^{-6}$	$1.6x10^{-36}$
		40-60	$4.6x10^{-13}$	$2.5x10^{-11}$		$3.10x10^{-4}$	$1.7x10^{-6}$	$3.6x10^{-18}$
	>100 **	0-25	$3.6x10^{-16}$				$1.6x10^{-6}$	
		25-40	$3.6x10^{-16}$				$\sim2x10^{-6}$	
		40-60	$3.6x10^{-16}$				$1.7x10^{-6}$	
	field		22	3		20	22	29
	shop		12	9		20	30	22
	safe end		16	1		9	3	12
	dominant		3	2	3	2	2	4
system frequencies	>0	0-25	--	0.019	$4.44x10^{-4}$	$2.43x10^{-2}$		$<1.6x10^{-9}$
		25-40	--	0.048	$1.78x10^{-3}$	$1.17x10^{-2}$		$1.5x10^{-6}$
		40-60	--	0.030	$2.58x10^{-3}$	$4.82x10^{-3}$		$5.2x10^{-5}$
	>0.1	0-25	$4.0x10^{-8}$	$6.0x10^{-7}$	$7.80x10^{-5}$	$1.15x10^{-2}$		--
		25-40	$4.0x10^{-8}$	$8.5x10^{-6}$	$4.05x10^{-4}$	$2.62x10^{-3}$		$2.8x10^{-12}$
		40-60	$4.0x10^{-8}$	$1.8x10^{-5}$	$3.96x10^{-4}$	$7.10x10^{-4}$		$1.3x10^{-7}$
	>1.5	0-25	$4.8x10^{-11}$	$2.8x10^{-8}$	$7.80x10^{-5}$	$8.52x10^{-3}$		--
		25-40	$4.8x10^{-11}$	$4.4x10^{-8}$	$4.05x10^{-4}$	$2.46x10^{-3}$		$1.4x10^{-21}$
		40-60	$4.8x10^{-11}$	$1.2x10^{-7}$	$3.96x10^{-4}$	$6.20x10^{-4}$		$1.6x10^{-11}$
	>5	0-25	$1.4x10^{-12}$	$1.9x10^{-10}$		$6x10^{-3}$		--
		25-40	$1.4x10^{-12}$	$4.6x10^{-9}$		$2.46x10^{-3}$		$1.0x10^{-24}$
		40-60	$1.4x10^{-12}$	$1.7x10^{-8}$		$6.20x10^{-4}$		$7.6x10^{-13}$
	>25	0-25	$1.4x10^{-12}$	$7.9x10^{-14}$		$3.92x10^{-3}$		--
		25-40	$1.4x10^{-12}$	$6.4x10^{-12}$		$2.46x10^{-3}$		$6.5x10^{-36}$
		40-60	$1.4x10^{-12}$	$5.0x10^{-11}$		$6.20x10^{-4}$		$1.4x10^{-17}$
	>100 **	0-25	$1.1x10^{-15}$			$2.6x10^{-6}$		
		25-40	$1.1x10^{-15}$			$4x10^{-6}$		
		40-60	$1.1x10^{-15}$			$3.7x10^{-6}$		

times in reactor years, 1 calendar year ~ 0.8 reactor years
shaded areas are estimates based on alternative procedure
leak rates in thousands of gallons per minute
cross-hatched cells are beyond maximum leak capability for that pipe size
** also applicable to > 1,900,000 lpm (500 kgpm) for hot leg if sufficient diameter

F.5 References

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APPENDIX G

PIPING BASE CASE RESULTS OF VIC CHAPMAN

APPENDIX G

PIPING BASE CASE RESULTS OF VIC CHAPMAN

Summary of Benchmarking Analysis
Carried out Using 'RR-PRODIGAL'

G.1 General Background to RR-PRODIGAL

RR PRODIGAL is a basic fatigue failure probability model developed by Rolls Royce for the Naval Nuclear program. When analysing a weld, it first simulates the weld construction in order to determine a start of life defect distribution and density for both buried and surface breaking defects. A failure probability using standard linear elastic fracture mechanics methods is then evaluated for both the buried and surface breaking defects (assumptions about break through of buried defects to surface defects are based on the ASME criteria). Failure is achieved when the defect either exceeds the R6 failure criteria or simply grows through to the full thickness. The failure probability for all initial defects is then combined to form the total failure probability.

For non-weld areas, a probabilistic crack initiation analysis is carried out with a correlated crack growth analysis to failure. This correlation means that short times to crack initiation imply that a fast crack growth follows this initiation. There is no positive data to confirm or deny this proposition. It was chosen simply because it is pessimistic.

The modelling contains a routine to assess the growth of the defect around a welded pipe at the same time as the defect grows through the weld thickness, however, this part was not used in this assessment. RR-PRODIGAL does not, at present, contain a routine to evaluate the crack growth of a through wall defect around the outer surface of a pipe weld.

At failure, the model evaluates a critical through wall defect size based again on the R6 criteria.

At present RR-PRODIGAL does not contain a verified and validated assessment of the PWSCC degradation mechanism.

There are several publications that describe RR-PRODIGAL, which include a recent benchmarking exercise as part of a European initiative. References G.1 and G.2 should provide sufficient information for any readers wishing to obtain further information on this code.

G.2 Leak Rate Evaluation

When estimating RR-PRODIGAL leak rates through the final through wall defect in a pipe weld, evaluations were made using an elastic crack opening displacement (COD) analysis. However, it was felt that the uncertainties associated with assessing both the defect length around the pipe circumference as well as the COD needed for estimating the flow rate through the crack, were too great and too subject to ongoing development, to allow a suitable analysis of the leak rate. Thus, RR-PRODIGAL does not contain, within itself, a routine for evaluating the flow rate from the final defect size.

Instead, it was concluded that the leak rate from a through wall defect could be considered independently of the probability of the breach, i.e. the leak rate from the defect is not dependent on the probability of the defect cracking through the pipe wall. Note, however, that the COD, crack length, and hence leak rate is not independent of the mechanism that led to the failure, only the probability of the failure itself.

Within the Naval Nuclear program, computer programs have been developed to assess the leak rate from different defects based primarily on the 'SQUIRT' model. However, for consistency within this program, the data on leak rate against defect area provided by the USNRC were used, as shown in Figure G.1.

G.3 Procedure

The procedures used to develop the base case numbers are as follows:

1 Evaluate the basic fatigue failure probability using RR-PRODIGAL code using the transient data supplied[1].
2 Evaluate an elastic COD as a function of defect size.
3 Use expert judgement to extend this COD beyond the elastic limit.
4 Evaluate a mean defect cross-sectional area for a given defect size using its associated COD.
5 Evaluate the mean leak rate from a given defect size using the data supplied by the USNRC, see Figure G.1.

[Note for Steps 2, 3, 4 and 5 above a defect length is given. Thus, Steps 2, 3, 4 and 5 provide a mapping from a given defect size at failure to the mean leak rate in gpm, given this defect exists.]

6 Use expert judgement to assess the distribution of the defect length at failure.
7 Combine Steps 5 and 6 to obtain the conditional probability of a leak rate greater than the given leak rates for Categories 1 through 6. These categories being as follows;

Table G.1 Leak Category Leak Rates

	Leak Rate Greater than (gpm)	Log Leak Rate
Leak Category 1	100	2
Leak Category 2	1,500	3.2
Leak Category 3	5,000	3.7
Leak Category 4	25,000	4.4
Leak Category 5	100,000	5.0
Leak Category 6	500,000	5.7

8 Combining the conditional probability of Step 7 with the basic fatigue failure probability in Step 1 gives the required final probability of a leak greater than each of the categories.

G.4 Example Base Case Analysis

As a way of demonstrating the procedure given above, the results for the 14-inch Surge Line elbow are reproduced in this section. Two situations are considered, the elbow and the adjacent weld. The transients were based on data supplied and are reproduced in Attachment G.1

G.4.1 Probability of Failure Surge Line Elbow – Base Case

This is a failure from base material and so the analysis assumed a fatigue based crack initiation followed by crack growth to failure. As stated earlier, the crack initiation and crack growth are assumed to be positively correlated. This assumption assumes that if the properties of the base material are such as to lead to an early crack initiation, it is very possible that these same properties

[1] This information needed to be supplied because the transient experience for the Naval Nuclear program is a) confidential and b) not applicable to commercial plants.

could result in a subsequently fast crack growth rate. The results of this analysis are shown in the following table:

Table G.2 Results for PWR Surge Line Elbow Base Case Analysis

Time (years)	Cumulative Probability of Failure
25	6.1×10^{-6}
40	7.8×10^{-6}
60	9.4×10^{-6}

RR-PRODIGAL gave the critical through wall defect length, based on the R6 criterion, as 14 inches.

G.4.2 Probability of Failure Surge Line Weld

The surge line elbow weld was analysed at a 60-year life assuming the same cyclic conditions as for the elbow itself, but with the stresses factored down by 20 percent as suggested at the Elicitation Base Case Review Meeting on June 4 and 5, 2003 in Bethesda, Maryland. The two hydro cases were, however, maintained at their original values.

In this analysis, RR-PRODIGAL first simulates the weld construction, including any build inspections, to establish the start of life defect density and distribution for both buried and surface breaking defects. As stated earlier, conditional failure probabilities are assessed for both situations and combined to give the final failure probability.

The failure probability evaluated for this case was:

Table G.3 Results for PWR Surge Line Weld Analysis

Cumulative Probability of Failure at 60 years	1.3×10^{-4}

It can be seen that this failure probability is over an order of magnitude higher than the base case. This is due to the difference between having to initiate a defect and then grow this defect to failure, and having the probability of pre-existing defects in the weld. The base case values from the base material failure as reflected in Table G.2, i.e. crack initiation leading to failure, have been used in Table D.1 in the main body of this report. Note, however, that the values reported in Table G.2 are cumulative probabilities of failure in 25, 40, and 60 years whereas the values reported in Table D.1 of the main body are frequencies. Consequently, the Table G.2 values need to be divided by 25, 40, and 60 years, respectively to facilitate any comparisons. Furthermore, the values in Table D.1 are for leak rates greater than the threshold leak rates, i.e., 380 lpm (100 gpm) while the values in Table G.2 reflect the totals.

G.4.3 COD and Leak Rate for a Given Defect Size

Having established a basic failure probability, the COD can be evaluated independently of this probability. Once this is established, the leakage area of the defect follows, and given this leak area, the flow rate can be evaluated using the information from Figure G.1. A mean power law was then used to calculate the mean flow rate given a leakage area. The table below gives the elastic COD values evaluated for this case and the resultant flow rate.

Table G.4 Elastic COD and Resultant Leak Rates for a Given Defect Length

Defect Length (inches)	Elastic COD (inches)	Flow rate (gpm)
1.98	0.0025	17
3.96	0.0049	48
5.94	0.0074	92
7.41	0.01	145
9.89	0.012	200
11.87	0.015 (Invalid Result)	270

Interpolating between the results in Table G.4, it can be seen that a defect approximately 160 mm (6.2 inches) long, which is approximately 15 percent of the pipe circumference, results in the first leakage category of 380 lpm (100 gpm).

Clearly it is the behaviour of the defect beyond the elastic range that is of interest for the larger leak categories. If it were to be assumed that at the critical defect size the pipe would simply tear, in an unstable manner, to result in a Double Ended Guillotine Break (DEGB) failure, then the leak rate would simply jump from a Category 1 failure to the gpm associated with the DEGB. In this case that would be 250,000 lpm (65,000 gpm) or a Category 4 leak. The probability of a Category 2 leak rate would then be the same as a Category 3, which would be the same as the Category 4!

Such an assumption could be considered valid. However, in this work, it was assumed that the defect would continue opening in a stable, but plastic manner. Whilst models do exist to evaluate the plastic deformation of defective pipes, no such model was used in this analysis. Instead expert judgement was used to assess how the COD would develop beyond this elastic point, and at what defect size the pipe would finally tear into a DEGB failure. The results of this judgement are shown in Figure G.2. The area of leakage can then be calculated, and the leak rate, given a defect length also follows. The resulting gallon per minute flow rate, for this example, is shown in Figure G.3.

The failure probability gives the basic probability of a breach of the pressure boundary. Figure G.3 shows the leak rate in gallons per minute, given a defect of a given length. In order to obtain the probability of a leak rate greater than 'X' gallons per minute, it only remains to provide a distribution of the defect size at the moment of failure.

G.4.4 Defect Distribution and Leak Rate at Failure – No Leak Detection

First consider the case with no leak detection. For this case the instantaneous size of the defect, and its associated COD, at the moment of snap through to a breach of containment is required. As an example, if the aspect ratio were of the order of 8/1 at snap through, then given a pipe wall thickness of about 36 mm (1.4 inches), the defect length would be approximately ten or eleven inches long. If it were then pessimistically assumed that this was the full through wall defect length, then the instantaneous leak rate would be just above (actually about twice) our 'Category 1' failure criteria of 380 lpm (100 gpm). Thus, the probability of a leak rate greater than Category 1 becomes the basic probability of failure times the probability that the defect at snap through was greater than 250 mm (10 inches), i.e., the defect had an aspect ratio at snap through of about 8/1 or greater. It then follows, from Figure G.3, that in order to exceed the Category 2 leak rate, the instantaneous defect size at snap through would have to be greater than 380 or 405 mm (15 or 16 inches), i.e., the defect had an aspect ration of about 11/1. Furthermore, the defect snapped straight open to the fully plastic COD.

As stated earlier, RR-PRODIGAL has the capability of simulating the crack growth both around and through the pipe wall. However, this is not generally used as the solutions require a detailed knowledge of the stress distribution around the pipe, including any weld residual stress, and generally such knowledge is not well enough defined. Thus, expert judgement was again used. The expert judgement required is to generate a defect distribution at the moment the defect snaps to the COD of Figure G.3, assuming no leak detection.

This base case is for the surge line elbow and it has been assumed that most of the deformation and high stress will result from large bending moments at the elbow. It was felt that this would initiate a defect preferential on the hogging side of the elbow, and promote a crack to grow through the wall thickness on this side of the elbow. This would then imply that the crack growth around the pipe diameter would be restricted. Figure G.4 represents the distribution decided upon for this analysis. This distribution shows the most likely defect length to be up to about 250 mm (10 inches), which is about a quarter of the way around the pipe circumference. The probability of the defect being over halfway around the pipe is seen as a rare event, being about 0.025 or a 1 in 40 chance. If the loading were not dominated by bending, then this distribution would probably be judged to be flatter, with perhaps a 1 in 10 chance of being greater than halfway round the pipe circumference.

Combining Figures G.3 and G.4 gives the conditional probability of a leak greater than a given leak rate. This final plot is given in Figure G.5 and is combined with the basic failure probability to derive the values given in Table D.1 in Section D of the main body of this report.

G.4.5 Defect Distribution and Leak Rate at Failure – With Leak Detection

In the previous section it was assumed that the defect would instantaneously snap open to the full COD associated with its length at the moment the pressure boundary was breached. In reality this will probably not happen. Instead, the very large defects, which are those of interest, will probably grow to different through wall depths at different points around the length. Thus, much smaller surface defects would begin to breach the boundary at different points around the defect. The COD of these small defects would then remain elastic until the whole defect progressed to the surface. In this scenario, the leak from the defect would start very small and grow, slowly at first and then probably very quickly before snapping open to the fully plastic COD.

During this time of surface crack combination, the leak rate may exceed the value at which the operators shut the reactor down to a safe state in order to investigate the leak. Provided this occurs before the crack reaches a critical size, i.e., before the leak rate moves very quickly to the final leak state. Whilst the high leak rate may still occur, the plant would be in a safe condition. This can be seen as leak detection.

This probability of leak detection is almost certainly associated with the length of the defect that is itself related to the rate of leakage in the previous section. Thus, expert judgement was again used to introduce a factor, based on the leak rate, which would represent this probability of leak detection. Figure G.6 shows this plot as a function of leak rate.

From this plot it can be seen that the reduction factor for Category 1 (380 lpm [100 gpm]) is about five, rising to a factor of about fifty at Category 6 (1,900,000 lpm [500,000 gpm]).

G.5 Effect of In Service Inspection

An assessment of the effect of ISI was carried out for the surge line elbow weld, the defect distribution and density being those generated by RR-PRODIGAL, see section G.4.2. A Probability of Detection (POD) curve was defined by the following equation:

$$f_{\mathrm{POD}} = \Phi\left(c_1 + c_2 \ln\left(\frac{a}{t}\right)\right) \qquad \text{where } c_1 = 1.526 \text{ and } c_2 = 0.533 \qquad \text{(G.1)}$$

This POD is shown in Figure G.7, and it can be seen that this sets the probability of detection at about 90 percent for defects 70 percent of the way through the wall thickness. This was felt to be representative of inspections carried out to date, but for future inspections that conform to modern standards, this POD could be much better.

The results are shown in the table below and in Figure G.8 for various ISI intervals.

Table G.5 Reduction Factors Due to ISI

ISI case	Cumulative Probability of Failure at 60 years	Factor for General Use
No ISI	1.3×10^{-4}	1
0 years (PSI)	4.2×10^{-5}	3
10 years	3.8×10^{-5}	3.4
10, 20 years	1.3×10^{-5}	10
10, 20, 30 years	6.5×10^{-6}	20
10, 20, 30, 40 years	4.8×10^{-6}	27
10, 20, 30, 40, 50 years	4.7×10^{-6}	28

These results suggest that even with this quite low inspection capability, and for a weld with a high failure probability, reductions of a decade can be achieved with two or three inspections during the life of the plant. It also indicates that going beyond three inspections gives little extra return.

An interesting conclusion from this figure would be that if a fourth inspection is carried out at the end of a forty year period, then, provided this inspection was clear, there would be little gain from an inspection at fifty years for a total life of sixty years! However, at this stage such a conclusion can only be taken as tentative and would require more investigation.

G.6 References

G.1 NUREG/CR-5505 PNNL-11898 'RR-PRODIGAL – A Model for Estimating the Probability of Defects in Reactor Pressure Vessel Welds.

G.2 NURBIM (Nuclear Risk-Based Inspection Methodology) WP4. Published by the European Commission under the EURATOM programme.

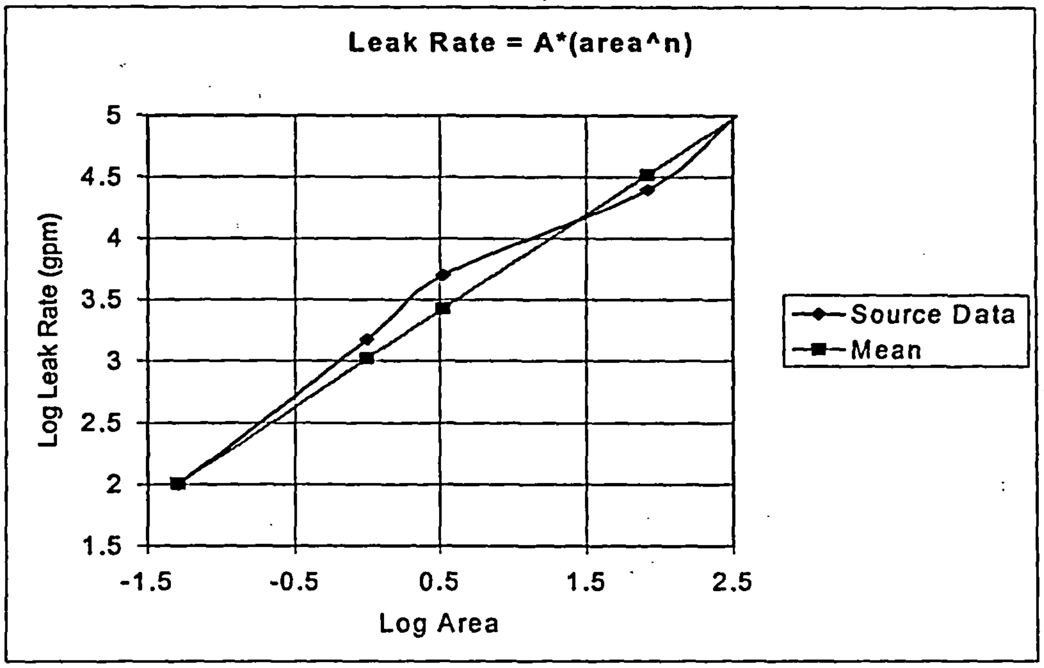


Figure G.1 Leak Rate as a Function of Leakage Area (Data Supplied by USNRC)

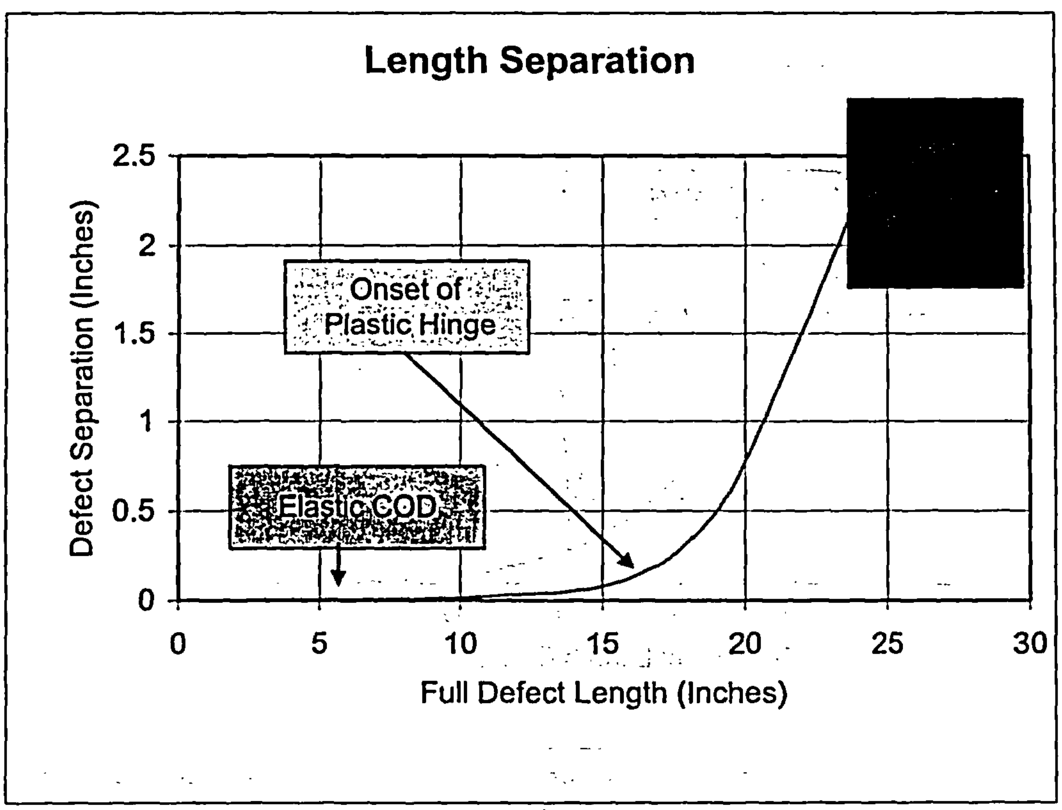


Figure G.2 Estimated Defect Separation (COD) Based on Expert Judgment as a Function of Defect Length Assuming Plastic Deformation

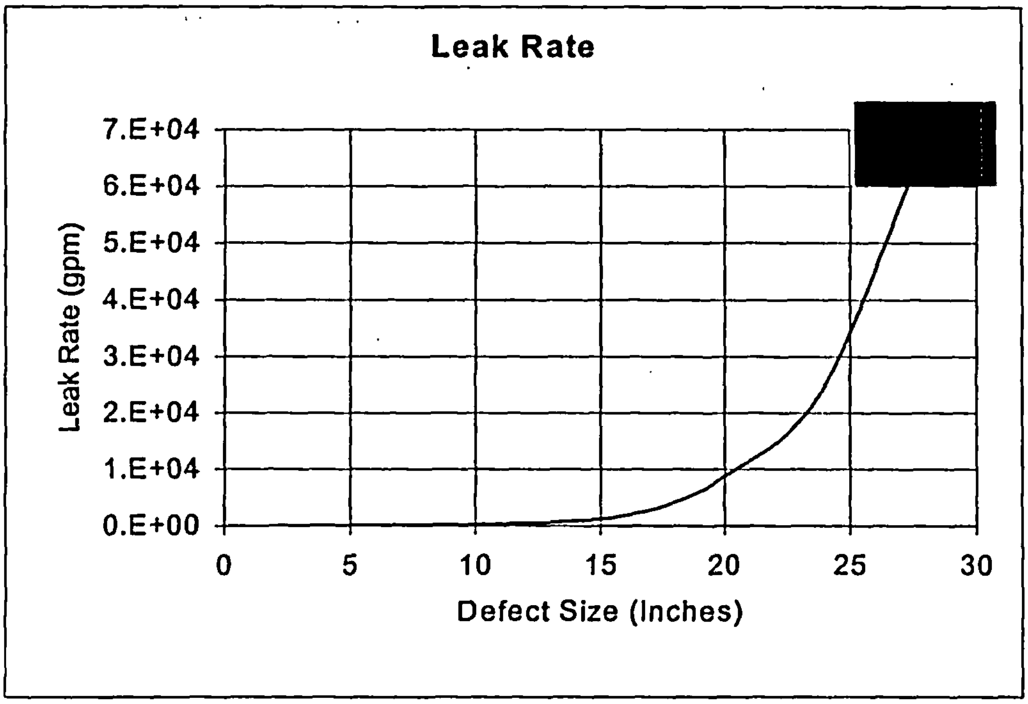


Figure G.3 Estimated Leak Rate Versus Defect Length Based on Expert Judgment

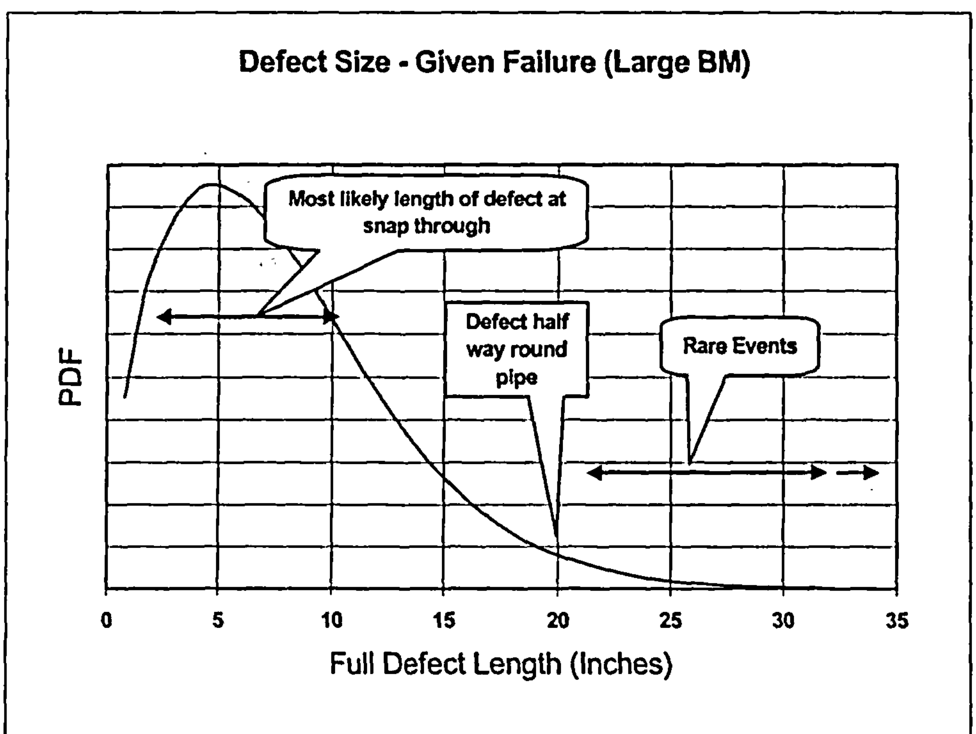


Figure G.4 Probability of the Existence of a Defect of a Certain Length for Surge Line Base Case

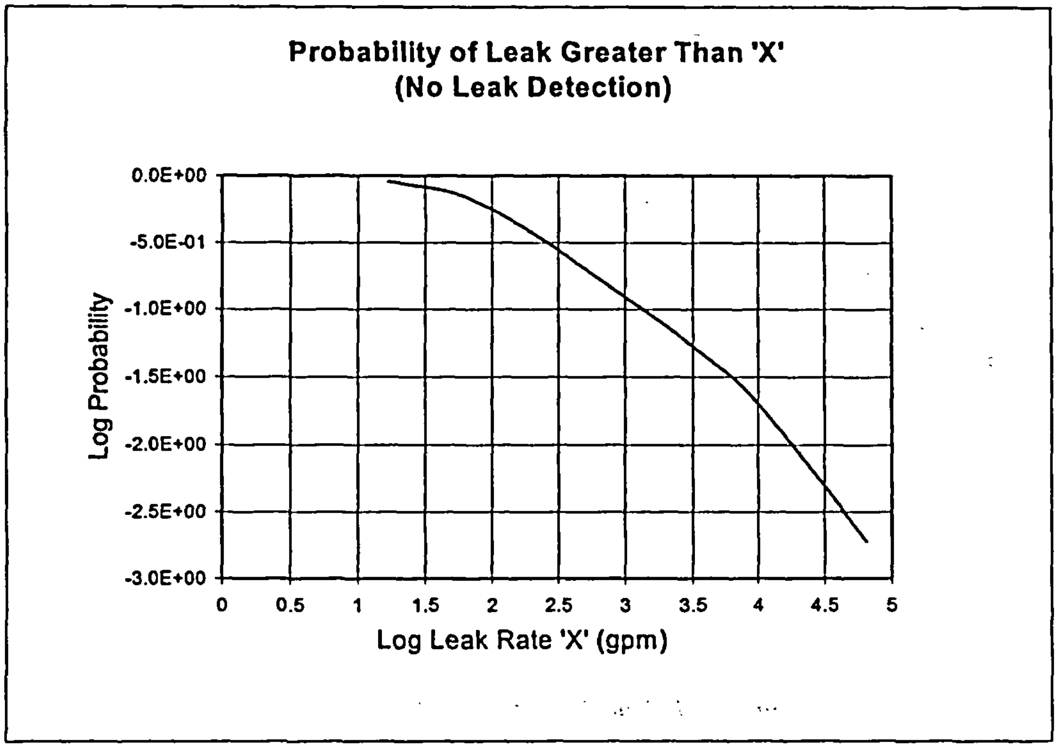


Figure G.5 Conditional Probability of a Leak of a Given Size

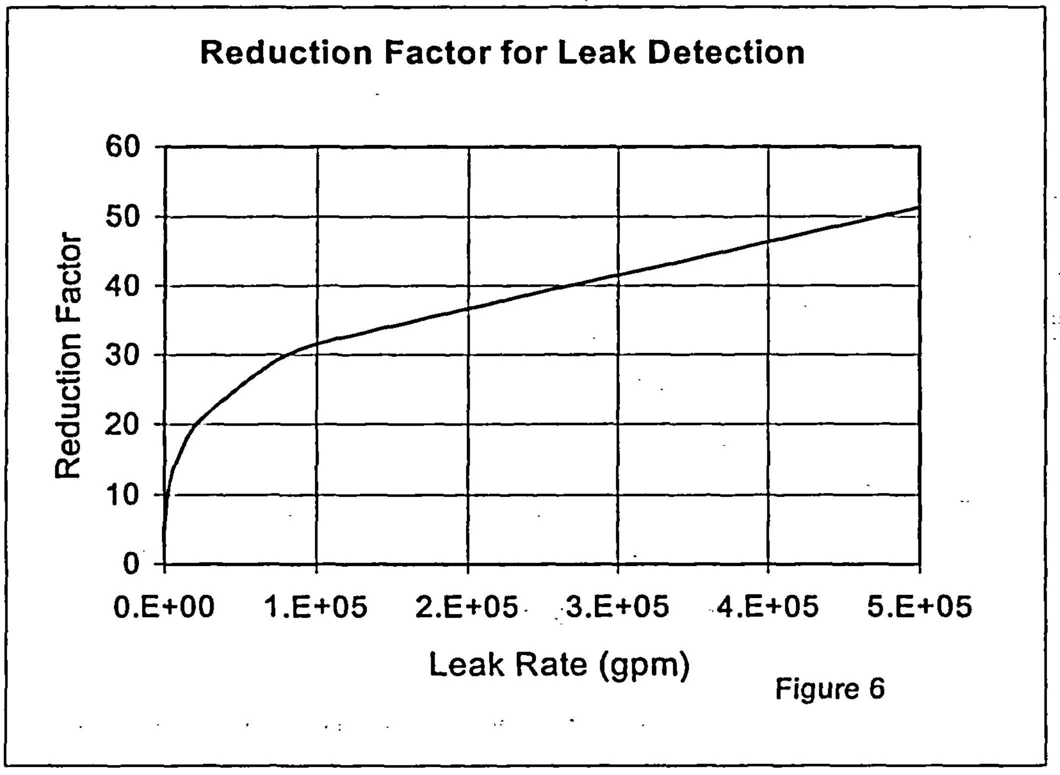


Figure G.6 Reduction Factors for Leak Detection Based on Expert Judgment

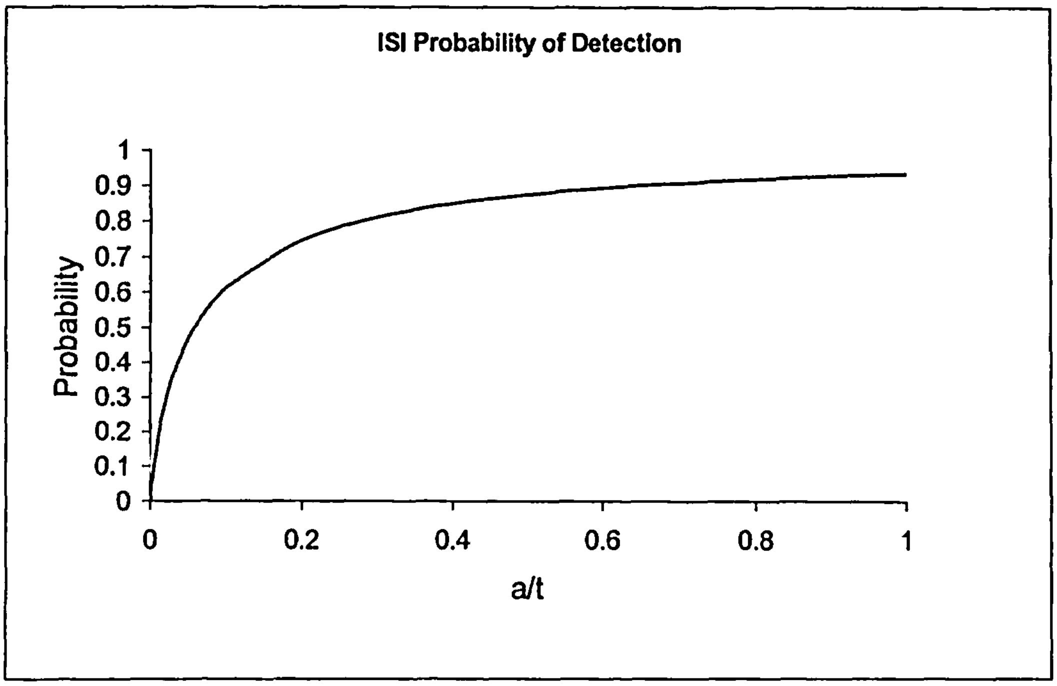


Figure G.7 Probability of Detection Curve

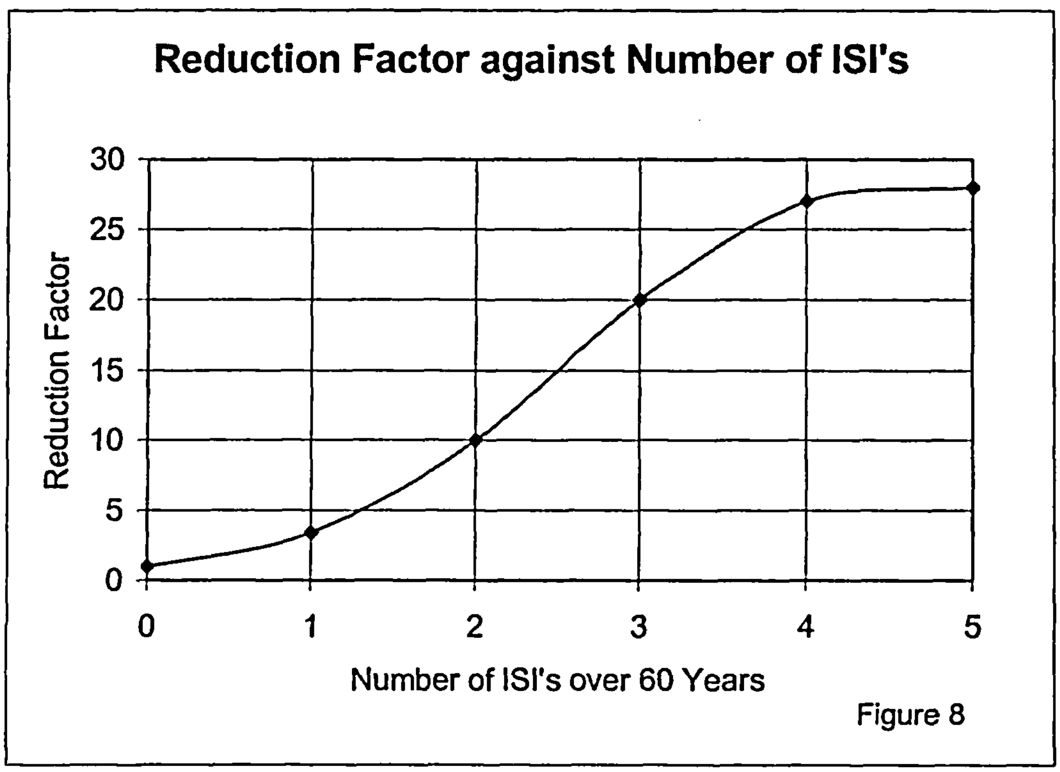


Figure G.8 Reduction Factors Due to ISI as a Function of the Number of Inspections

ATTACHMENT 1 TO APPENDIX G (ATTACHEMENT G.1) FROM NOTE 'SUMMARY OF BASE CASES STRESSES' APRIL 2003

Summary of Stress Cycles for Surge Line Elbow
(No seismic stresses)

Load Pair	Amplitude (ksi)	Number/40 years
HYDRO-EXTREME	190.17	6
9B-HYDRO	149.86	4
8A-UPSET 4	140.42	14
9B-UPSET4	139.43	10
8B-UPSET4	105.89	14
9A-UPSET4	105.13	2
9A-LEAK	103.86	12
8F-18	63.40	68
9C-11	63.38	68
9F-LEAK	63.37	68
8C-LEAK	63.37	35
2A-8C	62.30	33
8G-18	52.38	22
8G-17	52.35	90
9D-11	52.35	22
2A-8D	51.20	72
8H-9G	51.18	400
8G-UPSET3	51.00	30
9D-12	50.96	50
8G-12	50.96	40
8G-16	50.93	90
8G-9H	50.92	128
2A-8E	40.10	90
8H-9H	40.09	100
9H-10A	40.09	272
9E-13	39.82	90
3A-10A	33.10	4120
6-10A	33.10	200
3B-10A	33.10	4120
7-10A	33.10	4580
2B-SLUG1	32.87	100
2B-SLUG2	32.87	500
5-10A	29.90	9400
4A-10A	29.90	17040
4B-10A	29.90	17040
2B-10A	20.60	14400
2A-10A	20.60	14805
10A-UPSET1	20.59	70
10A-UPSET5	20.59	30
10A-UPSET6	20.59	5
10A-UPSET2	20.59	95
1B-10A	20.59	1533
1B-10B	20.00	87710

Many of the high stress contributors in Tables 2 and 3 are from rapid excursions of the coolant temperature. The largest stress amplitude (half the peak-to-peak) is 1,310 MPa (190 ksi), so the stresses are large (but localised). These are the stresses at the peak stress location, which is not at weld. The spatial stress gradients (both along the surface and into the pipe wall) are required for a thorough analysis. The radial gradient (into the pipe wall) can be estimated by the following procedure:

- Cyclic stresses associated with seismic loads were treated as 100 percent uniform stress.
- Cyclic stresses greater than 310 MPa (45 ksi) were treated as having a uniform component of 310 MPa (45 ksi), and the remainder were assigned to the gradient category.
- For those transients with more than 1,000 cycles over a 40-year life, it was assumed that 50% of the stress was uniform stress and 50% a through-wall gradient stress. In addition, for these transients, the uniform stress component was not permitted to exceed 70 MPa (10 ksi).

The gradient stress mentioned above is assumed to vary through the thickness as

$$\sigma(\xi) = \sigma_o\left(1 - 3\xi + \frac{3}{2}\xi^2\right) \tag{G.2}$$

In this equation, σ_o is the stress at the inner wall of the pipe, $\xi = x/h$, x is the distance into the pipe wall from the inner surface, and h is the wall thickness. The stresses and cycles are high enough that fatigue crack initiation is important, which has been considered in Reference 6. Reference 6 shows a probability of 0.981 of a leak in 40 years for this component. The LOCA probabilities will be less. The use of the gradient along the surface will reduce this.

As mentioned above, the more thorough results that include a better estimate of the radial gradient and also consider the spatial gradient along the surface are available, and could be used for the base case calculations.

APPENDIX H

DESCRIPTION OF NON-PIPING DATABASE

APPENDIX H

DESCRIPTION OF NON-PIPING DATABASE

H.1 Background

The non-piping database has been compiled with the intention that it will serve as one source of information supporting the development of estimates of LOCA frequencies attributable to non-pipe components. The data has been obtained from two primary sources. First, a search of licensee event reports (LERs) was made to identify those instances where failures[1] of non-pipe components of the primary reactor coolant pressure boundary were reported to the NRC. The second source of information is data that has been incidentally collected on non-pipe components during the development and maintenance of the pipe-based OECD and SLAP databases. LER events compose the majority of the events in the database (see Attachment A of this appendix for a description of the LER reporting requirements).

The database is accessible in two formats, *Table* and *Forms*. The *Table* named "Failure Data" lists the data in a spreadsheet type of format where each line of the table contains one data record and each column contains the various fields that make-up the records. In the *Forms* format, only one record of the "Failure Data" is displayed at a time, but in a manner that allows all of the fields to be view at the same time. Both formats display the same data, only the presentation is different. Also, sorting and filtering of the data can be done in both views.

H.2 Approach

A search of licensee event reports (LERs) was performed (see Attachment B for the specific search criteria) using the Sequence Coding and Search System (SCSS). This search returned 1,036 LERs. Each LER was reviewed and coded in the Non-Pipe database. The database structure is based on information generated during the elicitation meetings. In particular, the component, piece part, and degradation mechanism are all identified using the tables documented in the elicitation meeting notes. Other fields of the database were developed and defined as judged appropriate.

The initial screening of the 1,036 LERs to remove those that were judged to not be applicable reduced the total number to 213. As discussed in Attachment B, the data search simply looked for leak and crack events associated with primary coolant systems. This conservative search included LERs that identified *potential* and *possible* leak and crack events (e.g., a problem with ECCS such that the plant would not respond as designed to a loss of coolant accident. Screening out the potential failures resulted in a reduction to 213 records. A further 34 records were removed since they identified problems with pipes or seals. Then 37 records were added that had been collected previously during the development of the OECD and SLAP databases. This results in a current total of 216 records.

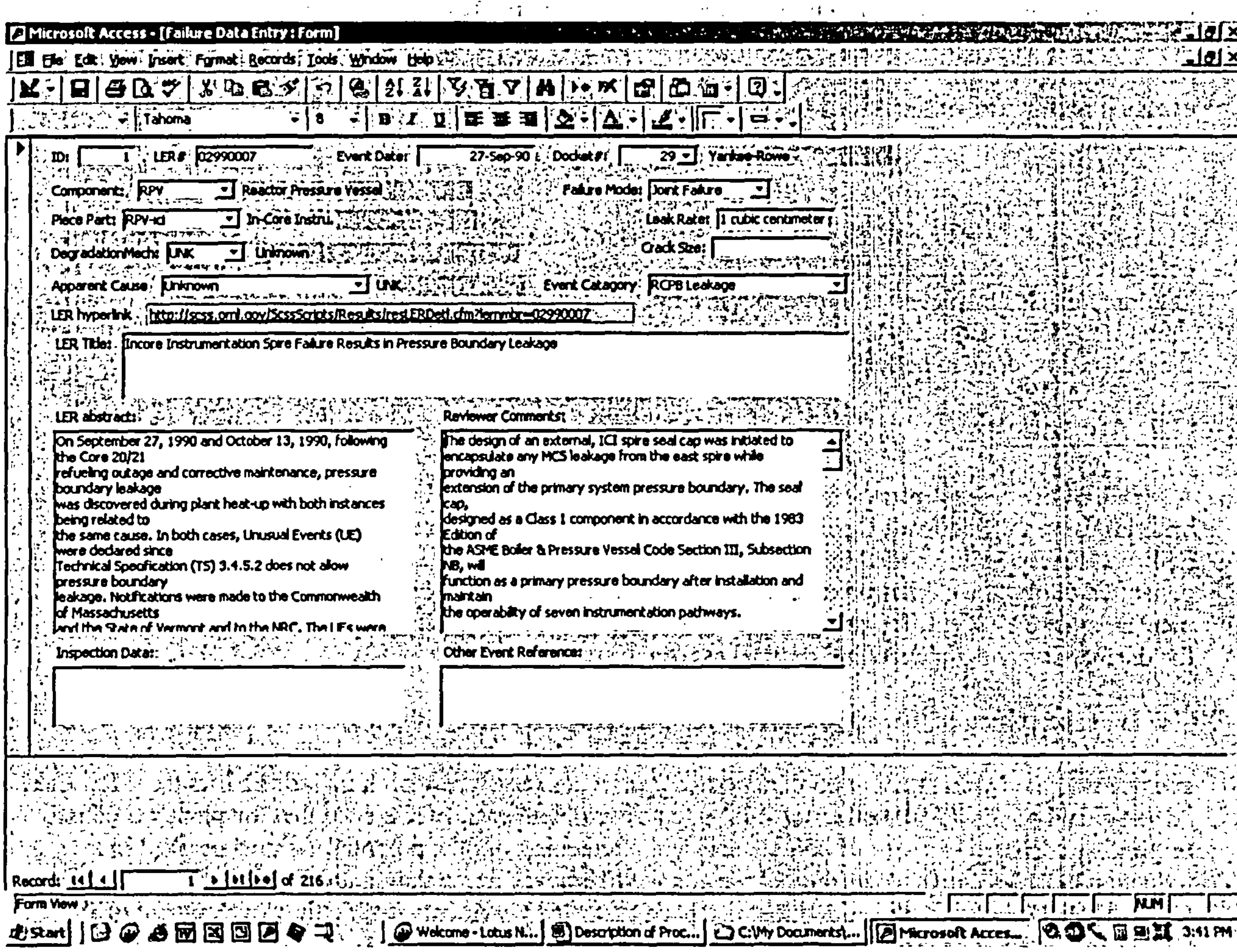
H.3 Description of Data

This process results in 216 data records that document crack (both partial and full) and leak events associated with non-pipe primary coolant system components. This dataset can be considered complete for U.S. NPP operation from 1990 through 2002, inclusive, in as much as the LER reporting requirements (Attachment A) can be relied upon to generate complete reporting. Additionally, the dataset does include

[1] Failures are classified using four categories: partial through-wall cracks, through-wall cracks without a significant leak rate (typically indicated by a boric acid deposit), leaks, and joint failures (i.e., non-welded connection).

a limited amount of data from outside this time frame and from non-U.S. plants. Nevertheless, the dataset can be considered to be internally consistent, that is, the various components, failures and degradation mechanisms are believed to be represented equally such that relative ratios (if not the absolute frequencies) can be assumed to be reasonably accurate. Several of the database records represent multiple cases of degradation or failure. Attachment C includes a sample of multiple event records, including a discussion on how to estimate flaw frequencies from the observed events as recorded in the database.

The figure below (*Forms* view of the database) identifies the various fields maintained by the database. For additional detail on those records based on LERs, the LER hyperlink can be clicked to retrieve the full LER (internet access is require for this).



H.4 Limitations

As with many reliability databases, the completeness of the data is always an issue. While relative frequencies (e.g., percentage distribution of events by component or degradation mechanism) might be reasonably accurate, the accuracy of any absolute frequency (e.g., events per year) calculations will depend directly on how complete the data are. That is, have all events that have occurred been included in the database? In the current situation this question has two parts. First, have all relevant event been discovered? Second, of the discovered events, have they been reported (via LER)?

The completeness issue is probably more of an issue for the partial through-wall cracks than it is for the more severe failures. There are two causes for this concern. One is ambiguity in the interpretation of the LER reporting requirement (Attachment A), and the second and probably primary cause is simple lack of detection. While effort is made to make the LER reporting requirements as clear as possible, the "seriously degraded" aspect of 50.73(a)(2)(ii)(A) is difficult to quantify. How far does a crack have to extend to seriously degrade the primary pressure boundary? It is possible that some cracks are being detected and repaired (which might be considered normal plant maintenance rather than corrective action), without being reported as a LER. These events have not been captured in this search. However, detection likelihood is probably a bigger reason for coverage deficiencies of part-through wall flaws. A leak (or a non-leaking through-wall crack) is simply more likely to be detected. This issue is clearly illustrated by events in the data in which a detected leak prompted the plant to do a thorough inspection that found partial through-wall cracks. If the leak had not occurred and motivated the inspection, the partial through-wall cracks would not have been found.

H.5 Selected Non-Pipe Database Summary Tables

The following tables list some summary data from the non-pipe database.

Table H.1 Non-Pipe Event Counts by Component and Degradation Mechanism (for Each Plant Type)

			Degradation Mechanism (see legend)							
Plant Type	Component		MA	FDR	SCC	LC	MF	TF	FS	UNK
BWR	RPV	10		1	9					
		100%	0%	10%	90%	0%	0%	0%	0%	0%
	Valve	1					1			
		100%	0%	0%	0%	0%	100%	0%	0%	0%
	Pump	2		2						
		100%	0%	100%	0%	0%	0%	0%	0%	0%
	Totals	13	0	3	9	0	1	0	0	0
	Adjusted*	17	0.5	3.5	9.5	0.5	1.5	0.5	0.5	0.5
		100%	3%	21%	56%	3%	9%	3%	3%	3%
			MA	FDR	SCC	LC	MF	TF	FS	UNK
PWR	Pzr	28	1	3	23	1				
		100%	4%	11%	82%	4%	0%	0%	0%	0%
	RPV	42		5	27	5				5
		100%	0%	12%	64%	12%	0%	0%	0%	12%
	Valve	3	1		1	1				
		100%	33%	0%	33%	33%	0%	0%	0%	0%
	SG	124	2	29	85				3	5
		100%	2%	23%	69%	0%	0%	0%	2%	4%
	Pump	2		2						
		100%	0%	100%	0%	0%	0%	0%	0%	0%
	Instr nozzles	4			4					
		100%	0%	0%	100%	0%	0%	0%	0%	0%
	Totals	203	4	39	140	7	0	0	3	10
	Adjusted*	207	4.5	39.5	140.5	7.5	0.5	0.5	3.5	10.5
		100%	2%	19%	68%	4%	0%	0%	2%	5%

DM	DM Description
MA	Material Aging
FDR	Fabrication Defect and Repair
SCC	Stress Corrosion Cracking
LC	Local Corrosion
MF	Mechanical Fatigue
TF	Thermal Fatigue
FS	Flow Sensitive (includes FAC and E/C)
UNK	Unknown

* A half failure (0.5) was added to all degradation mechanism (DM) totals to force the representation of all DMs.

Table H.2 Summary of Non-Pipe Database by Plant Type and Piece Part (see Table H.3 for Legend of Piece Part Acronyms)

Plant Type	Component	Piece Part	No. Records	Calendar Year																			
				78	84	86	87	88	89	90	91	92	93	94	95	96	97	98	99	00	01	02	03
BWR	Pipe	Pipe-w	1								1												
BWR	RecP	RCP-bdy	1												1								
BWR	RecP	RecP-hx	1									1											
BWR	RPV	RPV-crc	4								1			1							1	1	
BWR	RPV	RPV-crd	4											1				2			1		
BWR	RPV	RPV-hbt	1								1												
BWR	RPV	RPV-noz	1																	1			
PWR	LIV	FLG-fbs	1						1														
PWR	LIV	LIV-bon	2																	1	1		
PWR	Pipe	Pipe-w	4												1				1	1	1		
PWR	Pzr	Pzr-bbs	4							1	1			1				1					
PWR	Pzr	Pzr-hsl	8				2		1					1						2		1	1
PWR	Pzr	Pzr-noz	16			1			1	1		2	3	1	1		1	2	1	1		1	
PWR	RCP	RCP-noz	2															2					
PWR	RPV	RPV-crc	13							1			1					2			7	2	
PWR	RPV	RPV-crd	13		1	1		1		1				1		1		1	1		3	1	1
PWR	RPV	RPV-hdt	3				1						1									1	
PWR	RPV	RPV-ici	2							1				1									
PWR	RPV	RPV-noz	2																				2
PWR	RPV	RPV-pen	9				1				1	1						1	1	1	1	1	1
PWR	SG	Pipe-w	1																1				
PWR	SG	SG-mwb	1					1															
PWR	SG	SG-noz	3									1		1							1		
PWR	SG	SG-tub	119	1	1			1	1	10	13	9	10	10	13	11	11	10	6	5	4	3	
		Totals:	216	1	2	2	4	3	4	15	18	14	15	18	16	12	12	21	11	12	20	11	5

Table H.3 Piece Part Legend

PP-ID	Piece Part
FLG-fbs	Flange Bolts
LIV-bbs	Bonnet Bolts
LIV-bdy	Valve Body
LIV-bon	Bonnet
MSIV-bbs	Bonnet Bolts
MSIV-bdy	Valve Body
MSIV-bon	Bonnet
Pipe-w	Weld
PIV-bbs	Bonnet Bolts
PIV-bdy	Valve Body
PIV-bon	Bonnet
Pzr-bbs	Pzr valve bonnet bolts
Pzr-brv	Bolted Relief Valves
Pzr-hsl	Heater Sleeves
Pzr-mwb	Manway Bolts
Pzr-mwy	Manway
Pzr-noz	Pzr Nozzles
Pzr-rvb	Relief Valve Bolts
Pzr-shl	Shell
RCP-bdy	Pump Body
RCP-fwh	Flywheel
RCP-noz	Pump Nozzles
RCP-sel	Pump Seals
RecP-bbs	Bonnet Bolts
RecP-bdy	Pump Body
RecP-hx	Pump cooler
RecP-noz	Pump Nozzles
RecP-sel	Pump Seals
RPV-crc	CRDM connections
RPV-crd	CRDM
RPV-hbt	Head Bolts
RPV-hdb	Head (bottom)
RPV-hdt	Head (top)
RPV-ici	In-Core Instru.
RPV-noz	RPV Nozzles (incl. Instr.)
RPV-pen	Penetrations
SG-mwb	Manway Bolts
SG-mwy	Manway
SG-noz	SG Nozzles
SG-shl	Shell
SG-tbs	Tube Sheet
SG-tub	Tube

Table H.4 Summary of Non-Pipe Database by Plant Type, Piece Part and Failure Mode

Plant Type	Piece Part	Failure Mode[a]	No. Records	Calendar Year 78	84	86	87	88	89	90	91	92	93	94	95	96	97	98	99	00	01	02	03
BWR	RCP-bdy	Crack-Part	1												1								
BWR	RPV-crc	Crack-Part	1								1												
BWR	RPV-crd	Crack-Part	1															1					
BWR	RPV-noz	Crack-Part	1																	1			
		Subtotal:	4								1				1			1		1			
BWR	RPV-crc	Joint Failure	1											1									
BWR	RPV-hbt	Joint Failure	1								1												
		Subtotal:	2								1			1									
BWR	Pipe-w	Leak	1								1												
BWR	RecP-hx	Leak	1									1											
BWR	RPV-crc	Leak	2																		1	1	
BWR	RPV-crd	Leak	3											1				1			1		
		Subtotal:	7								1	1		1				1			2	1	
		BWR Totals:	13								3	1		2	1			2		1	2	1	
PWR	RPV-crc	Crack-Part	1																		1		
PWR	RPV-hdt	Crack-Part	3				1						1									1	
PWR	RPV-pen	Crack-Part	2									1											1
PWR	SG-noz	Crack-Part	1									1											
PWR	SG-tub	Crack-Part	99							7	13	5	9	9	12	9	10	8	6	4	4	3	
		Subtotal:	106				1			7	13	7	10	9	12	9	10	8	6	4	5	4	1
PWR	LIV-bon	Joint Failure	2																	1	1		
PWR	Pzr-bbs	Joint Failure	2											1				1					
PWR	RPV-ici	Joint Failure	2							1				1									
		Subtotal:	6							1				2				1		1	1		
PWR	Pipe-w	Crack-Full	4												1				2		1		
PWR	Pzr-hsl	Crack-Full	4											1						2		1	
PWR	Pzr-noz	Crack-Full	8									2		1	1			1	1	1		1	
PWR	RCP-noz	Crack-Full	1															1					
PWR	RPV-crc	Crack-Full	4																		4		
PWR	RPV-crd	Crack-Full	2											1							1		
PWR	RPV-pen	Crack-Full	3															1			1	1	

Plant Type	Piece Part	Failure Mode[a]	No. Records	78	84	86	87	88	89	90	91	92	93	94	95	96	97	98	99	00	01	02	03
				Calendar Year																			
PWR	SG-noz	Crack-Full	1																		1		
PWR	SG-tub	Crack-Full	2									1			1								
		Subtotal:	29									3		3	3			3	3	3	8	3	
PWR	FLG-fbs	Leak	1						1														
PWR	Pipe-w	Leak	1																	1			
PWR	Pzr-bbs	Leak	2							1	1												
PWR	Pzr-hsl	Leak	4				2		1														1
PWR	Pzr-noz	Leak	8			1			1	1			3				1	1					
PWR	RCP-noz	Leak	1															1					
PWR	RPV-crc	Leak	8							1			1					2			2	2	
PWR	RPV-crd	Leak	11		1	1		1		1						1		1	1		2	1	1
PWR	RPV-noz	Leak	2																				2
PWR	RPV-pen	Leak	4				1				1								1	1			
PWR	SG-tub	Leak	18	1	1			1	1	3		3	1	1		2	1	2		1			
PWR	SG-noz	Leak	1											1									
		Subtotal:	61	1	2	2	3	2	4	7	2	3	5	2		3	2	7	2	3	4	3	4
PWR	SG-mwb	NA	1					1															
		PWR Totals:	203	1	2	2	4	3	4	15	15	13	15	16	15	12	12	19	11	11	18	10	5

a. Crack-Part – Partial through-wall crack
Crack-Full – Complete through-wall crack, but no active leak, typically indicated by boric acid deposit
Leak – Measurable leak
Joint Failure – Failure of a bolted connection

ATTACHMENT A TO APPENDIX H – LER REPORTING REQUIREMENTS

The database relies upon licensee event reports (LERs) submitted by plants under the requirement of 10 CFR 50.73. Of the LERs reviewed for this effort, the two most commonly cited reporting requirements (each LER must reference the requirement that necessitates the LER.) are 50.73(a)(2)(i)(B) and 50.73(a)(2)(ii)(A). These are described below.

50.73(a)(2)(i)(B) – Any operation or condition which was prohibited by the plant's Technical Specifications. Westinghouse Standard Tech Specs (NUREG-1431, Vol. 1, Rev. 2, June 2001, Section 3.4.13) related to RCS leakage are as follows.
RCS operational leakage shall be limited to:

a. No pressure boundary leakage
b. 3.8 lpm (1 gpm) unidentified leakage
c. 38 lpm (10 gpm) identified leakage
d. 3.8 lpm (1 gpm) total primary to secondary leakage through all steam generators (SGs), and
e. 1,900 liters (500 gallons) per day primary to secondary leakage through any one SG

Pressure Boundary Leakage is defined as leakage through a non-isolable fault in an RCS component body, pipe wall, or vessel wall (except SG leakage). Leakage past seals and gaskets is not considered pressure boundary leakage.

50.73(a)(2)(ii) – Any event or condition that resulted in: **(A)** The condition of the nuclear power plant, including its principal safety barriers, being seriously degraded; or **(B)** The nuclear power plant being in an unanalyzed condition that significantly degraded plant safety.

NUREG-1022, Rev. 2 clarifies statement (A) as:
This criterion applies to material (e.g., metallurgical or chemical) problems that cause abnormal degradation of or stress upon the principal safety barriers (i.e., the fuel cladding, reactor coolant system pressure boundary, or the containment). Abnormal degradation of a barrier may be indicated by the necessity of taking corrective action to restore the barrier's capability . . .

PWR tech specs also contain reporting guidance (via LERs) associated with the plants SG tube surveillance program. Typically, this reporting requirement is triggered when an inspection reveals that greater than 1% of the tubes in a SG are found to be defective (i.e., greater than 40% thru-wall crack).

ATTACHMENT B TO APPENDIX H – SEQUENCE CODING AND SEARCH SYSTEM

The Sequence Coding and Search System (SCSS) is an NRC-sponsored database maintained by Oak Ridge National Laboratory (ORNL). It is a web-accessible database of licensee event reports (LERs) that can execute searches using a variety of criteria. It can be accessed at:

http://scss.ornl.gov/

The following search criteria were used to generate the LER portions of the non-pipe database.

LER SYSTEM EVENT SEARCH CRITERIA

Primary System(s) =SAB, SAF, SAE, SAA, SAD, SAI, SAH
Interfacing System(s) =Any
Include Trains/Channels =Yes
Include Components =Yes
Happening(s) =Any
Event Cause(s) =Any
Event Effect(s) =BH, BF, BE, BI, DE, BN, BL, BK, BP, BX, BC, BB, BA, BD
Event Timing(s) =Any
Detection Methods(s) =Any
Nuclear Plant =Any
Beginning Event Date =01/01/1990
Ending Event Date =1/1/2003
Maximum LERs =2000

This search returned 1,036 LERs. Basically, this search criteria looks for any leaking or cracking event associated with any primary coolant related system. The above search criteria rely upon the coding effort performed by the staff at ORNL as part of the SCSS program. In that effort, each LER is reviewed and characterized for possible relevance to each related system. This characterization includes both actual and possible system failures. Therefore, these search criteria returned both pipe and non-pipe failures, as well as many "non-failure" events. Each of the returned 1,036 LERs was reviewed and approximately 80% (823 LERs) were judged to be non-failures and coded as not-applicable (NA). Most of these NA events were of the type where an engineering review or some other analysis was performed by the plant, and it was found that a pipe was inadequately (compared to the design requirements) constrained such that if an earthquake were to occur, there was an increased chance that the pipe might fail. Another common "non-failure" example is of a problem unrelated to the integrity of the primary coolant system, which would have adversely affected the ability of the plant to respond to a loss of coolant accident (i.e., a failure of the primary coolant system). These potential or possible issues were judged to not be actual failures and hence were deleted from the list. A further 34 LERs were removed from the set of LERs when they were found to document problems associated with pipe defects (or pipe-weld defects).

ATTACHMENT C TO APPENDIX H - ESTIMATION OF FLAW FREQUENCY

In general, a point estimate of the frequency of pipe failure (where 'failure' includes both small and large leaks and through-wall cracks, but excludes partial-through wall cracks), λ, is given by the following expression:

$$\lambda = \frac{n_F}{NT} \tag{H.C.1}$$

Where:

n_F = the number of failure events including both small and large leaks in the service data;
T = the total time over which failure events were collected;
N = the number of components that provided the observed pipe failures.

A point estimate of the total frequency of flaws (cracks and leaks), ϕ, is given by the following expression:

$$\phi = \frac{n_C}{N \cdot T \cdot f \cdot P_{FD}} + \frac{n_F}{NT} = \frac{n_C}{N \cdot T \cdot f \cdot P_{FD}} + \lambda \tag{H.C.2}$$

Where:

n_C = the number of crack or flaw events
f = the fraction of welds inspected for cracks or flaws
P_{FD} = the probability that an inspected weld will find an existing flaw

Nearly all through-wall leaks are found from independent observations such as routine leak inspections and not from NDE inspections. However, part-through cracks are only typically found by NDE and thus the number is a function of the number of inspection locations. In Equation H.C.2 we account for the observed cracks in the data base and the fact that only a fraction (f) of the welds in the database are inspected according to ISI programs looking for cracks. The number of flaws actually discovered in in-service inspection is subject to a finite NDE reliability, which is characterized by the factor P_{FD}.

If we now take the ratio of ϕ *to* λ, we get an expression for the factor by which to multiply the pipe failure rate to obtain the flaw (non-through wall crack) rate:

$$R_{C/F} = \frac{\phi}{\lambda} = \frac{n_C}{n_F \cdot f \cdot P_{FD}} + 1 \tag{H.C.3}$$

Where:

$R_{C/F}$ = Number of non-through wall cracks per leak event:

One approach to assess the $R_{C/F}$ ratio is to evaluate those records where both cracks and leaks were found during a single inspection of a component of interest. Ideally, the best data would be found in those instances where the component was 100% inspected. Without complete inspection, some assumptions about the inspection coverage, f, are required to assess this ratio. An example of this approach and the effect of the inspection coverage and probability of detection is provided by analyzing the database for CRDM nozzle failures. For the component type 'CRDM Nozzles' in B&W PWR plants the database includes 6 LERs (= 6 database records) as identified in Table H.C.1. A detailed review of each of these LERs revealed multiple failures and degradation. Equation (C.3) together with an assumption about the inspection scope (f) makes it possible to estimate $R_{C/F}$.

Table H.C.1 B&W CRDM Nozzle Failures in 'Non-Pipe' Database

Plant	Date	LER Number	No. Components Leaking	No. Components Cracked	Population	Comment
Oconee-3	2/18/2001	2001-001	9	N/A[2]	69	Expanded inspection of an additional 9 nozzles. No recordable flaws.
Oconee-3	5/2/2003	2003-001	2	N/A	69	RVH replaced
Crystal River-3	10/1/2001	2001-004	1	N/A	69	5 flaws found in CRDM Nozzle #32. Expanded inspection of 8 nozzles found no flaws.
Three Mile Island-1	10/12/2001	2001-002	5	7	69	Inspection scope included 12 nozzles
ANO-1	3/24/2001	2001-002	1	N/A	69	Visual inspection only of remaining nozzles.
ANO-1	10/7/2002	2002-003	1	6	69	NDE of all nozzles
		Totals:	19	13		

Estimates of $R_{C/F}$ for the data set in Table H.C.1 are presented in Table H.C.2 for different assumptions about the probability of detection and fraction of welds inspected. This analysis also assumes that the inspection criteria and the cracking characteristics of the events listed in Table H.C.1 are representative of the entire population. The fraction of welds inspected is a function of the ISI program requirements. In Table H.C.2, a lower bound for f is calculated using insights from piping reliability studies. This low f estimate results in the high $R_{C/F}$ estimate presented in the table.

The current ASME Section XI requirements are to inspect 25% of the Class 1 pipe welds and 7.5% of the Class 2 pipe welds. The current inspection practice for most if not all plants calls for the same welds to be inspected each inspection interval as opposed to randomly selecting a different set of welds for each interval. When cracks or significant flaws are found, the ASME code requires that an expanded search be made; however, the frequency of flaws and failures is so rare that this requirement adds very few additional inspections to the total population of inspected welds. Using data from an operating 4-loop Westinghouse PWR unit on the number of Class 1 and Class 2 welds of 1,605 and 1,800, the following estimate of the parameter f is obtained for Westinghouse PWR plants:

[2] N/A in Table H.C.1 means that a full-scope NDE was not pursued.

$$f = \frac{1{,}605(0.25) + 1{,}800(0.075)}{(1{,}605 + 1{,}800)} = 0.157 \tag{H.C.4}$$

We assume this estimate of f to be representative of the non-pipe components. With additional assumptions about the reliability of the NDE we get the results as indicated in Table H.C.2.

Table H.C.2 Estimates of $R_{C/F}$ for B&W CRDM Nozzles

Parameter	Data Source	PWSCC		
		High Est.	***Median Est.***	***Low Est.***
Number of cracks	Table H.C.1	NA	13	NA
Number of leaks	Table H.C.1	NA	19	NA
Fraction of components inspected, f	Equation (H.C.4) – lower bound	1.0	0.5	0.157
$R_{C/F}$ with $P_{FD} = 0.5$	--	9.72	3.74	2.37
$R_{C/F}$ with $P_{FD} = 0.75$	--	6.81	2.82	1.91
$R_{C/F}$ with $P_{FD} = 0.9$	--	5.84	2.52	1.76

Hence, the relative number of flaws and leaks observed does not predict the relative frequency of flaws and leaks at a given weld. The estimate for the ratio of cracks to leaks obtained in Table H.C.2 reflects the degree to which components are exposed to PWSCC and inspection coverage.

APPENDIX I

REACTOR VESSEL LOCA PROBABILITY BASE CASE ANALYSES (BWR VESSELS AND PWR TOP HEAD NOZZLES)

APPENDIX I

REACTOR VESSEL LOCA PROBABILITY BASE CASE ANALYSES (BWR VESSELS AND PWR TOP HEAD NOZZLES)

I.1 Introduction

The LOCA expert panel elicitation team charter includes estimating the contribution to LOCA frequency from reactor vessels and other non-piping components. Extensive analyses were performed by members of the elicitation panel to develop LOCA frequencies for five piping "base cases" that were formulated by the panel in early meetings (documented as Appendices D, E, F and G to this NUREG). The piping base cases include failures on the piping side of vessel nozzles, including safe-ends. However, they do not include small diameter, partial penetration welded nozzles such as control rod drive mechanism (CRDM) penetrations and other small nozzles, such as instrument nozzles, that aren't connected to piping systems. In addition, the piping base cases do not include consideration of a leak from or rupture of other regions of the reactor vessel, such as the irradiated reactor vessel beltline or the low alloy steel portions of large vessel nozzles. LOCA frequency estimates for these cases are presented in this appendix, based on prior PFM analyses performed for PWR top head nozzles [I.1, I.2], the BWR Reactor Vessel Beltline Region [I.3, I.4], and BWR reactor vessel feedwater nozzles [I.5]. These estimates are used to construct a complete set of LOCA frequency tables for BWR and PWR reactor vessels, for all LOCA categories defined in the elicitation, a comparison of them to the aforementioned piping base cases is also presented.

I.2 PWR Reactor Vessel Top Head Nozzles

Extensive probabilistic fracture mechanics (PFM) analyses have been conducted over the past several years to estimate the probability of leakage and rupture associated with the PWR CRDM penetration PWSCC problem [I.1, I.2]. The analysis model incorporates the following major elements:

- computation of applied stress intensity factors for circumferential cracks in various nozzle geometries as a function of crack length,
- determination of critical circumferential flaw sizes for nozzle failure,
- an empirical (Weibull) analysis of the probability of nozzle cracking or leakage as a function of operating time and temperature of the RPV head,
- statistical analysis of PWSCC crack growth rates in the PWR primary water environment as a function of applied stress intensity factor and service temperature, and
- modeling of the effects of inspections, including inspection type, frequency and effectiveness.

The model has been benchmarked with respect to field experience, considering the occurrence of cracking and leakage and of circumferential cracks of various sizes. Figures I.1 and I.2 illustrate the benchmarking. Figure I.1 presents a Weibull analysis of inspection results at thirty plants, of which 14 detected leakage or cracking (data points in the figure). The remaining plants that were inspected and found clean were treated as "suspended tests" according to standard Weibull analysis theory [I.2]. The data are plotted in terms of effective degradation years (EDYs) which are equivalent operating years at 315 C (600°F), using an activation energy (Arrhenius) model [I.1] to adjust for different head operating temperatures. For plants in which multiple cracked nozzles were detected in the inspections, the data were extrapolated back to the expected time of first cracking or leakage, using an assumed Weibull slope of 3. The straight line through the data represents a medium rank Weibull regression (also with a slope of 3) upon which the probability of leakage predictions in the model are based. Figure I.2 illustrates the benchmarking process used for the crack growth analysis algorithm in the model with respect to CRDM nozzles that exhibited circumferential cracks of various sizes. (Eleven (11) nozzles out of a total of 881 inspected nondestructively through the spring of 2003 exhibited circumferential cracking. No additional circumferential cracking has been detected in more recent inspections.) The figure shows that, when using original analysis parameters, the crack growth model under-predicted the probability of circumferential cracking somewhat, but after adjustment of selected analytical parameters, the PFM model was "benchmarked" so as to very accurately predict the field results, especially for the most important, larger crack sizes.

The benchmarked model was then used to evaluate the probabilities of nozzle failure and leakage in actual plants. A sample of the results is presented in Figures I.3 and I.4. Figure I.3 illustrates the probability of nozzle failure (ejection of a nozzle) for a head operating temperature of 304 C (580° F), the approximate average of U.S. PWRs. No inspections were assumed to be performed during the first 25 years of plant operation, resulting in the probability of nozzle failure constantly increasing with time during that period. The analysis then assumed that inspections begin after 25 years, at intervals and detection levels representative of current requirements [I.6]. It is seen from the figure that the current inspection regimen reduces the nozzle failure probability significantly.

Ejection of a 4 inch CRDM nozzle [~70 mm (2.75 inch) ID] due to a circumferential crack would yield a one-sided LOCA corresponding approximately to Category 2 LOCA [> 5,700 lpm (1,500 gpm) but < 19,000 lpm (5,000 gpm)]. If periodic inspections are continued, with any nozzles in which leakage or cracking are detected repaired or the heads replaced (as is common practice), the nozzle ejection probability will be even lower in the future. Table I.1 below provides a summary of the average failure probabilities from Figure I.3, between 0 and 25 years, and from 25 to 40 years. The probability of failure for 40 to 60 years was not calculated, but was assumed to be the same as 25 to 40 years, on the basis that the current inspection regimen will be maintained, or the heads replaced. A Category 3 break was assumed to require multiple nozzle failures, the probability of which was computed via a binomial distribution for the typical number of nozzles in a head. As seen in Table I.1, the probabilities of simultaneous multiple nozzle failures is quite low.

Figure I.4 illustrates similar PFM results (based on the above Weibull model) for the probability of small amounts of leakage from a top head CRDM nozzle. The same inspection regimen was assumed as in the nozzle ejection analysis (no inspections from 0 to 25 years, inspections in accordance with current requirements thereafter). A small leak from a CRDM nozzle was assigned as a Category 0 break [less than 3.8 lpm (1 gpm)] in Table I.1, and the intermediate, Category 1 break size was obtained by logarithmic interpolation between Categories 0 and 2.

Table I.1 Summary of PWR CRDM Nozzle PFM Results

Break Category	Leak Rate >(gpm)	Average LOCA Probabilities During Operating Years:		
		0-25	25-40	40-60
0	1	2.00E-02	5.00E-03	5.00E-03
1	100	1.27E-03	2.75E-04	2.75E-04
2	1,500	2.50E-04	5.00E-05	5.00E-05
3	5,000	4.00E-08	2.00E-09	2.00E-09
4	25,000	-	-	-
5	100,000	-	-	-
6	500,000	-	-	-

I.3 BWR Reactor Vessels

Analyses have been previously submitted and approved [I.3, I.4] that establish reduced inspection requirements for BWR reactor vessels relative to ASME Section XI requirements. Specifically, BWRVIP-05 [I.3] justifies that only axially-oriented welds in the vessel beltline region need be examined on a ten year interval, versus the Section XI requirement to inspect all axial and circumferential welds on this interval. This relief was based on PFM calculations demonstrating, for the BWR fleet, that circumferential weld inspections contribute negligibly to reduction in the already small failure probability of a BWR vessel. The methodology used for the PFM analysis is a computer program (VIPER [I.7]) developed by Structural Integrity Associates for EPRI and the BWRVIP. To address this LOCA frequency contributor, the VIPER software was run for a typical BWR vessel, extending the analysis period from 40 to 60 years. A modification to the software (VIPER-NOZ) was also used to estimate leakage and failure probabilities for BWR Reactor Vessel feedwater nozzles. Feedwater nozzles were selected because they are subject to thermal fatigue cycling, which caused serious nozzle cracking in the 1970s [I.5]. Both analyses take credit for routine in-service inspection programs that are conducted on these components on ten-year inspection intervals. The feedwater nozzle analysis also takes credit for nozzle modifications and thermal sleeve improvements that were installed in all U.S. BWRs to reduce the severity of the thermal fatigue cycling.

I.3.1 BWR Vessel Beltline Region

In the VIPER software, cracks are assumed to exist in BWR vessel welds due to two causes – original manufacturing defects and service-induced cracks which initiate in the stainless steel cladding. These cracks are assumed to grow as a function of operating time due to fatigue crack growth and stress corrosion cracking of the low alloy steel vessel material. Simultaneously, the vessel beltline region is assumed to embrittle due to irradiation. Monte Carlo simulations of these processes are employed in VIPER, which include fracture mechanics crack growth calculations due to fatigue and stress corrosion cracking, and a comparison of predicted crack sizes to the critical crack size due to normal operation as well as possible transient conditions. The governing transient condition was determined to be a low temperature over-pressurization (LTOP) event, since BWRs are not subject to pressurized thermal shock (PTS).

The effects of in-service inspections are imposed at appropriate inspection intervals, assuming a probability of detection (POD) curve for the inspections. Flaws that are detected during in-service inspections are assumed to be repaired, and thus eliminated from the population, such that they can no longer grow to a leak or vessel failure.

The axial vessel beltline welds are divided into a series of segments, and each segment is analyzed separately to account for axial gradients of irradiation fluence in the welds, which peaks at the core centerline, and decays at elevations above and below that location. The failure frequencies from each segment are weighted by their respective weld volume, and summed to determine failure frequency for the entire vessel.

Modes of failure considered are:

1. Vessel fracture during normal operation ($K_I > K_{Ic}$)
2. Vessel fracture during an assumed LTOP event. The LTOP event considered is pressurization to 7.93 MPa (1,150 psi) at 31 C (88°F), which is assumed to occur at a frequency of 1E-3.
3. Predicted crack growth to 80% of wall thickness before failure modes 1 or 2 occurs (Leak Before Break)

The results for a typical BWR are given in the following table:

Table I.2 Summary of BWR RPV Beltline PFM Results

Break Category	Leak Rate >(gpm)	Average LOCA Probabilities During Operating Years:		
		0-25	25-40	40-60
1	100	1.00E-08	2.98E-08	4.57E-08
2	1,500	2.32E-09	4.31E-09	2.84E-08
3	5,000	1.21E-09	1.83E-09	2.30E-08
4	25,000	5.04E-10	5.79E-10	1.73E-08
5	100,000	2.38E-10	2.15E-10	1.36E-08
6	500,000	9.86E-11	6.79E-11	1.02E-08

These provide an estimate of the probability (per vessel year) of breaks of various sizes due to vessel beltline failures. To complete this table, it was assumed that a leak (LBB mode failure) corresponds to a crack of length = 1525 mm (60 inches) that breaks through and begins leaking as a through-wall crack of this length (since the wall thickness is approximately 150 mm (6 inches), and cracks in VIPER are assumed to have a ten to one aspect ratio). Dave Harris ran this case using the PRAISE code leakage rate prediction capability (See Appendix F for description), and computed a leak rate of 733 lpm (193 gpm) for an axial crack of this size in a BWR vessel. Thus, predicted LBB mode failures from the RPV beltline were treated as Category 1 breaks. Predicted vessel fractures, either during normal operation or due to LTOP events were treated as complete RPV ruptures, which were assumed to result in very large, Category 6 breaks. Intermediate break sizes were then determined by log-log interpolation between these two extremes. The sharp increase in large break probability between years 40 and 60 is attributable to the combined effect of two aging mechanisms – crack growth and RPV embrittlement.

I.3.2 BWR Feedwater Nozzles

RPV nozzles constitute another potential non-piping LOCA concern. The example used to address this concern here is BWR feedwater nozzles, which have in the past been subject to thermal fatigue cracking [I.5]. The thermal fatigue problem was caused by mixing of hot reactor water and relatively cold feedwater (see Figure I.5) during reactor startups, shutdowns and other periods of low power operation, when feedwater heating is generally unavailable. Cracking of various depths, up to 38 mm (1.5 inches), was detected in a number of BWRs in the 1970s (see Figure I.6). At that time, the standard feedwater nozzle design incorporated a loose-fitting thermal sleeve/sparger configuration, as shown in Figure I.5. Since then, all U.S. BWRs have installed some type of fix, employing either welded-in spargers or multiple-sleeve designs with shrink fits and piston rings to protect the nozzle from the effects of the cold feedwater. No subsequent cracking has been discovered since the improved thermal sleeves were installed.

In order to perform a base case analysis of this problem, a modification to the software (VIPER-NOZ) was developed to estimate leakage and failure probabilities for BWR Reactor Vessel feedwater nozzles. The substantive changes to the VIPER software in VIPER-NOZ were the addition of thermal fatigue crack initiation and growth algorithms specific to the feedwater nozzle thermal cycling phenomenon, and zeroing out the effects of irradiation embrittlement, since feedwater nozzles are far enough from the reactor core region that neutron fluence effects are small. The VIPER-NOZ software was run for conditions representative of the original nozzle/sparger designs, to confirm that cracking probabilities consistent with early field experience (Figure I.3) are predicted. The boundary conditions were then modified to represent improved nozzle/sparger designs, which reduce the effects of thermal fatigue on the nozzle. The analyses were conducted for a 60 year operating lifetime, and included the effects of periodic in-service inspections, which are performed for these nozzles on ten-year intervals. The results are given in the following table:

Table I.3 Summary of BWR Feedwater Nozzle PFM Results

Break Category	Leak Rate >(gpm)	Average LOCA Probabilities During Operating Years:		
		0-25	25-40	40-60
0	1	<1.00E-06	1.47E-06	1.25E-06
1	100	<1.00E-06	<1.00E-06	<1.00E-06
2	1,500	<1.00E-06	<1.00E-06	<1.00E-06
3	5,000	<1.00E-06	<1.00E-06	<1.00E-06
4	25,000	<1.00E-06	<1.00E-06	<1.00E-06

The predicted leakage cases were treated as Category 0 breaks in this case, and since the nozzle is attached to a 12 inch diameter pipe, the maximum credible break size was assumed to correspond to single ended rupture of a 12 inch pipe, which corresponds to a Category 4 break. A total of 1 million simulations were run, and except for LBB type failures at 40 and 60 years, no other failures were predicted. Thus a failure frequency of less than 1E-6 is given for most entries in the above table.

I.4 Combined LOCA Frequencies due to Reactor Vessel Failures

Tables I.4 and I.5 summarize and combine the above RPV LOCA frequency results for BWRs and PWRs, respectively. For BWRs, three RPV LOCA contributors are addressed: RPV vessel beltline region, large nozzles (6 through 28 inch diameter), and small penetrations (partial penetration welded nozzles, 4 inch in diameter or less, such as CRDs). The individual LOCA probability contributions for each of these are provided in the top three sections of Table I.4, and they are summed in the bottom section of the table. These address all LOCA categories as well as the three time periods under consideration (0-25 yrs, 25-40 yrs, and 40-60 yrs). Note, in Table 4.4 in Section 4 of the main body of this report, only the results for the BWR beltline region and the large feedwater nozzles are presented. The results for the BWR CRDs and other small penetrations listed in Table I.4 of this appendix were not included in Table 4.4 since the estimates for the BWR CRDs and other penetrations are not based on analysis, but instead, were based on engineering judgment, i.e., BWR CRDs and other penetrations LOCA frequencies were simply assumed to be a factor of 10 less than the PWR CRDM LOCA frequencies, which were based on analysis.

The first contributor is the BWR shell region, the failure probabilities for which are dominated by the irradiated reactor vessel beltline region. The upper section of Table I.4 summarizes the results of this analysis from Table I.2, in terms of the probability of leaks of various sizes due to degradation or failure of the RPV beltline region. (Note that these were modified slightly relative to those in Table I.2 to eliminate the negative time factor for Category 5 and 6 LOCAs in the 25 – 40 year period.)

The second section of Table I.4 addresses large nozzle contributions to LOCA probability, which in BWRs are assumed to correspond to the 12 inch diameter Feedwater Nozzles that experienced thermal fatigue cracking in the 1980s [I.5]. LOCA probabilities due to this contributor are given for break Categories 1 through 4, taken from Table I.3 as this nozzle size could not lead to larger break sizes. For the Category 2 through 4 LOCAs for 0-25 year time frame, an assumption was made as to the size factor that each successive LOCA size greater than Category 1 was 5 times less likely to occur than the previous size LOCA. Then, for the 25-40 and 40-60 year time frames, the same time factor as determined for the Category 0 LOCAs in Table I.3 was assumed for the larger size LOCAs. Breaks of the other, larger diameter nozzles, such as recirculation outlet nozzles, are considered to be adequately encompassed by the vessel beltline case.

Finally, the third section of Table I.4 lists LOCA probabilities due to failures of CRDs and other small penetrations in the BWR vessel. These were estimated from the detailed analysis of PWR CRDM penetrations described above (Table I.1) but they assume that the BWR penetrations have about an order of magnitude lower LOCA probability than similar penetrations in a PWR. The order of magnitude reduction is deemed appropriate, because problems in small vessel penetrations in BWRs have occurred at a much lower frequency than the recent PWSCC experience in PWRs, upon which Table I.1 is based. The problems in BWR penetrations have also been attributed to a fairly well-understood phenomenon (IGSCC) and in most cases the nozzles of concern have been mitigated by design and materials changes.

Table I.5 provides a similar summary for PWR RPVs. In this case, LOCA probabilities are reported for only two categories of LOCA contributors, the shell region (RPV beltline) and small penetrations. Again, as was the case for BWRs, Table 4.5 in Section 4 only includes the results for PWR CRDMs. It does not include the results for the PWR beltline region as reported in Table I.5 of this appendix. As was the case for BWR CRDs and other penetrations, the PWR beltline results in Table I.5 are not based on analysis. Again, they are based on engineering judgment, i.e., the PWR beltline LOCA frequencies in Table I.5 of this appendix were simply assumed to be a factor of ten greater than the BWR beltline LOCA frequencies from Table I.4. It was judged that the large nozzles in a PWR RPV do not pose a significant LOCA risk because they are not subject to significant thermal cycles such as the BWR Feedwater nozzles, and except for the safe-ends (which are covered in the piping elicitation), they have not experienced any degradation mechanisms to date. The contributions for the two PWR RPV LOCA contributors are summed in the bottom section of Table I.5.

For the PWR beltline region, results from a prior analysis of a PWR vessel using a third version of the VIPER software (VIPER-PWR) were reviewed. Based on this review, it was estimated that the PWR RPV beltline region presents about an order of magnitude increase in large rupture probability relative to that of a BWR, because PWR beltlines are more highly irradiation embrittled, and because they are potentially subject to pressurized thermal shock (PTS) transients. Thus, the BWR RPV beltline region LOCA frequency entries in Table I.4 were multiplied by a factor of ten and entered in the upper section of Table I.5.

PWR CRDM penetrations results were entered directly from the above PFM analysis results in Table I.1.

I.5 Summary and Comparison to Piping Base Cases

Figures I.7 and I.8 present plots of these RPV base cases, compared to the piping base cases from Appendices D, E, F, and G. For purposes of this comparison, a single set of piping base case LOCA frequencies were derived that are a composite of the results from the four appendices. Plots are presented for the 0-25 year (Figure I.7) and the 25-40 year (Figure I.8) periods. Since the RPV LOCA frequencies for the 40-60 year period are not significantly different than the 25-40 year results, a separate plot for that case is not included. It is seen from these figures that the RPV base cases are at the low end of the piping LOCA probabilities for the large break Categories 5 and 6, but are at the high end for small, Category 1 and 2 breaks, due largely to the small penetration (CRDM) contributions discussed above. Note also that the small LOCA probability estimates are substantially lower in the outlying years (25-40 and 40-60) because of inspection programs implemented as a result of these issues. In general, small break LOCA frequency contributors (Categories 1 and 2) from PWR RPVs are seen to be greater than those for BWRs, due to the

PWSCC concern in CRDM and other small penetrations. Large break LOCA contributors (Categories 5 and 6) are also estimated to be greater for PWR RPVs due to higher irradiation embrittlement and the potential for Pressurized Thermal Shock (PTS) transients.

Table I.4 LOCA Frequencies for BWR Reactor Pressure Vessel Base Case

BWR RPV Beltline

Break Cat.	Break Size gpm	NPS	0-25 yrs	25-40 yrs	TimeFactor	40-60 yrs	TimeFactor
1	100	0.5	1.00E-08	2.98E-08	2.98	4.57E-08	4.57
2	1,500	1.5	2.32E-09	6.19E-09	2.67	2.84E-08	12.24
3	5,000	3.5	1.21E-09	3.12E-09	2.58	2.30E-08	19.01
4	25,000	7	5.04E-10	1.25E-09	2.47	1.73E-08	34.33
5	100,000	16	2.38E-10	5.65E-10	2.37	1.36E-08	57.14
6	500,000	30	9.86E-11	2.32E-10	2.35	1.02E-08	103.45

(Average LOCA Probabilities During Operating Years)

BWR FW Nozzles

Break Cat.	Break Size gpm	NPS	0-25 yrs	25-40 yrs	TimeFactor	40-60 yrs	TimeFactor
1	100	0.5	1.00E-06	1.47E-06	1.47	1.25E-06	1.25
2	1,500	1.5	2.00E-07	2.94E-07	1.47	2.50E-07	1.25
3	5,000	3.5	4.00E-08	5.88E-08	1.47	5.00E-08	1.25
4	25,000	7	8.00E-09	1.18E-08	1.47	1.00E-08	1.25
5	100,000	16					
6	500,000	30					

(Average LOCA Probabilities during Operating Years)

BWR CRDs & Other Small Penetrations

Break Cat.	Break Size gpm	NPS	0-25 yrs	25-40 yrs	Factor	40-60 yrs	Factor
0			2.00E-03	5.00E-04		5.00E-04	
1	100	0.5	1.27E-04	2.75E-05	0.22	2.75E-05	0.22
2	1,500	1.5	2.50E-05	5.00E-06	0.20	5.00E-06	0.20
3	5,000	3.5	4.00E-09	2.00E-10	0.05	2.00E-10	0.05
4	25,000	7					
5	100,000	16					
6	500,000	30					

(Average LOCA Probabilities during Operating Years)

BWR Vessel - Totals

Break Cat.	Break Size gpm	NPS	0-25 yrs	25-40 yrs	TimeFactor	40-60 yrs	TimeFactor
1	100	0.5	1.28E-04	2.90E-05	0.23	2.88E-05	0.23
2	1,500	1.5	2.52E-05	5.30E-06	0.21	5.28E-06	0.21
3	5,000	3.5	4.52E-08	6.21E-08	1.37	7.32E-08	1.62
4	25,000	7	8.50E-09	1.30E-08	1.53	2.73E-08	3.21
5	100,000	16	2.38E-10	5.65E-10	2.37	1.36E-08	57.14
6	500,000	30	9.86E-11	2.32E-10	2.35	1.02E-08	103.45

(Average LOCA Probabilities during Operating Years)

Table I.5 LOCA Frequencies for PWR Reactor Pressure Vessel Base Case

PWR RPV Beltline

Break Cat.	Break Size gpm	NPS	Pete Riccardella Estimate 0-25 yrs	25-40 yrs	Factor	40-60 yrs	Factor
1	100	0.5	1.00E-07	2.98E-07	2.98	4.57E-07	4.57
2	1,500	1.5	2.32E-08	6.19E-08	2.67	2.84E-07	12.24
3	5,000	3	1.21E-08	3.12E-08	2.58	2.30E-07	19.01
4	25,000	7	5.04E-09	1.25E-08	2.47	1.73E-07	34.33
5	100,000	14	2.38E-09	5.65E-09	2.37	1.36E-07	57.14
6	500,000	30	9.86E-10	2.32E-09	2.35	1.02E-07	103.45

PWR CRDMs

Break Cat.	Break Size gpm	NPS	Pete Riccardella Estimate 0-25 yrs	25-40 yrs		40-60 yrs	
0			2.00E-02	5.00E-03	Factor	5.00E-03	Factor
1	100	0.5	1.27E-03	2.75E-04	0.22	2.75E-04	0.22
2	1,500	1.5	2.50E-04	5.00E-05	0.20	5.00E-05	0.20
3	5,000	3.5	4.00E-08	2.00E-09	0.05	2.00E-09	0.05
4	25,000	7					
5	100,000	16					
6	500,000	30					

PWR Vessel - Totals

Break Cat.	Break Size gpm	NPS	Pete Riccardella Estimate 0-25 yrs	25-40 yrs	Factor	40-60 yrs	Factor
1	100	0.5	1.27E-03	2.75E-04	0.22	2.76E-04	0.22
2	1,500	1.5	2.50E-04	5.01E-05	0.20	5.03E-05	0.20
3	5,000	3.5	5.21E-08	3.32E-08	0.64	2.32E-07	4.45
4	25,000	7	5.04E-09	1.25E-08	2.47	1.73E-07	34.33
5	100,000	16	2.38E-09	5.65E-09	2.37	1.36E-07	57.14
6	500,000	30	9.86E-10	2.32E-09	2.35	1.02E-07	103.45

I.6 References

I.1 Peter Riccardella, Nathaniel Cofie, Angah Miessi, Stan Tang, Bryan Templeton, "Probabilistic Fracture Mechanics Analysis to Support Inspection Intervals for RPV Top Head Nozzles" U.S. Nuclear Regulatory Commission / Argonne National Laboratory Conference on Vessel Head Penetration Inspection, Cracking, and Repairs, September 29 – October 2, 2003, Gaithersburg, Maryland.

I.2 Materials Reliability Program, MRP-105, "Probabilistic Fracture Mechanics Analysis of PWR Reactor Pressure Vessel Top Head Nozzle Cracking," EPRI Report 1007834 (EPRI Licensed Material), May, 2004.

I.3 EPRI Report, "BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)," TR-105697, September 1995.

I.4 NRC Report, "Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report (TAC No. M93925)," Division of Engineering, Office of Nuclear Reactor Regulation, May 1998.

I.5 NUREG-0619, "BWR Feedwater Nozzle and CRD Return Line Nozzle Cracking, Resolution of Generic Tech Activity A-10," November 1980.

I.6 U.S. NRC Order EA-03-009, "Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors", issued on February 11, 2003.

I.7 VIPER Version 1.2, Structural Integrity Associates, Report # SIR-95-098 Rev. 1, Feb. 1999.

All inspection data adjusted to 600 °F (Q = 50 kcal/mole)

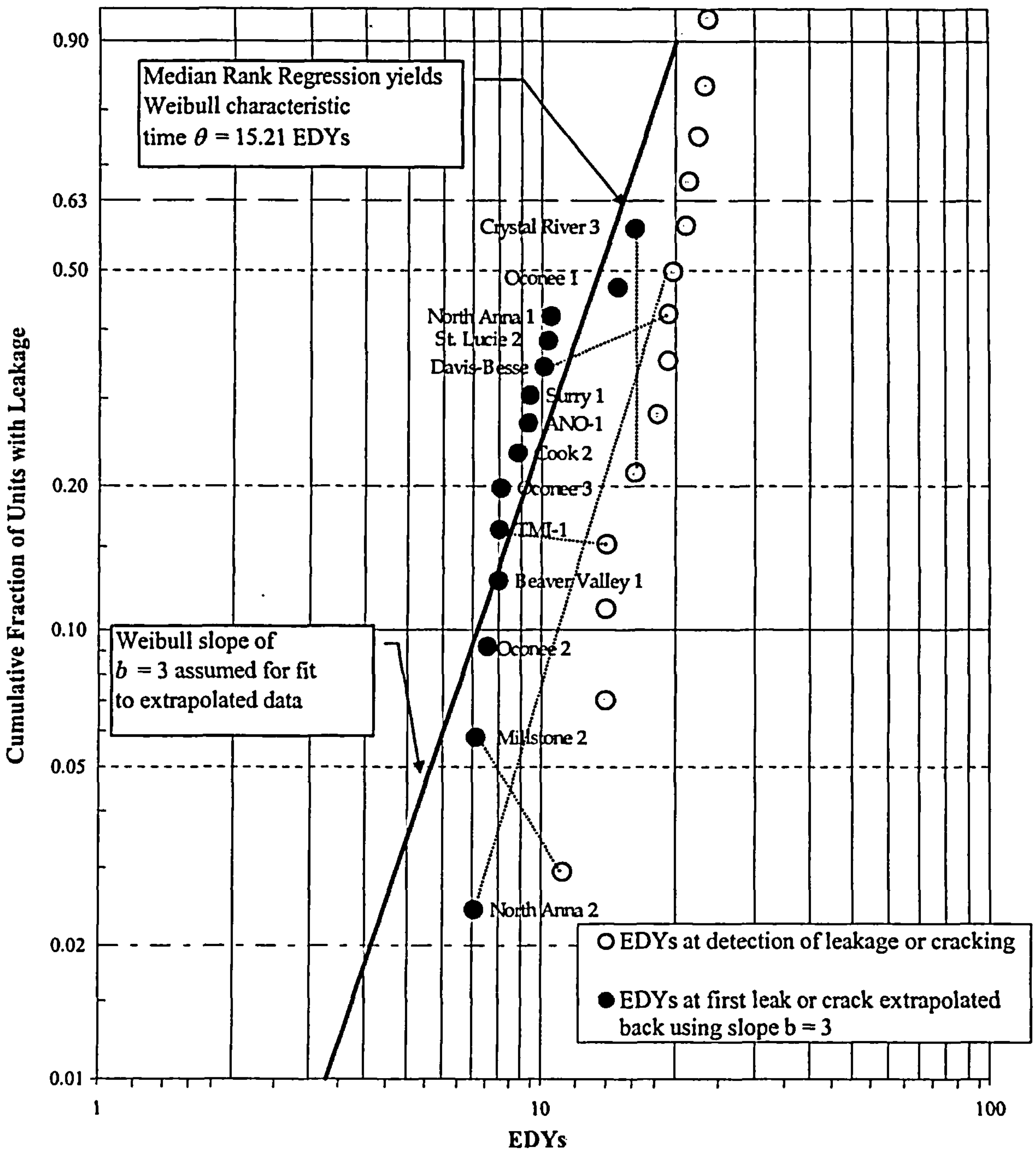


Figure I.1 Weibull Plot of Plant Inspection Data Showing Extrapolation Back to Time of First Leakage or Cracking. Plants that Performed NDE and were Found Clean are Treated as Suspensions.

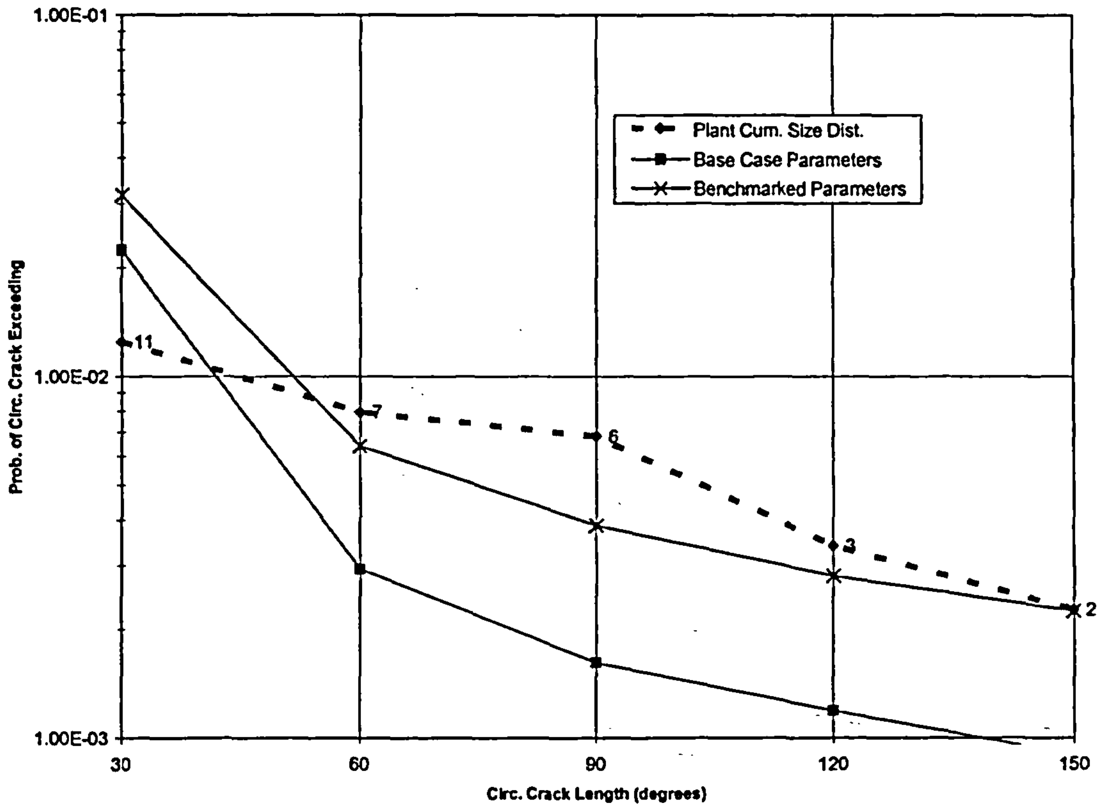


Figure I.2 Benchmarking of PFM Crack Growth Analyses with Respect to Field-Observed Circumferential Cracking of Various Lengths

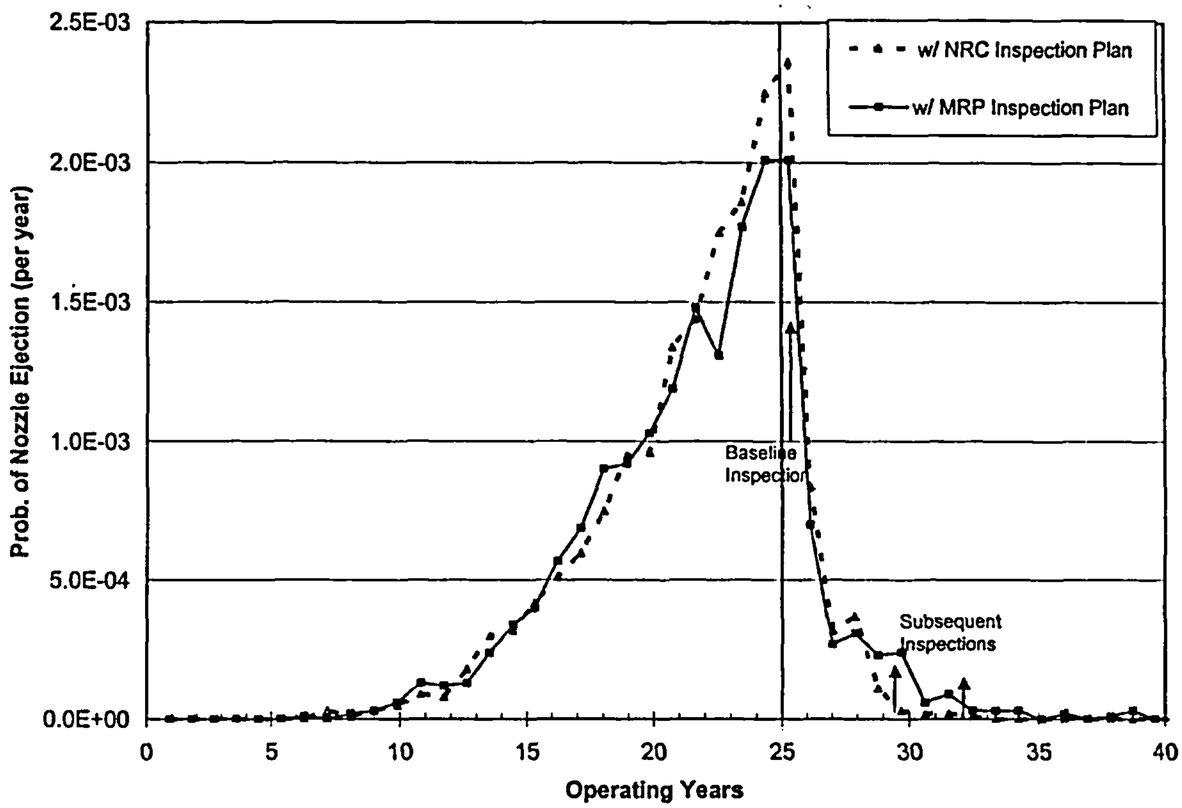


Figure I.3 RPV Top Head PFM Analysis Results for Plant with 304 C (580°F) Head Temperature – Probability of CRDM Nozzle Failure (i.e. Ejection of Nozzle from Vessel Head)

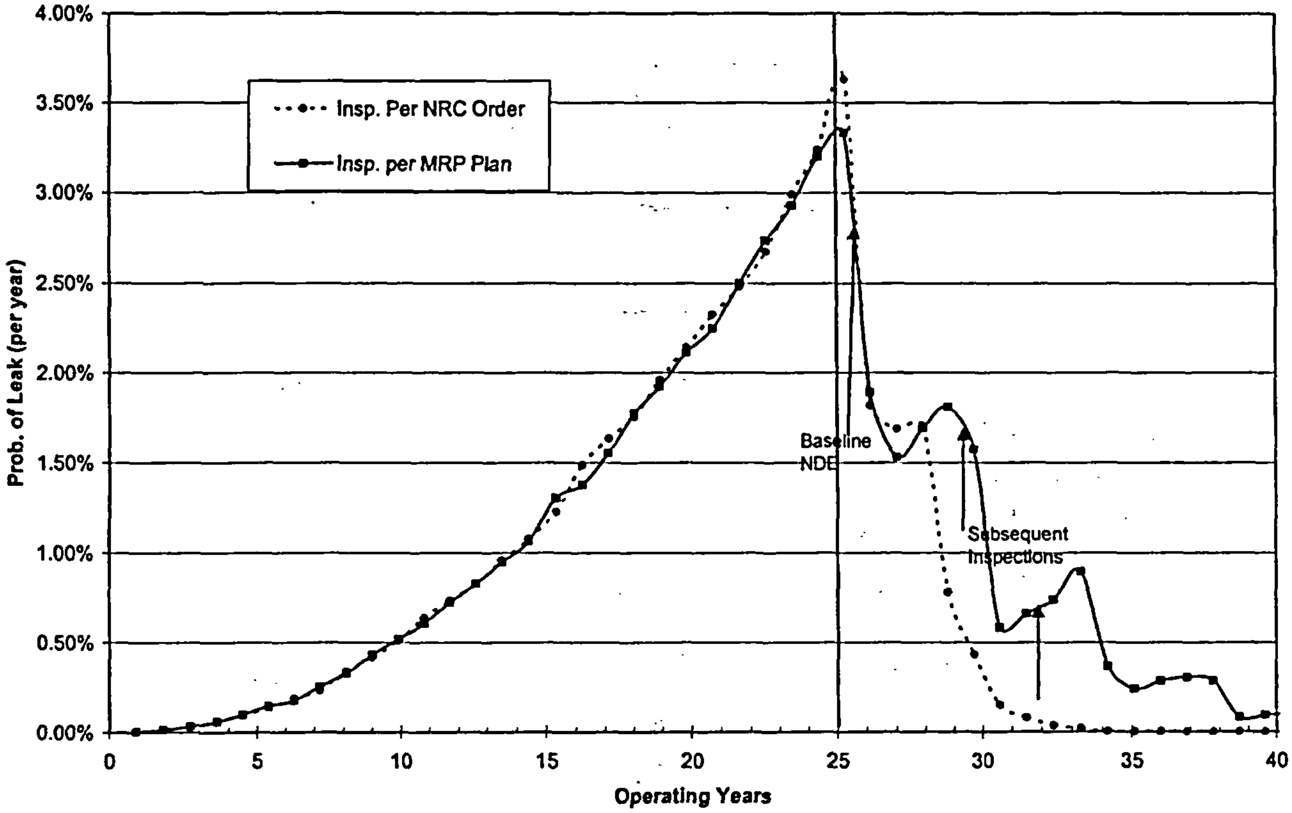


Figure I.4 RPV Top Head PFM Analysis Results for Plant with 304 C (580°F) Head Temperature – Probability of Leakage from CRDM Nozzle

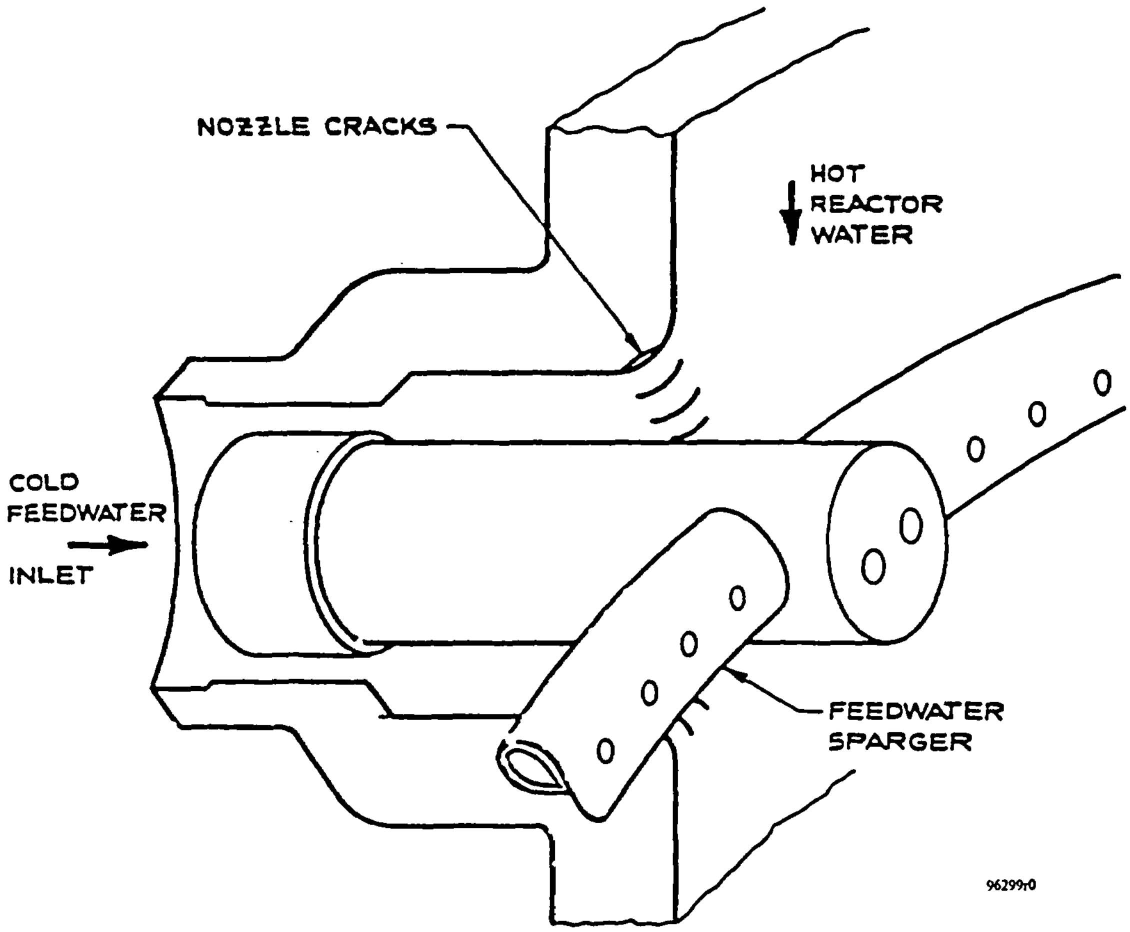


Figure I.5 Schematic of Thermal Fatigue Cracking in BWR Feedwater Nozzles

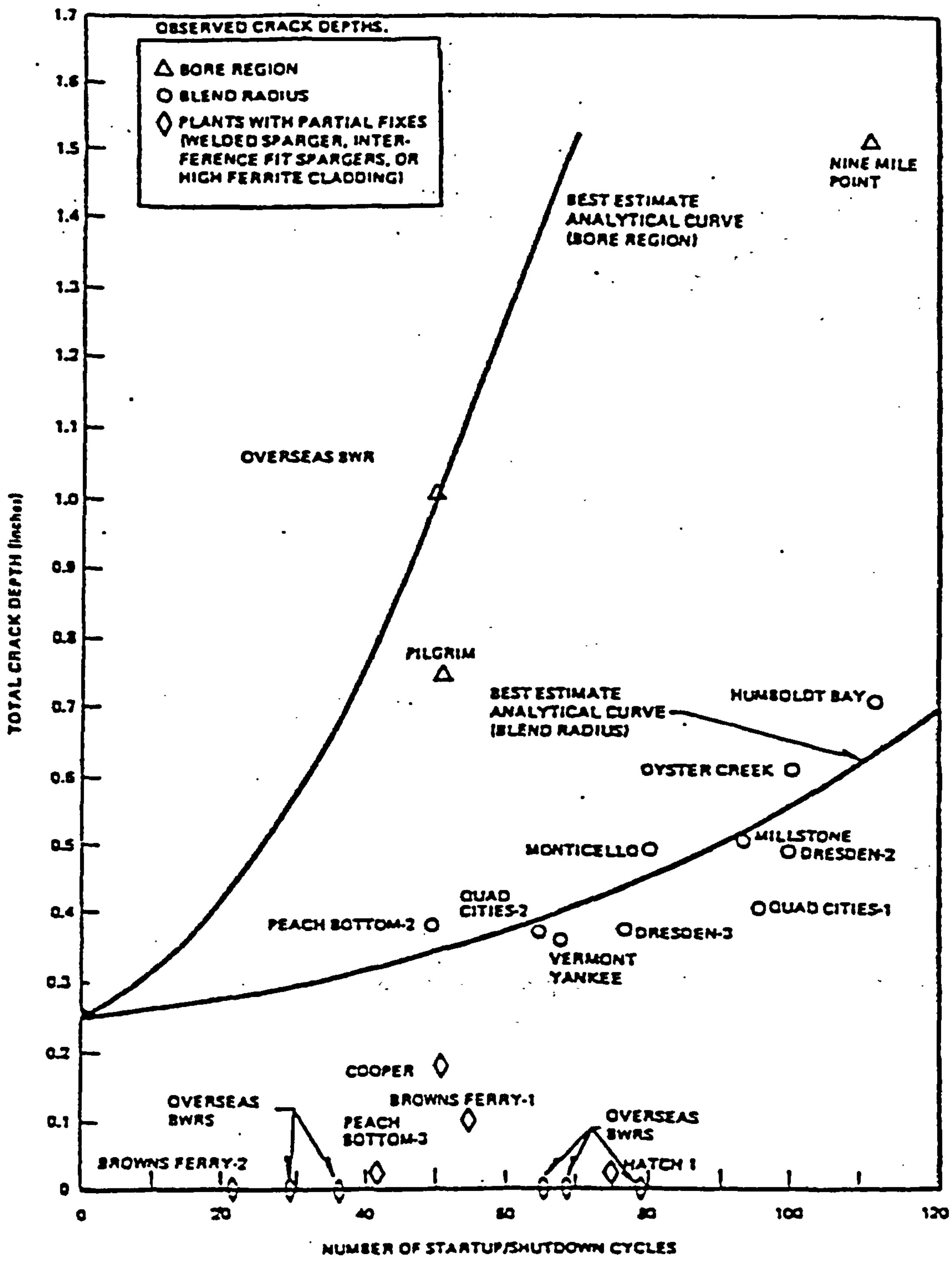


Figure I.6 Historical BWR Feedwater Nozzle Cracking Experience (circa 1980)

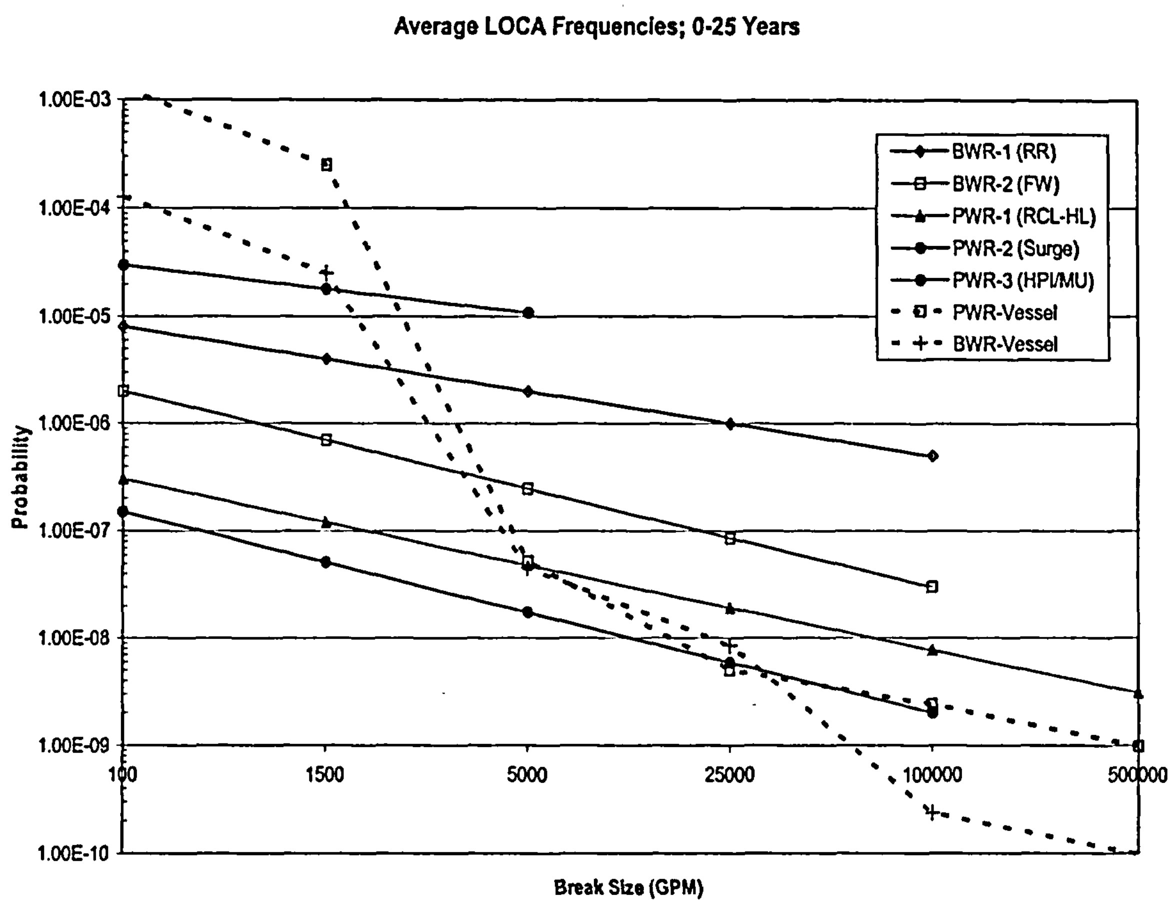


Figure I.7 Comparison of RPV and Piping Base Case LOCA Frequencies Versus Break Size (0 to 25 Years)

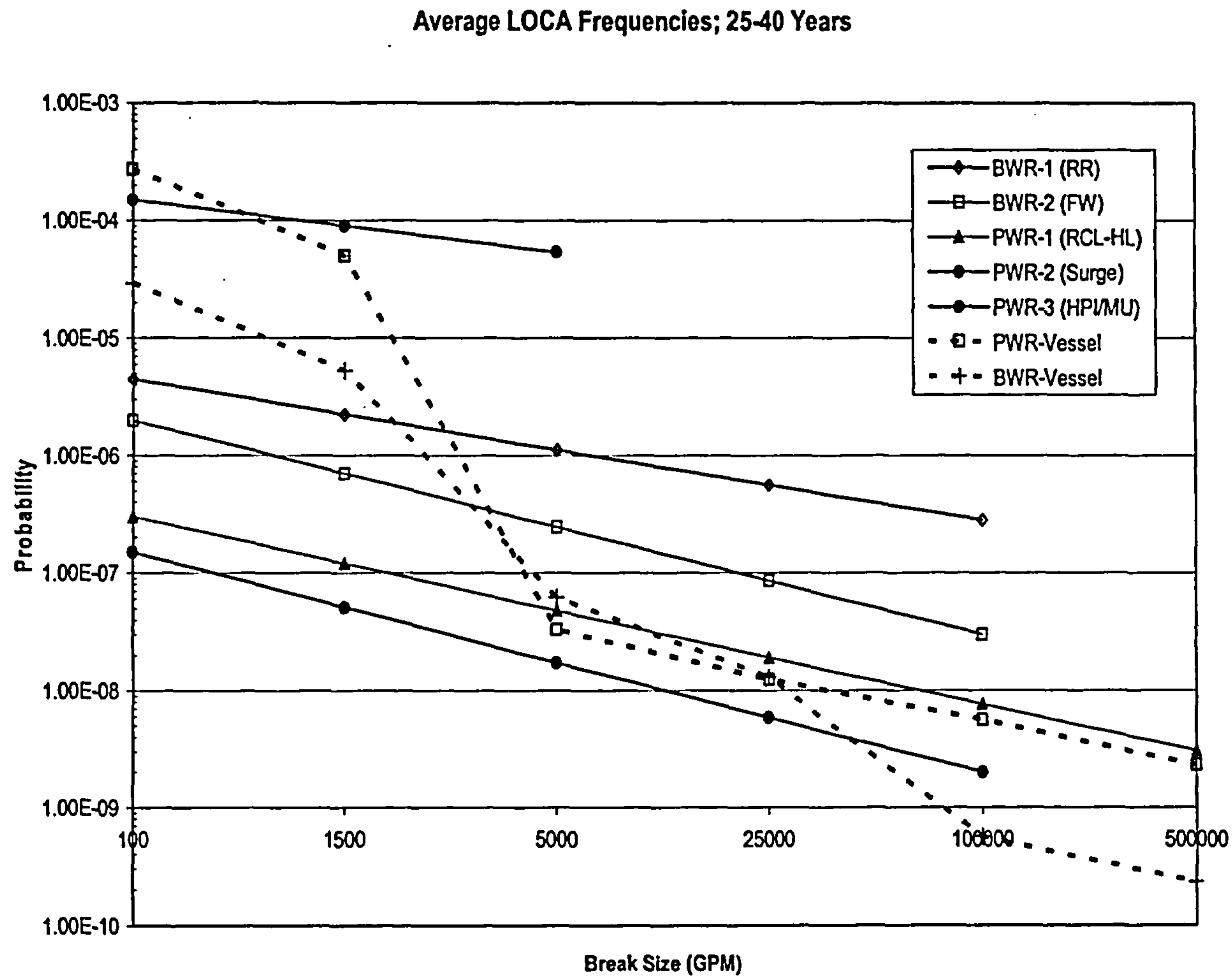


Figure I.8 Comparison of RPV and Piping Base Case LOCA Frequencies Versus Break Size (25 to 40 Years)

APPENDIX J

ELICITATION QUESTIONS

APPENDIX J

ELICITATION QUESTIONS

J.1 Instructions

There are four basic quantities that are the ultimate focus of this exercise: the LOCA frequencies of piping components, the LOCA frequencies of non-piping components, the LOCA probabilities of piping components after emergency faulted loading, and the LOCA probabilities of non-piping components after emergency faulted loading. The elicitation will be structured so that each of these questions can be answered using one of two question sets. The question sets are structured to decompose the underlying issues using different approaches so increase your flexibility.

The bottom-up approaches (3A, 4A, 5A, 6A) could entail significantly more work if every piping and non-piping system is evaluated. It is recommended that people choosing this approach focus on significant contributing issues in only significant piping and non-piping systems to reduce the workload. Similarly the people choosing the top-down approaches (3B, 4B, 5B, 6B) may want to ensure that their significant issues are manifested correctly within relevant systems. These strategies allow you to combine features of each methodology.

Only a few additional examples of these questions are provided in this document. Many examples will be similar to those included in the *Elicitation Question Development* document. Please refer to this document and the *Kick-off Meeting Notes* document as indicated within the notes section for the questions.

J.1.1 Specific Instructions: Minimum Requirements Prior to Your Elicitation

A1. Provide answers to the questions in the "Base Case Evaluation" area.
A2. Provide mid-value estimates for the question set within the "Regulatory and Utility Safety Culture" area.
A3. Provide mid-value estimates for at least one question set within the "LOCA frequencies of Piping Components" area.
A4. Provide mid-value estimates for at least one question set within the "LOCA frequencies of Non-Piping Components" area.
A5. Provide mid-value estimates for at least one question set within the "LOCA Probabilities of Piping Components under an Emergency Faulted Load" area.
A6. Provide mid-value estimates for at least one question set within the "LOCA Probabilities of Non-Piping Components under an Emergency Faulted Load" area.
A7. Categorize the uncertainty ranges (90% coverage intervals) associated with your mid-value estimates in A2 – A6 as low, medium, or high.

J.1.2 Specific Instructions: Additional Questions During Your Elicitation

We will be asking for your response within the following general areas.

B1. Provide rationale and discuss those important issues that you identified and quantified in questions A2 – A6.
B2. Quantify the uncertainty ranges (90% coverage intervals) associated with estimates provided for A2 – A6. This will quantify the initial responses in A7.
B3. Provide mid-value estimates for the question sets that you did not initially answer in A2 – A6.
B4. Quantify uncertainties associated with answers in B3.
B5. Ensure that the critical issues for LOCA frequencies are captured.

J.2 Elicitation Questions

J.2.1 Elicitation Question 1: Base Case Evaluation

The following questions will be asked to solicit your opinion about the base case evaluation. These questions are necessary to determine how the rest of your responses will be anchored, i.e., if the base case ***conditions*** will be used for anchoring or if you prefer to anchor using a ***specific set of results***. Therefore, you should only consider the ***general approach used by each base case team member*** and not specific results. You will be given additional opportunity later to provide your assessment of the specific results. Of course, you can also provide additional information that can be used to provide anchoring to the rest of your estimates, including your own set of results for the base cases. All the questions below will be asked for each of the LOCA size categories and the evaluation time periods.

1A.1. Do you think the base case results reflect the same conditions in each of the four team member's calculations?

1A.2. If not, which experts' results best describe the base case conditions?

1B.1. Do you think that the differences in the four base case team members' results are a reasonable reflection of the range of variability in the true LOCA probability?

1B.2. If not, do the results under or over estimate your opinion of the true uncertainty?

1C. Do you think that any particular base case results are more accurate?

1D. Do you wish to anchor your responses on either the base case conditions, or on a specific team member's results?

J.2.2 Elicitation Question 2: Safety Culture

2A.1. Consider the current utility safety culture that exists after approximately 25 years (current day) of plant operation and how it influences Category 1 LOCAs. Express the relative change, or ratio, in the utility safety culture's effect on LOCA frequencies after 15 additional years (40 years of operation) compared to its current day effect. Next, express the ratio of the utility safety culture's effect on LOCA frequencies ratio in 35 years (60 years of plant operation) to its current day effect. Include the 90% coverage interval for all estimates.

2A.2. Repeat 2A.1 but now considering the effect of the regulatory safety culture on LOCA frequencies.

2A.3. If you believe that safety culture effects are a function of leak rate category, repeat 2A.1 and 2A.2 for Category 2 through Category 6 LOCA frequencies.

2A.4. Do you believe that the utility safety culture and regulatory safety culture are correlated? If so, is the correlation high, medium, or low?

Notes:

a. Some aspects of regulatory and utility safety culture are discussed in the *Kick-off Meeting Notes*. These aspects can be considered independently and then combined or the aspects can be considered in the aggregate in question 2A.

b. Please see EQ 9 in the *Elicitation Question Development* document for additional information and an example for this question.

c. A ratio greater than 1 indicates that the safety culture will result in an proportional increase in the future LOCA frequency compared to the current LOCA frequency. Similarly, ratios less than 1 indicate that the LOCA frequencies will decrease as a function of safety culture.

J.2.3 Elicitation Question 3: LOCA Frequencies of Piping Components

Question Set 3A

3A.1.1. Consider Category 1 LOCAs for the PWR cold leg reference case. Choose a base case to compare with this reference case for this LOCA size at 25, 40, and 60 years of plant operation. Determine the ratio of LOCAs in the cold leg reference case to the chosen base case at each time period. Also, estimate the 90% coverage interval for these ratios.

3A.1.2. Repeat 3A.1.1 for each LOCA size category for the cold leg.

3A.1.3. Repeat 3A.1.1 and 3A1.2 for all other PWR and BWR reference cases.

Notes:

a. Piping base and reference case conditions are described in the *Kick-off Meeting Notes* document.

b. Please see EQ 10 in the *Elicitation Question Development* document for additional information and an example for this question.

c. Any base case can be chosen at any specified period of time (25, 40, and 60 years) for anchoring. Please note the time period of your chosen base case time period is different than the time period being analyzed.

3A.2.1. List the specific combinations of the **variables** (i.e., material, geometry, degradation mechanism, loading, and mitigation/maintenance) which are the most significant contributors to PWR cold leg Category 1 LOCA frequency as a function of plant operating time (25, 40, 60 years). The list should total **at least 80%** of the total contribution to all cold leg Category 1 LOCAs. Estimate the mid-value contribution of these systems (> 80%). Also, provide the 90% coverage interval for the total contribution estimate of these systems.

3A.2.2. Repeat 3A.2.1 for each LOCA size category for the cold leg.

3A.2.3. Repeat 3A.2.1 and 3A.2.2 for all other PWR and BWR LOCA-susceptible piping systems.

Notes:

a. The list of possible values for each variable class is provided in the *Kick-off Meeting Notes* document.

b. Please see EQ 4 in the *Elicitation Question Development* document for additional information and an example for this question.

3A.3.1. Estimate the relative LOCA likelihood, or ratio, of each unique variable combination for Category 1 cold leg LOCA developed in 3A.2 to the cold leg reference case (or another suitable base or reference case) as a function of plant operating time (25, 40, 60 years).

3A.3.2. Repeat 3A.3.1 for each LOCA size category for the cold leg.

3A.3.3. Repeat 3A.3.1 and 3A.3.2 for all PWR and BWR LOCA-susceptible piping.

Notes:

a. Piping reference case conditions are described in the *Kick-off Meeting Notes* document.

b. Please see EQ 6 in the *Elicitation Question Development* document for additional information and an example for a similar question.

Question Set 3B

3B.1.1. List the PWR piping systems that provide a minimum of 80% of the total contribution for Category 1 (leak rates > 380 lpm [100 gpm]) LOCAs in US plants after 25, 40, and 60 years of operation. Now estimate the mid-value contribution of these systems (> 80%). Provide the 90% coverage interval for the total contribution estimate of these systems.

3B.1.2. Repeat 3B.1.1 for Category 2 through 6 LOCAs in PWR piping systems.

3B.1.3. Repeat 3B.1.1 and 3B.1.2 for BWR piping systems.

Notes:

a. Relevant BWR and PWR piping systems are described in the *Kick-off Meeting Notes* document.

b. Please see EQ 1 in the *Elicitation Question Development* document for additional information and an example for a similar question.

3B.2.1. Estimate the percentage contribution for each PWR piping system in 3B1.1 for Category 1 LOCAs as a function of plant operating time.

3B.2.2. Repeat 3B.2.1 for Category 2 through 6 LOCAs in PWR piping systems.

3B.2.3. Repeat 3B.2.1 and 3B.2.2 for BWR piping systems.

Notes:

a. Please see EQ 2 in the *Elicitation Question Development* document for additional information and an example for a similar question.

3B.3. If a base case piping system(s) is listed within your significant PWR piping systems for Category 1 LOCAs (3B.1.1), go to 3B.5. If not, go to 3B.4.

3B.4.1. Estimate the ratio of the reference case for the Category 1 LOCA contribution of your most important BWR piping system to a suitable base case as a function of plant operating time. Also, provide the 90% coverage range for this ratio.

3B.4.2. Repeat 3B.3 for Category 2 through 6 LOCAs in PWR piping systems.

3B.4.3. Repeat 3B.3 for BWR piping systems.

Notes:

a. Base and reference case conditions for piping systems are defined within the *Kick-off Meeting Notes* document.

b. Please see EQ 10 in the *Elicitation Question Development* document for additional information and an example for a similar question.

3B.5.1. Estimate the ratio of all Category 1 LOCA contributions for this piping system to those contributions represented by the base (or reference) case conditions as a function of plant operating time. Provide the 90% coverage range for this ratio.

3B.5.2. Repeat 3B.3 for Category 2 through 6 LOCAs in PWR piping systems.

3B.5.3. Repeat 3B.3 for BWR piping systems.

Notes:

a. Please see the appendix for an example for this question.

J.2.4 Elicitation Question 4: LOCA Frequencies of Non-Piping Components

Question Set 4A

4A.1.1. Examine the failure scenarios for each of the five PWR non-piping components (pressurizer, valves, pumps, reactor pressure vessel, and steam generator). For each component, list the failure scenarios that provide a minimum of 80% of the total contribution for Category 1 (leak rates > 380 lpm [100 gpm]) LOCAs in US plants after 25, 40, and 60 years of operation. Estimate the mid-value contribution of these failure scenarios (> 80%). Also, provide the 90% coverage interval for the total contribution estimate of these systems.

4A.1.2. Repeat 4A.1.1 for Category 2 through 6 LOCAs for the non-piping PWR components.

4A.1.3. Repeat 4A.1.1 and 4A.1.2 for BWR non-piping components (valves, pumps, reactor pressure vessel).

Notes:

a. A failure scenario is associated with a specific non-piping component, material, degradation mechanism, etc.

b. Relevant BWR and PWR non-piping failure scenarios and components are discussed in the *kick-off meeting notes* document (called failure modes instead of scenarios in this document). These are also summarized in the elicitation summary tables.

c. Please see EQ 1 in the *Elicitation Question Development* document for additional information and an example for a similar question.

4A.2.1. Choose a piping or non-piping base case which results in the most natural comparison for each of the failure scenarios described in 4A.1.1 for all five PWR non-piping component classes. Provide a mid-value estimate of the ratio for the Category 1 LOCA contribution of the chosen non-piping failure scenario to the chosen base case.

4A.2.2. Repeat 4A.2.1 for Category 2 through 6 LOCAs for the non-piping PWR components.

4A.2.3. Repeat 4A.2.1 and 4A.2.2 for BWR non-piping components (valves, pumps, reactor pressure vessel).

Notes:

a. Please see EQ 6 in the *Elicitation Question Development* document for additional information and an example for a similar question.

b. Non-piping base cases are currently being quantified to determine the leaking frequencies due to **all degradation mechanisms** for each non-piping component listed in the *kick-off meeting notes* document. There will also be non-piping base cases frequencies for items that have failed such as steam generator tube ruptures. Additionally, non-piping base cases can still be chosen for making relative comparisons. For instance if an expert considers valve body failure due to cavitation erosion to be significant for Category 1 PWR LOCAs, then valve body leakage can be chosen as the base case.

Question Set 4B

4B.1.1. List the PWR non-piping failure scenarios that provide a minimum of 80% of the total contribution for Category 1 (leak rates > 380 lpm [100 gpm]) LOCAs in US plants after 25, 40, and 60 years of operation. Now estimate the mid-value contribution of these failure scenarios (> 80%). Also, provide the 90% coverage interval for the total contribution estimate.

4B.1.2. Repeat 4B.1.1 for Category 2 through 6 LOCAs for the non-piping PWR failure scenarios.

4B.1.3. Repeat 4B.1.1 and 4B.1.2 for BWR non-piping failure scenarios.

Notes:

a. This question differs from Elicitation Question 4A in that only the significant failure scenarios, regardless of component, need to be considered.

b. A failure scenario is associated with a specific non-piping component, material, degradation mechanism, etc.

c. Relevant BWR and PWR non-piping failure scenarios are discussed in the *kick-off meeting notes* document (called failure modes instead of scenarios in this document). These are also summarized in the elicitation summary tables.

d. Please see EQ 1 in the *Elicitation Question Development* document for additional information and an example for a similar question.

4B.2.1. Estimate the percentage contribution for each PWR non-piping failure scenario in 4B1.1 for Category 1 LOCAs after 25, 40, and 60 years of operation.

4B.2.2. Repeat 4B.2.1 for Category 2 through 6 LOCAs for the non-piping PWR scenarios.

4B.2.3. Repeat 4B.2.1 and 4B.2.2 for BWR non-piping failure scenarios.

Notes:

a. Please see EQ 2 in the *Elicitation Question Development* document for additional information and an example for a similar question.

4B.3.1 Pick either a piping or a non-piping base case (or a piping reference case) for comparison with one or more of your significant non-piping failure scenarios from 4B1.1 for Category 1 LOCAs. The comparison should be natural, but if possible, should be made with one of the most significant failure scenarios that you listed. Provide a mid-value estimate of the ratio of the non-piping failure scenario to the chosen base case as a function of operating time (40 and 60 years). Also, provide the 90% coverage range for this ratio.

4B.3.2 Repeat 4B.3.1 for Category 2 through 6 LOCAs for the non-piping PWR failure scenarios.
4B.3.3 Repeat 4B.3.1 and 4B.3.2 for BWR non-piping failure scenarios.

Notes:

a. Base case conditions for piping systems and are defined within the *kick-off meeting notes* document. Base case conditions for non-piping components are being developed as discussed in the notes to Elicitation Question 4A.2
b. Please see EQ 6 in the *Elicitation Question Development* document for additional information and an example for a similar question.

J.2.5 Elicitation Question 5: LOCA Probabilities of Piping Components under an Emergency Faulted Load

An emergency faulted load represents an initial design consideration for a possible large transient load that was not expected to occur over any particular plant's operating life of 40 years (rare event), or a frequency less than approximately 0.025 yr^{-1}. These loads could be due to seismic loading or any other large pressure transients. Base cases have been developed which examine the conditional failure probability for ASME Service Level B loading. This loading level was estimated for several plants to conservatively approximate a 1*SSE (safe shutdown earthquake) event on a pipe which is flawed up to the allowable technical specification leakage rate for the given piping system and degradation mechanism. An SSE event was initially a design-level earthquake amplitude that was thought to occur once in 40 years; however, operating experience to date suggests that the frequency of an SSE event occurring is less than that.

This question will ask you to list and quantify the effect of the most significant piping systems and degradation mechanisms that contribute to each LOCA category. The quantification will be done for two emergency faulted load sizes (ASME Service Levels B and D) for three assumed damage states. The damage states will consist of tech. spec. leakage rates, the onset of leakage through a slow drip (perceptible leak), and a surface crack with $a/t = 0.5$. The surface crack length will be assumed by each expert and may be a function of degradation mechanism and material. A relationship between the failure loads for a circumferential through-wall-crack and surface cracks with $a/t = 0.5$ and different lengths is provided in the "*Piping Seismic Base Cases*" document. The likelihood of each damage state will also be ascertained by each expert relative to the service history data for the leak-rate frequencies corresponding to each system listed, regardless of degradation mechanism. This assessment will require nine different relative comparisons for each LOCA size category and plant type (BWR or PWR).

The appendix of this document and the "*Piping Seismic Base Cases*" document provide the philosophy behind the seismic piping elicitation questions and detail the seismic piping base case calculations. Both documents should be read prior to answering this elicitation question.

5A.1.1. List the piping systems and degradation mechanisms which most significantly contribute to Category 1 LOCAs given that an assumed emergency faulted load occurs with an equivalent magnitude of an ASME Service Level B (SLB) event for PWRs. This total list should summarize at least the top 80% contributing factors to Category 1 LOCAs under assumed faulted loading conditions. Also, for each system, list the loads which may result in SLB loading and indicate if they are primary (loading-controlled) or secondary (displacement-controlled). Provide the total contribution and also the 90% coverage interval for this estimate.
5A.1.2. Repeat 5A.1.1 for ASME Service Level D (SLD) loading
5A.1.3. Repeat 5A.1.1 and 5A.1.2 for each PWR LOCA size category.
5A.1.4. Repeat 5A.1.1 - 5A.1.3 for BWRs.

Notes:

a. Information on relevant piping systems, degradation mechanisms, and piping sizes is contained in the "Elicitation Meeting Notes" from the kick-off meeting.

b. In this question, pick your piping systems assuming that the pipes will completely fail. Therefore, the LOCA size category will be directly related to the pipe size.

5A.2.1. Pick a representative set of seismic base-case conditions to use for comparison for each of your significant contributors to Category 1 LOCAs in PWRs.

5A.2.2. Repeat 5A.2.1 for each PWR LOCA size category.

5A.2.3. Repeat 5A.2.1 and 5A.2.2 for BWRs.

Notes:

a. A PWR and BWR base case have been defined in the *"Piping Seismic Base Cases"* document for a specific degradation mechanism, pipe size, and material. Additionally, figures are available which show the effects of changing materials, piping size, and service level loading with respect to the base case definitions.

b. Comparisons to the selected base cases will be made in subsequent questions.

c. A relationship between the failure loads for a circumferential through-wall-crack and surface cracks with a/t = 0.5 and different lengths is given at the end of the Piping Seismic Base Case/Background section.

5A.3.1. Consider a single piping system and degradation mechanism combination identified for Category 1 PWR LOCAs in 5.A.1 and the associated seismic base case identified in 5A.2. Determine the ratio of the conditional failure probability (CFP) for this system/degradation mechanism combination (P_{TSL} or $P_{TSL@SLB}$) to the CFP for the chosen seismic piping base case (P_{BC}). Assume that a SLB emergency faulted load occurs and that the piping system is degraded by a through-wall crack that is leaking at the technical specification limit. Also provide the 90% coverage interval of this ratio.

5A.3.2. Consider the same piping system and degradation mechanism combination identified for Category 1 PWR LOCAs in 5.A.3.1. Next, determine the ratio of the CFP for a crack that has just formed a perceptible leak (P_{PL}) to the CFP for a crack leaking at the technical specification limit assuming (P_{TSL}) a SLB load. Also provide the 90% coverage interval of this ratio.

5A.3.3. Again, consider a single piping system and degradation mechanism combination identified for Category 1 PWR LOCAs in 5.A.3.1. Next, determine the ratio of the conditional failure probability for a crack with a maximum depth of 50% of the wall thickness (P_{50}) to the CFP for a crack that has just formed a perceptible leak (P_{PL}) assuming a SLB load. Also provide the 90% coverage interval of this ratio.

5A.3.4. Repeat 5A.3.1 – 5A.3.3 for each significant piping system/degradation mechanism combination listed for PWR Category 1 LOCAs in 5A.1.

5A.3.5. Repeat 5A.3.1 - 5A.3.4 for each PWR LOCA size category.

5A.3.6. Repeat 5A.3.1 - 5A.3.5 for BWRs.

Notes:

a. The leaking crack size is a function of the degradation mechanism and is the major contributor to the differences with the base-case conditional failure probabilities.

b. A perceptible leak is a leak which has just initiated.

5A.4.1. Again consider a single piping system and degradation mechanism combination identified for Category 1 PWR LOCAs in 5.A.1. Next, determine the ratio of the CFP for a SLD event ($P_{TSL@SLD}$) to the CFP for a SLB event ($P_{TSL@SLB}$) assuming a crack leaking at the technical specification limit in both cases. Also provide the 90% coverage interval of this ratio.

5A.4.2. Consider the same piping system and degradation mechanism combination identified for Category 1 PWR LOCAs in 5.A.4.1. Next, determine the ratio of the CFP for a crack that has just formed a perceptible leak (P_{PL}) to the CFP for a crack leaking at the technical specification limit (P_{TSL} or $P_{TSL@SLD}$) assuming a SLD load. Also provide the 90% coverage interval of this ratio.

5A.4.3. Again, consider a single piping system and degradation mechanism combination identified for Category 1 PWR LOCAs in 5.A.4.1. Next, determine the ratio of the conditional failure probability for a crack with a maximum depth of 50% of the wall thickness (P_{50}) to the CFP for a crack that has just formed a perceptible leak (P_{PL}) assuming a SLD load. Also provide the 90% coverage interval of this ratio.

5A.4.4. Repeat 5A.4.1 – 5A.4.3 for each significant piping system/degradation mechanism combination listed for PWR Category 1 LOCAs in 5A.1.

5A.4.5. Repeat 5A.6.1 - 5A.6.4 for each PWR LOCA size category.

5A.4.6. Repeat 5A.6.1 - 5A.6.5 for all BWRs.

Notes:

a. If your system and degradation mechanism list in 5A.1.2 for SLD loading is different from that in 5A.1.1 for SLB loading, pick a seismic base for reference in 5A.4.1 instead of referencing with respect to the SLB loading magnitude.

J.2.5.1 Estimation of Piping Damage Likelihood: Now consider the relative likelihood of the occurrence of a particular level of damage (50% through-wall, perceptible leak, tech. spec. leakage) due to the piping system/degradation mechanism combination chosen in 5A.1. All answers will be ultimately referenced to a piping base-case damage probability as in earlier questions. However, there are no numbers assigned to the base-case damage probabilities at this time, so the comparisons should be made with respect to a piping base-case damage *condition.* A separate piping base-case condition is defined for each piping system and LOCA size category identified in 5A.1, as the service history frequency of *all* leaks *regardless* of the degradation mechanism. This frequency will be determined from service history data.

5A.5.1 Consider a single piping system and degradation mechanism combination identified for Category 1 PWR LOCAs in 5.A.1. Next, determine the ratio of the likelihood of a pipe having a perceptible leak due to that degradation mechanism in that piping system (L_{PL}) after 25 years of operation (L_{PL}) to the base case (L_{BC}), which is the likelihood of a leak due to any degradation mechanism. Also provide the 90% coverage interval for this estimate.

5A.5.2 Consider the same single piping system and degradation mechanism as in 5A.5.1. Next, determine the ratio of the likelihood of a technical specification leak (L_{TSL}) to a perceptible leak (L_{PL}) due to that degradation mechanism after 25 years of operation. Also provide the 90% coverage interval for this estimate.

5A.5.3 Consider the same single piping system and degradation mechanism as in 5A.5.1. Next, determine the ratio of the likelihood of a 50% through-wall leak (L_{50}) to a perceptible leak (L_{PL}) due to that degradation mechanism after 25 years of operation (current day). Also provide the 90% coverage interval for this estimate.

5A.5.4 Now determine if you believe the relative likelihood ratios in 5A.5.1 – 5A.5.3 will increase, decrease, or remain constant with continued operating time. First consider all three likelihood estimates (L_{PL}/L_{BC}, L_{TSL}/L_{PL}, and L_{50}/L_{PL}) at 40 years and then 60 years of continued operation. Determine the ratio of these estimates at 40 years of operation to the current day estimates. Next, determine the ratio these estimates at 60 years of operation to current day estimates.

5A.5.5 Repeat 5A.5.1- 5A.5.4 for each significant piping system/degradation mechanism combination listed for PWR Category 1 LOCAs in 5A.1.

5A.5.6 Repeat 5A.5.1 - 5A.5.5 for each PWR LOCA size category.

5A.5.7 Repeat 5A.5.1 - 5A.5.6 for all BWRs.

J.2.6 Elicitation Question 6: LOCA Probabilities of Non-Piping Components under an Emergency Faulted Load

An emergency faulted load represents an initial design consideration for a large transient load that was not expected to occur over any particular plant's operating life of 40 years (rare event). These loads could be due to seismic loading or any other large pressure transients. Similar to the piping evaluation, base cases will be used for anchoring on the conditional failure probability. However, the actual base cases will not be developed until after the experts' identify the non-piping components which provide significant LOCA contributions. In the interim, each expert should use a particular set of base case conditions for anchoring. More information on this selection will follow in Elicitation Question 6A.2.

This question will ask you to list and quantify the effect of the most significant non-piping systems and degradation mechanisms that contribute to each LOCA category. The quantification will be done for two emergency faulted load sizes (SLB and SLD) for three assumed damage states. The damage states will consist of tech. spec. leakage rates, the onset of leakage through a slow drip (perceptible leak), and a surface crack with a/t = 0.5. The surface crack length will be assumed by each expert and may be a function of degradation mechanism and material. The likelihood of each damage state will also be ascertained by each expert relative to the service history data for the leak-rate frequencies corresponding to each non-piping component listed, regardless of degradation mechanism. This assessment will require nine different relative comparisons for each LOCA size category and plant type (BWR or PWR).

The structure of this question is almost identical to Elicitation Question 5. The appendix contains information on the philosophy behind these two questions.

6A.1.1. List the non-piping components and degradation mechanisms (or failure scenarios) which most significantly contribute to Category 1 LOCAs given that an assumed emergency faulted load occurs with an equivalent SLB magnitude for PWRs. This total list should summarize at least the top 80% contributing factors to Category 1 LOCAs under assumed faulted loading conditions. Also, for each component, list the loads which may result in SLB loading and indicate if these loads are primary (load-controlled) or secondary (displacement-controlled). Provide the total contribution and also the 90% coverage interval for this estimate.

6A.1.2. Repeat 6A.1.1 for Service Level D (SLD) loading

6A.1.3. Repeat 6A.1.1 and 6A.1.2 for each PWR LOCA size category.

6A.1.4. Repeat 6A.1.1 - 6A.1.3 for BWR non-piping components.

Notes:

a. Information on relevant non-piping components and degradation mechanisms, are contained in the "Elicitation Meeting Notes" from the kick-off meeting and subsequent revisions to tables 13 – 17 in this document.

b. In this question, pick your non-piping component assuming that it will completely fail. Therefore, the LOCA size category will be directly related to the component size.

6A.2.1. Pick a representative set of seismic base-case conditions to use for comparison for each of your significant contributors to Category 1 LOCAs in PWRs.

6A.2.2. Repeat 6A.2.1 for each PWR LOCA size category.

6A.2.3. Repeat 6A.2.1 and 6A.2.2 for BWR non-piping components.

Notes:

a. The base case conditions should correspond to at least one (or several, or all) of the significant non-piping LOCA contributors identified in 6A.1. Assume that the component is damaged with a fatigue flaw which results in technical specification leakage. Assume that the base case loading magnitude is an SLB load. Assume that absolute size of this flaw and the actual conditional failure probability to a SLB load magnitude will be quantified at a later date.

b. Comparisons to the selected base cases will be made in subsequent questions.

6A.3.1. Consider a single non-piping component and degradation mechanism combination identified for Category 1 PWR LOCAs in 6.A.1 and the associated seismic base case identified in 6A.2. Determine the ratio of the conditional failure probability (CFP) for this system/degradation mechanism combination (P_{TSL} or $P_{TSL@SLB}$) to the CFP for the chosen seismic non-piping base case assuming (P_{BC}) that an SLB emergency faulted load occurs and that the non-piping component contains a through-wall crack that is leaking at the technical specification limit. Also provide the 90% coverage interval of this ratio.

6A.3.2. Consider the same non-piping component and degradation mechanism combination identified for Category 1 PWR LOCAs in 6.A.3.1. Next, determine the ratio of the CFP for a crack that has just formed a perceptible leak (P_{PL}) to the CFP for a crack leaking at the technical specification limit (P_{TSL}) assuming a SLB load magnitude. Also provide the 90% coverage interval of this ratio.

6A.3.3. Again, consider the single non-piping component and degradation mechanism combination identified for Category 1 PWR LOCAs in 6.A.3.1. Next, determine the ratio of the CFP for a crack with a maximum depth of 50% of the wall thickness (P_{50}) to the CFP for a crack that has just formed a perceptible leak (P_{PL}) assuming a SLB load. Also provide the 90% coverage interval of this ratio.

6A.3.4. Repeat 6A.3.1 – 6A.3.3 for each significant non-piping component/degradation mechanism combination listed for PWR Category 1 LOCAs in 6A.1.

6A.3.5. Repeat 6A.3.1 - 6A.3.4 for each PWR LOCA size category.

6A.3.6. Repeat 6A.3.1 - 6A.3.5 for BWR non-piping components.

Notes:

a. The leaking crack size is a function of the degradation mechanism and is the major contributor to the differences with the base-case conditional failure probabilities.

6A.4.1. Again consider a single non-piping component and degradation mechanism combination as identified for Category 1 PWR LOCAs in 6.A.1. Next, determine the ratio of the CFP for a SLD event ($P_{TSL@SLD}$) to the CFP for a SLB event ($P_{TSL@SLB}$). Assume that a crack exists which is leaking at the technical specification limit in both cases. Also provide the 90% coverage interval of this ratio.

6A.4.2. Consider the same non-piping component and degradation mechanism combination identified for Category 1 PWR LOCAs in 6.A.4.1. Next, determine the ratio of the CFP for a crack that has just formed a perceptible leak (P_{PL}) to the CFP for a crack leaking at the technical specification limit (P_{TSL} or $P_{TSL@SLD}$). Assume a SLD loading magnitude in each case. Also provide the 90% coverage interval of this ratio.

6A.4.3. Again, consider the same non-piping component and degradation mechanism combination identified for Category 1 PWR LOCAs in 6.A.4.1. Next, determine the ratio of the CFP for a crack with a maximum depth of 50% of the wall thickness (P_{50}) to the CFP for a crack that has just formed a perceptible leak (P_{PL}). Assume a SLD loading magnitude in each case. Also provide the 90% coverage interval of this ratio.

6A.4.4. Repeat 6A.4.1 – 6A.4.3 for each significant non-piping component/degradation mechanism combination listed for PWR Category 1 LOCAs in 5A.1.

6A.4.5. Repeat 6A.6.1 - 6A.6.4 for each PWR LOCA size category.

6A.4.6. Repeat 6A.6.1 - 6A.6.5 for all BWR non-piping components.

Notes:

a. If your system and degradation mechanism list in 6A.1.2 for SLD loading is different from that in 6A.1.1 for SLB loading, pick a seismic base for reference in 6A.4.1 instead of referencing with respect to the SLB loading magnitude.

J.2.6.1 Estimation of Piping Damage Likelihood: Now consider the relative likelihood of the occurrence of a particular level of damage (50% through-wall, perceptible leak, tech. spec. leakage) due to the non-piping component/degradation mechanism combination chosen in 6A.1. All answers will be ultimately referenced to a non-piping base-case damage probability. However, there are no numbers assigned to the non-base-case damage probabilities at this time. Comparisons should therefore be made with respect to a non-piping base-case damage *condition*. A separate non-piping base-case condition is defined for each non-piping component identified in 6A.1, as the service history frequency of all component leaks regardless of the degradation mechanism.

6A.5.1 Consider a single non-piping component and degradation mechanism combination identified for Category 1 PWR LOCAs in 6.A.1. Next, determine the ratio of the likelihood of the non-piping component having a perceptible leak after 25 years of operation (L_{PL}) due to that degradation mechanism to the base case (L_{BC}), which is the likelihood of a leak due to any degradation mechanism. Also provide the 90% coverage interval for this estimate.

6A.5.2 Consider the same non-piping component and degradation mechanism as in 6A.5.1. Next, determine the ratio of the likelihood of a technical specification leak (L_{TSL}) to a perceptible leak (L_{PL}) due to that degradation mechanism after 25 years of operation. Also provide the 90% coverage interval for this estimate.

6A.5.3 Consider the same single non-piping component and degradation mechanism as in 6A.5.1. Next, determine the ratio of the likelihood of a 50% through-wall leak (L_{50}) to a perceptible leak (L_{PL}) due to that degradation mechanism. Also provide the 90% coverage interval for this estimate.

6A.5.4 Now determine if you believe the relative likelihood ratios in 6A.5.1 – 6A.5.3 will increase, decrease, or remain constant with continued operating time. First consider all three likelihood estimates (L_PL/L_{BC}, L_{TSL}/L_{PL}, and L_{50}/L_{PL}) at 40 years and then 60 years of continued operation. Determine the ratio of these estimates at 40 years of operation to the current day estimates. Next, determine the ratio these estimates at 60 years of operation to current day estimates.

6A.5.5 Repeat 6A.5.1- 6A.5.4 for each significant non-piping component/degradation mechanism combination listed for PWR Category 1 LOCAs in 6A.1.

6A.5.6 Repeat 6A.5.1 - 6A.5.5 for each PWR LOCA size category.

6A.5.7 Repeat 6A.5.1 - 6A.5.6 for all BWR non-piping components.

ATTACHMENT A TO APPENDIX J: ADDITIONAL EXAMPLE FOR ELICITATION QUESTION 3B.5.1

3B.5.1. Estimate the ratio of all Category 1 LOCA contributions for this piping system to those contributions represented by the base (or reference) case conditions as a function of plant operating time. Provide the 90% coverage range for this ratio.

For question set 3B, Expert A has previously listed the following important PWR piping systems and their individual contributions to Category 1 LOCAs after 25 years of plant operation (see Table J.A.1). For this example, this expert does not expect the relative contributions to change after either 40 or 60 years of plant operation.

Table J.A.1 PWR Piping System Contributions for Category 1 LOCAs

Case	Piping System Lines	Percentage Contribution
1	Instrumentation	50
2	Drain Lines	10
3	Reactor Coolant Pressure Hot Leg	10
4	Chemical Volume Control System	10
5	Safety Injection System Accumulator	10

The reactor coolant pressure hot leg has an associated base case. The base case geometry is a 30 inch diameter pipe, manufactured from Type 304 stainless steel with an Alloy 600 safe end. The safe end to pipe weld is a nickel-based bimetallic weld. The base case degradation mechanisms are thermal fatigue and primary water stress corrosion cracking. The loading is pressure, thermal, residual stress, dead weight, with a pressure pulse transient. Expert A next needs to estimate the ratio between all Category 1 LOCAs in the hot leg compared to those represented solely by the base case conditions. His results are summarized in Table J.A.2.

Table J.A.2 Expert A's Ratio Between Entire Piping System and Base Case Contributions for Category 1 PWR Hot Leg LOCAs

Base/ref case	25 Years			40 Years			60 Years		
	5% LB	MV	5% UB	5% LB	MV	5% UB	5% LB	MV	5% UB
SL BC	1	2	3	1	3	4	1	4	5

Expert A believes that the total hot leg Category 1 LOCAs are twice the number represented by those captured by the base case conditions after 25 years of service (present day). However, this ratio increases with time until after 60 years, the total hot leg Category 1 LOCA probability is 4 times what is predicted by the base case conditions. This initial difference is due to the fact that Expert A believes that an equal number of Category 1 LOCAs will be due to thermal fatigue of other piping materials than are represented by the base case (specifically cast stainless steel and stainless steel clad carbon). This expert also believes that these mechanisms will become more apparent with time than either thermal fatigue or primary water stress corrosion cracking of stainless steel material. Future increases are also included for unanticipated mechanisms.

ATTACHMENT B TO APPENDIX J: PHILOSOPHY BEHIND ELICITATION QUESTION 5

The conditional LOCA probability for a given piping system, degradation mechanism, and emergency faulted load can be determined by multiplying the likelihood curve (L, red in Figure J.B.1) by the conditional piping failure probability (P, black in Figure J.B.1) and then integrating over all the possible damage states. This conditional LOCA probability will likely be a function of LOCA size (pipe size), piping system, applied emergency faulted load, and degradation mechanism. Figure J.B.1 above provides a schematic for a fixed ASME Service Level B (SLB) load, piping system (instrument lines), LOCA size (Category 1), and degradation mechanism (PWSCC). The curve shapes/trends in Figure J.B.1 are an illustration and do not represent any expert opinion. A separate set of likelihood and conditional failure probability curves exists for each unique combination of these four variables.

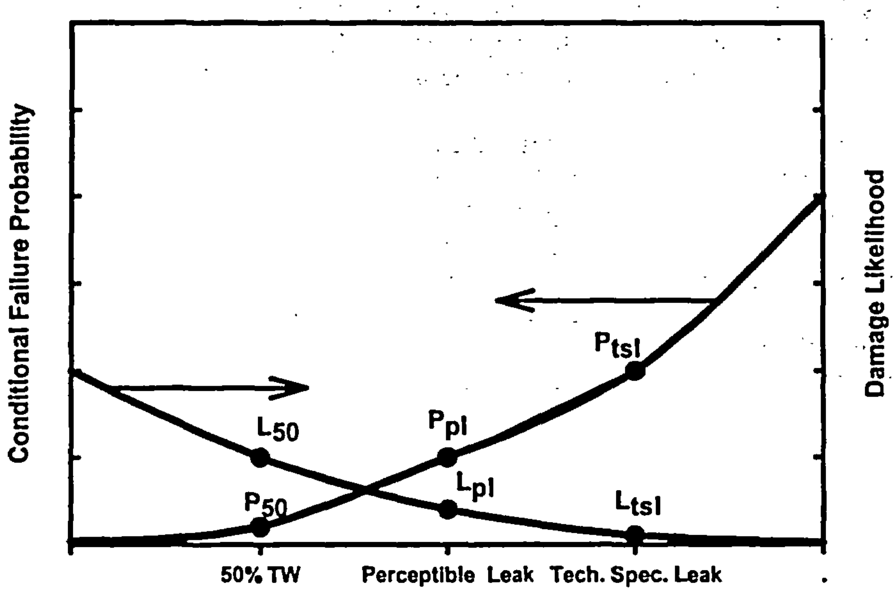


Figure J.B.1 Conditional Failure Probability for Service Level B Loading for Category 1 LOCAs in PWR Instrument Lines due to PWSCC

The elicitation question first asks each expert to identify only the significant piping system and degradation mechanisms to conditional LOCA for each LOCA size category (Elicitation Question 5A.1). Then, the expert will pick seismic base case conditions for either the PWR hot leg or BWR feedwater seismic base cases described in the "*Seismic Base Case*" document (Elicitation Question 5A.2).

Next, the expert will provide ratios of the relative likelihood and conditional failure probabilities (Figure J.B.1) among three different damage states (a 50% through-wall crack, a perceptible leak, and a technical specification leak) at a fixed load for each LOCA size category. All the estimates are initially for a Service Level B load. Here are the ratios that will be provided: P_{tsl}/P_{bc} (Elicitation Question 5.A.3.1),

P_{pl}/P_{tsl} (Elicitation Question 5A.3.2), and P_{50}/P_{pl} (Elicitation Question 5A.3.3). The P_{bc} value is defined in the "*Seismic Base Case*" document while all other variables are defined in Figure J.B.1. In this way, all answers are linked back to the case quantification of the base-case conditions. The next elicitation question (5A.4.1) asks for the relationship between the Service Level D and Service Level B event for the given degradation mechanism and system which is the ratio of $P_{tsl\ at\ SLD}/P_{tsl\ at\ SLB}$. The other questions ask for the ratio of P_{pl}/P_{tsl} (Elicitation Question 5A.4.2), and P_{50}/P_{pl} (Elicitation Question 5A.4.3) for solely the SLD loading event.

Finally, the likelihood of the 3 damage states is estimated. The likelihood base case to be used for each piping system and degradation mechanism listed by each expert will be the leaking frequency for that system over the piping size range of interest for all degradation mechanisms (L_{bc}). Obviously, this frequency will be more than the frequency for any single degradation mechanism, but dominant degradation mechanisms may provide a ratio close to 1. This frequency has not been quantified, but it will be after the elicitation once a complete listing of important systems and degradation mechanisms has been provided by the experts. However, leaking rate propensity has been provided as part of the piping base case analysis for those systems (i.e., see Bill Galyean and Bengt Lydell's results). The elicitation questions ask each expert to provide the following three likelihood ratios: L_{pl}/L_{bc} (5A.5.1), L_{tsl}/L_{pl} (5A.5.2), and L_{50}/L_{pl} (5A.5.3). These variables are defined in Figure J.B.1.

The curve is developed for the 3 damage states and 2 loads in an attempt to capture the most significant contributions to the conditional LOCA probability. These events will be interpolated and extrapolated as necessary to develop continuous relationships as a function of damage state and loading magnitude. This information can then be combined with a plant's seismic hazard curve as well as knowledge about the relationship between the hazard curve and actual piping stresses to determine actual LOCA frequencies due to a seismic event. They could also be used to determine the LOCA contribution of other large transients knowing both the transient frequency and the relationship between the transient applied loading magnitude and the ASME service level loading magnitudes.

APPENDIX K

GENERAL APPROACH AND PHILOSOPHY OF EACH PANEL MEMBER

APPENDIX K

GENERAL APPROACH AND PHILOSOPHY OF EACH PANEL MEMBER

In this appendix the general approach and philosophy that each expert followed as part of this elicitation exercise is presented.

BRUCE BISHOP

For PWR piping frequencies, the median probability of a 19,000 lpm (5,000 gpm) leak after 40 years of operation comes from the average point estimate for 7 plants that used the PFM methodology for the Westinghouse Owners Group (WOG) Piping Risk-Informed ISI (WCAP-14257, Rev. 1-NP-A, Supplement 1). These seven plants were selected to provide a representative sampling of all plants with a Westinghouse NSSS design. Characteristics considered in the sampling included number of primary loops, old and new design vintage and foreign and domestic utility operators. The variability in 40-year probability with leak-rate comes from a WOG supported sensitivity study that reflected both the decrease in probability with increasing leak rate of one pipe size and the number of pipes of a given size that could contribute to a given leak rate. All piping leak probabilities consider the effects of leak-before-break with a minimum detectable leak of 3.8 lpm (1 gpm) per typical plant tech-spec requirements. The increase in failure probability in going from 40 years to 60 years of operation is based upon another WOG sensitivity study. This study and its results are described in a paper presented at the 1999 Pressure Vessel and Piping Conference of ASME and included in PVP-Volume 383.

Non-Piping Frequencies are based upon the degradation mechanism of primary water stress corrosion cracking (PWSCC) initiation and through-wall growth, which is currently the primary cause of unexpected leaks in non-piping components in the primary system. Most other degradation mechanisms are being effectively mitigated. The relative frequencies by component type are based upon a proprietary best-estimate of PWSCC susceptibility by Westinghouse experts for unmitigated Alloy 600/182 base/weld metal. The uncertainties are based upon the variability between the best-estimate susceptibility for PWSCC and observed leak experience.

VIC CHAPMAN

In order to derive a basic set of failure probabilities, the values generated by the 'Base Cases' analysis were initially considered. However, in the end, a decision was made to use the results from some previous work that involved a 'Risk-Informed ISI' application. That work considered a full plant assessment using fatigue as the basic degradation mechanism. Initially, the results from this full plant assessment were compared with the appropriate base cases in order to ascertain whether they were in general agreement with each other. Once it was decided that the two sets of results were in agreement, it was decided to proceed with using the full plant assessment results. These results provided a set of pipe weld failures over a full range of pipe weld sizes that could be considered as a form of global values for each weld size. Factoring would then be from this base set.

Since the leak rate, given a failure, is independent of the failure probability, this can be evaluated separately to obtain a conditional probability. The basic method developed for the base case was expanded to include lower and upper estimates at each step. These basic steps are as follows:

1 Use expert judgment to estimate the COD, up to full rupture, as a function of defect size.
2 Evaluate the defect cross-sectional area for a given defect size using its associated COD.
3 Evaluate the leak rate from a given defect size using some data supplied by the USNRC.
4 Use expert judgment to assess the distribution of the defect length at failure.
5 Combine Steps 3 and 4 to obtain the conditional probability of a leak rate greater than the prescribed leak rate.

The final probabilities were obtained by combining the conditional probability above with the basic fatigue failure probability.

The effect of leak detection was introduced via a factor that was a function of the leak rate. This reduction factor varied from about 5 for a Category 1 leak, up to about 50 for a Category 6 leak.

For non-weld areas, such as the pump bowls and nozzle crotch corners, the basic probabilities were first factored. Next, the basic steps to derive the conditional leak rates as discussed above were followed to adjust the distributions as appropriate.

The effect of PWSCC was introduced as a multiplying factor on the basic fatigue failure rates. It was assumed that for small pipes, 2 inch diameter, that they would still have a significant contribution from fatigue, but that for the largest pipes, the full three orders of magnitude implied by the PWR-1 Base Case (i.e., hot leg base case) should be applied.

Finally, the failure rates for each system were derived by simply summing over all the elements within a given system.

WILLIAM (BILL) GALYEAN

The approach taken by Bill Galyean is based on the total operating experience of U.S. commercial nuclear power plants. This experience consists of approximately 2,650 LWR-years with zero category-1 (> 380 lpm [100 gpm]) loss of coolant accidents. The average age of these plants is approximately 25 years, with a number of plants being 30-plus years old. During this time, a number of RCS degradation issues have arisen and been addressed, for example, IGSCC in BWRs and thermal fatigue in PWRs. The operating experience therefore indicates that degradation will occur, but it will likely be detected and corrected before it can lead to a catastrophic failure. Consequently, this data is the basis for estimating an average LOCA frequency using a Bayesian update of a non-informative prior distribution. Since both PWRs and BWRs have zero LOCAs, the reasonable assumption is that the two designs share a similar LOCA frequency. The operating experience for the two designs is therefore pooled (i.e., use zero failures and the total 2,650 LWR-years of experience). Assuming the LOCA frequency has been (and will be) relatively constant over time (again, this seems reasonable given the history of degradation mechanisms being detected and subsequently mitigated), the resulting LOCA frequency of 1.9E-4/LWR-year produces a probability of one or more LOCA events in the 2,650 LWR years of experience of 39% (again not unreasonable, given there have been zero LOCA events). By contrast, separating the PWR and BWR experience and analyzing them separately produces LOCA frequencies of 2.8E-4/PWR-yr and 5.6E-4/BWR-yr, and a probability of seeing one or more LOCA events (either PWR or BWR) in 2,650 LWR-years experience of 63%. Again, given that there have been no LOCA events, the first (pooled) estimate seems to be the more realistic.

This assumed LOCA frequency (1.9E-4/LWR-yr) was used for the category 1 LOCA (> 380 lpm [100 gpm]). Note that as defined in the elicitation effort, category 1 LOCA includes all larger size categories. So the approach followed by this panel member was to assume a ½ order of magnitude reduction in frequency for each next larger size category. This general approach (if not the precise value of the reduction) has been followed by virtually every LOCA frequency estimate ever made, and is supported by studies on precursor events documented in NUREG/CR-5750, Appendix J.

The time-independent assumption for the LOCA frequency is also based upon the historical experience, if only qualitatively. There seems to be no doubt that the LOCA frequency fluctuates over the age of the plant, but there is reason to believe it will both increase and decrease over time. The IGSCC experience seems to support the assertion that times of increasing frequency will be followed by times of decreasing frequency as degradation mechanisms are identified, understood, and mitigated. Indeed, even the recent RPV-head corrosion event at Davis Besse supports this model of a LOCA frequency increase as degradation occurs undetected, then a decrease as mitigation programs are implemented (e.g., in the case of Davis Besse, replacing RPV head).

The last issue to be addressed is the allocation of the total LOCA frequency among the systems and components that compose the reactor coolant system. This aspect again relies upon operating experience data, this time in the form of the relative frequency of crack and leak events (i.e., precursor events). Basically, these precursor data were collected from LER and foreign reactor experience, and then sorted by degradation mechanism and RCS subsystem/component. In many cases the information provided on the precursor event was somewhat unclear or incomplete. Also, there is little assurance that all precursor events have been captured. However, assuming there is no bias in the reporting of the events such that the data samples for each subsystem/component are equally representative of the all events for that subsystem/component, then the data can be used to support estimates of the relative contribution from each subsystem/component. That is, the precursor events do not have to be completely reported, just consistently reported. Further, the RCS subsystem/component boundaries have not been clearly defined. Hence, the relative contributions to the overall LOCA frequency would likely change somewhat if the precursor data were reviewed and categorized by a different analyst. Nevertheless, this aspect of the analysis was performed simply to allocate the total LOCA frequency (described above) to the general subsystems/components that make-up the RCS.

In summary, the entire U.S. LWR operating experience is used to estimate an average industry-wide total LOCA frequency. This frequency is used for both BWRs and PWRs, not because they are believed to be the same, but on the basis that the operating experience data do not support different frequencies. Time-independence is assumed using the rationale that variation (both increases and decreases) in the frequency will occur as degradation mechanisms manifest themselves and are subsequently addressed by the industry. This total LOCA frequency is allocated by LOCA size categories using the argument that as pipe-size increases, the LOCA frequency decreases. This argument is supported by a number of studies on precursor data and if nothing else has been reflected in all LOCA frequency estimates since WASH-1400 (1975). The total LOCA frequency is also allocated by RCS subsystem/component using data collected on primary system crack and leak events (although the details of this allocation are view as somewhat subjective with respect to the boundaries of the different subsystems/components).

KAREN GOTT

The approach taken by Karen Gott to the elicitation was to first consider how her experience from the Swedish nuclear fleet was applicable to the US fleet of nuclear power stations. In this respect she took into account the known histories of the various degradation mechanisms which have troubled the two fleets as well as the mitigation methods which have been developed. This led her to amongst other things to the conclusion that the likelihood of an unexpected mechanism leading to failure is probably larger than the likelihood of a known mechanism resulting in failure in a region which is inspected on a regular basis. In general a new area of concern with regard to component degradation has arisen on a seven to ten year cycle over the lifetime of commercial nuclear plants.

To produce the numbers she used her database of failures and degradation in mechanical components in Swedish plants. The degradation mechanisms are the same, but the numerical figures are different because of differences in design and construction. She based her elicitation figures on the number of leaks in proportion to the number of reported cases (many were detected early) and took these to be the current figures for a good safety culture situation. The database includes other mechanical components than pipes, but does not cover steam generator tubes, so she was able to generate figures for pump and valve housings, for example. She then considered the differences in the philosophies concerning qualification and application of inspection programmes between the two countries. This she incorporated into her thinking about the safety culture aspects, both for the current time and for the extrapolation to 40 and 60 years.

DAVE HARRIS

For each plant type and for piping and non-piping Dr. Harris selected a reference system and attempted to scale other systems relative to that reference system. He tried to use estimates based on service experience to the maximum extent, and then scaled the relative frequencies for the LOCA categories using results from the probabilistic fracture mechanics (PFM) analyses. In many instances, service experience was not applicable, so he then relied more on the PFM results. If he felt that a given system was not a significant contributor to the leak frequency for a given LOCA category, then he was less concerned about the accuracy of the frequency estimate for that system.

<u>PWR Non-seismic LOCA:</u> For the PWR case, he used the hot leg as the reference system for large leak flow rates. Service experience is not readily applicable. The PFM analyses for the hot leg showed a very wide range of results depending on the assumptions and input to the analyses. Therefore, he scaled the hot leg results by use of the reactor pressure vessel (RPV) reference case results. Results presented at the wrap-up meeting in February 2004 provided an estimate of the RPV > 1,900,000 lpm (500,000 gpm) as 10^{-10} (per plant-year) for the first 25 years. He doubled this value to account for 2 hot legs. He assumed that the leak frequency for 380 lpm (100 gpm) LOCA is 3 ½ orders of magnitude higher, and then interpolated on a log-log scale. This fixes the frequency-leak rate for the hot leg at 25 years. He assumed that the frequency for > 1,900,000 lpm (500,000 gpm) in the time increment 25-40 years is twice that for 0-25 years, and four times as large in the increment 40-60 years. The leak frequency for 380 lpm (100 gpm) is assumed to be independent of time.

The cold leg is then assumed to have frequencies 1/3 those of the hot leg, because the cold leg operates at a somewhat lower temperature. The surge line is assumed to have leak frequencies 100 times as large as the hot leg, because the surge line sees a lot more cycles than the hot leg, and is just as hot. These estimates then define the very large leak frequencies for the entire plant.

At the low end of the leak rate scale, he assumed the plant results to be bounded by the past service experience for steam generator tubes. An estimate from the wrap-up meeting for the steam generator LOCA frequencies is 3.5×10^{-3} per plant-year.

He used the HPI make-up nozzle as a surrogate for all 2 to 6 inch diameter lines. He used the reference case results from the PFM results that was presented at the wrap-up meeting, but reduced the leak frequencies by an order of magnitude at 19,000 lpm (5,000 gpm). He assumed that the SIS accumulator and RHR systems have about an order of magnitude less contribution than the surge line, so they have a small contribution to the overall plant.

This procedure provides his best estimate. The 5% and 95% estimates are scaled up and down from the best estimate. He estimated the 5% to be 1 ½ orders of magnitude below the best estimate (multiply by 0.03), independent of time and leak rate. He varied the multiplier for the 95% estimate, making it larger for the larger leak categories. The multiplier varied from 30 to 1000. He believes that we have a better handle on the smaller leak rates, because they are bounded by steam generator tubing experience, which is plentiful.

<u>BWR Non-seismic LOCA:</u> He selected the recirculation system for the reference for BWRs. For intermediate leak rates 380 to 95,000 lpm (100 to 25,000 gpm), the 12 inch diameter portion of the recirculation system dominates. He used the base case results from the wrap-up meeting, but reduced them by an order of magnitude, because his PFM analysis underestimated the benefit of the post-remedial action residual stress.

The feedwater system is also important, because it has lots of welds, and is prone to flow accelerated corrosion (which is not related to welds). Since this is a dominant system, he assumed it to be comparable to the surge line in a PWR (which is a dominant system for that type of plant). The steam line is about the same size and same material as the feedwater, but is not prone to flow accelerated corrosion, therefore he assumed the steam line to have leak frequencies that are two orders of magnitude below the feedwater system. He assumed the RHR line to be the same as the PWR surge line, which is about the same size. NUREG/CR-6674 shows very low probabilities of through-wall cracks in the HPCS/LPCS system, so the contribution of this system was assumed to be negligible. The recirculation, feedwater, steam line and RHR are assumed to be the dominant systems, and no estimates were made for other systems.

The estimated uncertainty bands are generally tighter than the PWR estimates, because they are based more on experience for the dominant system (recirculation). They are independent of time, but do vary somewhat with leak category.

<u>Non-piping:</u> Dr. Harris felt less confident making estimates for non-piping components, because most of his experience is related to piping. He relied heavily on results provided by others in the wrap-up meeting, and used CRDM nozzle PWSCC, rector pressure vessel and steam generator tubing data for reference purposes. He also scaled relative to piping in many instances. He did not estimate time dependencies. For instance, for pumps and valves, he figured that they are less failure prone than the piping system in which they are located (passive failures). He estimated probabilities, and then calculated relative contributions of failure scenarios.

BENGT LYDELL

The approach is documented in "Base Case Report No. 2." For systems other than the five Base Case Systems, the base case results established anchor distributions for BWR and PWR Code Class 1 reference piping systems. As an example, the base case results for PWR hot legs were applied to PWR cold legs but adjusted to account for insights about the service conditions and degradation susceptibility specific to cold legs. For the other BWR and PWR piping systems not covered by the base case study, the base case results were adjusted downwards or upwards as appropriate by accounting for unique piping design features (e.g., size, material, and weld population), service conditions and field experience. For non-piping passive components the base case report again was used as the main reference (or source of calibration parameters) in combination with reviews of relevant service experience. In summary, this Panel Member's response to the elicitation questions is based on insights from degradation mechanism analyses in combination with reviews and statistical evaluations of service experience data.

SAM RANGANATH

One of the challenges in the LOCA frequency estimation is trying to predict the probability of an event that has never happened before, but which has enormous consequences if it did. It is important to maintain a sense of balance in this effort and aim for a realistic approach that is based strictly on technical considerations. As in any probabilistic analysis, the success depends on how realistic the inputs are and how the approach reflects actual field experience. Having worked in the BWR industry for almost 30 years, I felt that my most important contribution to the elicitation process is make sure that that frequency estimates reflect BWR field experience. For example, use of probabilistic defect distribution data is acceptable as long as the prediction is consistent with actual field behavior. My philosophy was to start from actual field data and to predict future behavior based on my understanding of failure mechanisms, mitigation measures and BWR systems design. Since my knowledge is mainly on BWR systems, I focused my attention on BWRs rather than PWRs. I did not want to speculate in areas where I did not necessarily have the expertise. There are other people who are more knowledgeable about PWRs and they can do a better job on the estimates. I felt that the diversity of the elicitation panel and their expertise and the open mindedness of the NRC team helped in coming up with the best estimates.

PETE RICCARDELLA

The first step in the expert panel elicitation was to develop an amalgamated set of base case LOCA frequencies upon which the elicitation responses are anchored. The generic base case LOCA frequencies developed for the panel represented the work of four teams: two teams used an empirical approach based on operating plant experience with leakage and other precursor events, while two other teams used theoretical, probabilistic fracture mechanics analyses. Each of these approaches has different strengths and weaknesses, such that a better estimate of base case LOCA probabilities can be achieved by selectively combining the results in a manner that optimizes the strengths of both. The method and rationale for combining the base case results of the individual teams were documented, ultimately producing a revised set of LOCA frequencies for the five piping base cases.

LOCA frequencies for each of the LOCA sensitive piping systems identified for PWRs and BWRs were then estimated. This was done by picking the base case which is most representative of the specific LOCA sensitive system, considering plant type, material of construction, operating conditions and relevant degradation mechanisms, and then scaling the base case frequencies for each LOCA category based on judgment of any substantive differences between the base case and the system under evaluation. One of the main factors accounted for in this scaling process was differences in the size of the systems, in terms of number of welds of various pipe sizes (based on a system-by-system weld census that was provided to the panel). Scaling was also used to account for system specific factors, such as whether repairs or mitigation have been applied to address degradation mechanisms considered in the base cases, and the timing of such actions. An estimate of the probability of breaks in small diameter socket welded piping (instrument, vent and drain lines) due to vibration fatigue was made, which was not included in any of the base cases. This estimate was based on prior experience with this relatively common failure mechanism.

It was felt that there is a relatively large uncertainty band in all of the above probability estimates; plus or minus an order of magnitude. Included in this uncertainty band is the potential development of new, as yet unseen degradation mechanisms in the future, which obviously weren't considered in the base cases.

A set of base cases for non-piping LOCAs was developed, the methodology for which is documented in Appendix I to this NUREG report. These base cases included potential breaks due to small vessel penetrations such as Control Rod Drive nozzles, medium size breaks due to larger diameter nozzles (excluding safe-end ruptures which are included in the piping base cases), and very large breaks due to pressure vessel ruptures (specifically addressing irradiation embrittlement of the RPV). The resulting base cases were then used to estimate contributions to LOCA frequency from non-piping LOCAs.

The detailed rationale used in developing the elicitation response for each system was documented in a report, to permit the reconstruction of the logic in the future if it becomes necessary.

HELMUT SCHULZ

The general approach and philosophy used follows the approached taken by GRS to estimate frequency of LOCA initiating events at passive systems for German PSAs.
The major steps and assumptions of this approach are as follows:

- In principle, wall penetration of pipes which would result into a leak follows in their geometries either

- a slit type penetration originating from cracks caused by fatigue or corrosion or
- a bulging type penetration caused by wall thinning.

Beyond critical dimensions wall penetrating stable defects turn into a full break. This means in practice that for each pipe size there are two or more leak sizes which are of a distinct different probability of occurrence governed by the likelihood of the respective active failure mechanism and the reliability to detect leaks and to take actions to avoid aggravation of the situation e. g. isolation of the leak, stop operation.

The maximum leak size related to a wall penetrating stable defect (undercritical crack, bulge, pit) depends on the actual load specifically the relationship between membrane and bending stresses. The majority of systems being considered in the safety analysis of NPP's fall either into the category of high pressure or low pressure systems. For reasons of simplicity upper bound values can be taken to describe maximum leak sizes connected to wall penetrating stable defects for each pipe size. Based on experimental evidence as well as fracture mechanics calculations the maximum leak size resulting from an undercritical crack is rather limited, expressed in terms of fractions of the pipe cross section it is only a few percent. This approach uses 2 % of the cross section as a rule of the thumb for high pressure systems. Through wall corrosion pits are generally very small. Bulge-type wall penetrations caused by wall thinning have a potential for stable leaks of a considerable size.

- The frequency of leaks is estimated based on the operating experience of the national population of nuclear power plants and in addition the worldwide experience is considered as far as applicable and available. In general, the operating experience give indications of leaks in a sense of precursors for most classes of piping or give indications of zero failures statistics only.

- To estimate the probability of a break (which is connected by the diameter of the piping to the flow rate) a relationship is used to describe the frequency of breaks in relation to the frequency of leaks as the function of the diameter of piping systems being designed to the same design parameters. For the small bore piping (less than 2 inch) the relationship between leak and break is arrived from operating experience. For the largest pipe (main primary pipe) the relationship between leak and break is based on a number of technical arguments and probabilistic fracture mechanics analyses. For the pipe sizes in between a linear relationship is used between the upper and lower bound as described before.

- For reasons of simplicity and in accordance with technical experience it is assumed that within the piping systems only so called leak relevant elements are contributing to the frequencies. These leak relevant elements are essentially welds which are adjacent to changes in geometry (nozzles, branches, reducers etc.) which in itself would introduce enhanced loads and to some extend represent more difficult areas for manufacturing and inspection.

- The whole population of pipes, nozzles and penetrations connected to the main components are divided into subpopulations taking pipe diameters as orientation values, using e. g. 5 or 6 subpopulations to represent the difference pipe sizes, materials and operating conditions. For each subpopulation the frequency of leaks is based according to the procedure described before (operating experience, zero failure statistics), the frequency of leaks is adjusted to the size of the relevant population each time and the frequency of breaks within the subpopulation is estimated using the described relationship.

- The frequency of the different subpopulation which could contribute in a different way (leak or break) to the specified LOCA classes is then summed up. In view of the limitation regarding the verification of very low values of estimated frequencies a cut-off value is used.

FRED SIMONEN

Operating experience was applied as the best method for estimating frequencies for more common failure events such as small detectable leakage and of ruptures for small diameter piping. Operating experience has the advantage of reflecting contributions from all degradation mechanisms and is not limited to a particular mechanism that can be addressed by a probabilistic fracture mechanics model. For lower frequency events, for which there are little or no data from operating experience, the data were therefore supplemented by trends from probabilistic fracture mechanics models. The fracture mechanics models were taken to provide relative frequencies such as for (for a given pipe size) the ratios of frequencies of for different categories of failures (in terms of leak-rates). Similarly, models can indicate ratios for one failure category of leak for differing pipe sizes. Reports of small detectable leakage (from data bases) were taken to be precursor events that can be used to calibrate estimates of frequencies for categories of larger leakage events. Another consideration was that only the larger pipe sizes contribute to the frequencies for larger leak rates. It was implicitly assumed that contributions from smaller pipe sizes dominate for the smaller categories of leak rate frequencies.

Non-nuclear experience was also used to support estimates of failure frequencies for nuclear components. Component designs, materials, construction codes, operating conditions etc. are much the same for nuclear applications and as for many non-nuclear applications. Non-nuclear experience however provides a much larger number of years of plant operation (by orders of magnitudes) than available from nuclear experience. Non-nuclear experience therefore provides additional justification for very low failure frequencies for components such as pump bodies, tube sheets, manways, etc. that imply large extrapolations from the limited years of nuclear plant operation.

GERY WILKOWSKI

The piping non-seismic LOCA evaluations were conducted for PWR and BWR piping separately from a bottom-up approach using reference cases for certain pipe systems in each type of plant. The reference cases were determined from a combination of the base case results supplied to the elicitation panel. The base cases supplied consisted of two independent probabilistic fracture mechanics analyses, and two independent service history evaluations for certain pipe systems. The probabilistic pipe fracture mechanics analyses (PRAISE or PRODIGAL) base cases were not chosen since I did not believe that those computer codes properly determined the probability of a long surface crack occurring, which is the actual way that a LOCA would occur, i.e., a through-wall leaking crack will be readily discovered by leakage before failing at normal operating conditions. Consequently, the two historical base cases were averaged, but only up to 25 years of operation (current time period). I did not believe that the historical based approaches would be that good for extending the LOCA reference cases to 40 or 60 years. Consequently, my reference cases were only for 25-year time period (present), and the 40- and 60-year evaluations were adjusted depending if I thought the particular pipe system would be susceptible to some near term or long-term degradation mechanism (e.g., PWSCC), and if that mechanism could produce a large surface crack. These evaluations were done for 12 different PWR pipe systems and 13 different BWR pipe systems, with six different LOCA flow-rate categories. The uncertainties (5 and 95 percent bounds) in the predictions generally increased as the amount of time increased, i.e., the uncertainty for 25 years was less than 40 years, and the uncertainty in the 40-year predictions was less than for the 60-year predictions.

For non-piping, Dr Wilkowski felt he did not know enough about failure modes of all the different categories of non-piping components (with the exception of a few categories like CRDM nozzle ejection). He therefore chose the steam generator tube historical failure frequencies for small LOCA as a controlling PWR case, but all the other cases were governed by the piping failure probabilities.

APPENDIX L

DETAILED RESULTS

APPENDIX L

DETAILED RESULTS

The detailed results from the elicitation exercise are presented in this appendix. The detailed quantitative responses to the elicitation questions from each of the individual members of the elicitation panel are included with this report at: http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1829. Both the quantitative (i.e., numerical LOCA frequency estimates) and qualitative results (i.e., rationale) are presented in this appendix. The quantitative results are often presented in the form of "box and whisker plots". Box and whisker plots (often referred to simply as "box plots") are a statistical representation of a data array typically showing the median value plus the 10^{th}, 25^{th}, 75^{th}, and 90^{th} percentiles of the array. When overlaid with a horizontal scatter plot of the same data array (see Figure L.1), the left most point is the minimum value and the right most point is the maximum value in the array. The two vertical lines at the ends of the "tails" in Figure L.1 represent the 10^{th} and 90^{th} percentiles. The two ends of the shaded box are the 25^{th} and 75^{th} percentiles and the range in the data set encompassed by the shaded box (i.e., the range between the 25^{th} and 75^{th} percentiles) is referred to as the interquartile range (IQR) of the data array. Finally, the vertical line near the center of the shaded box represents the median value of the array. In the plots included in this appendix, letter designators are often included for the minimum and maximum values (e.g., the letters G and D in the example shown in Figure L.1). The letters designate the code for the panel member whose estimated value was either lowest or highest of all the panel members that provided a response to the particular question.

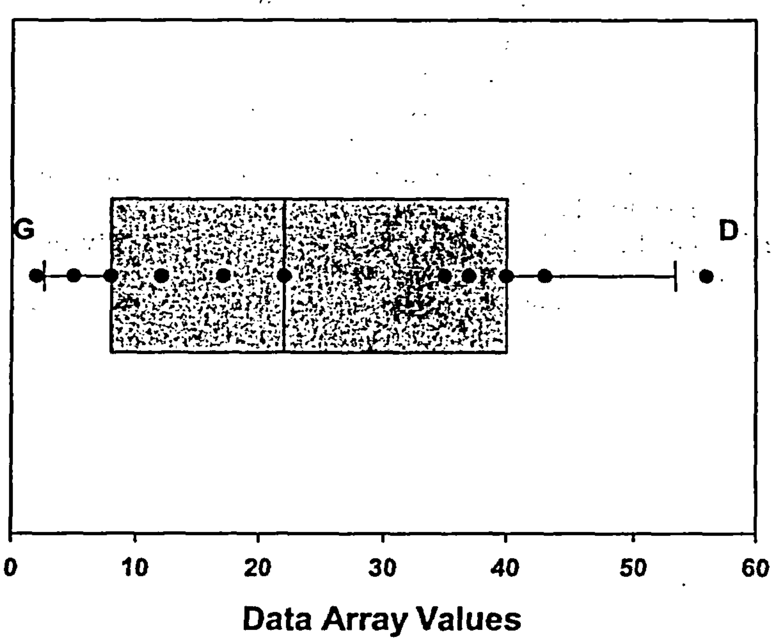


Figure L.1. Example "Box and Whisker" Plot

Generally, the source of the qualitative responses (i.e., the rationale) came from the individual elicitations although there were some opinions expressed during the various group panel meetings that were also included. For each of the individual elicitations, minutes were taken. Minutes were also taken at each of the group panel meetings, see Appendix B. In addition, the participants often provided a handout to lead the discussion at their individual elicitations. After each elicitation, most of the participants also provided formal written responses to the elicitation questions. It was from these minutes, handouts, and written responses that the rationale provided below was gleaned from. In addition, each of the elicitations was audio taped and each meeting was video taped to provide a permanent record of the exercise.

Most of the participants believed that precursor events (e.g., cracks and leaks) were a good barometer of LOCA susceptibility. This is reflected by the fact that almost all of them anchored their responses against some form of the available service history data. A distinct advantage of the service history data is its inclusion of all degradation mechanisms which have emerged to date, whereas the probabilistic fracture mechanics (PFM) approaches only address selected mechanisms. The advantage of the PFM approaches is that they are best suited for addressing LOCA size and operating time effects. A number of participants used the PFM results as a basis for adjusting the service history data in this manner.

A major assumption embodied in this elicitation exercise is that everything (piping and non-piping components) was fabricated in accordance with applicable code standards, e.g., there were no counterfeit bolts used.

L.1 Safety Culture

Figures L.2 and L.3 show the effect of the industry and regulatory safety culture, respectively, on the LOCA Ratio (i.e., the ratio of the LOCA frequency in the future to the LOCA frequency at 25 years) for Category 1 LOCAs. Figures L.4 and L.5 show the effect of industry and regulatory safety culture on the LOCA Ratio for Category 4 LOCAs. Ratios less than 1.0 are indicative of a perceived reduction in the LOCA frequency as a result of improvements in the safety culture mindset. As can be seen in these figures, the panel members overwhelming expected the safety culture to either improve or remain constant over the next ten to fifteen years and beyond. The panel felt that the industry as whole was acting in a consistent manner. However, a few plants with a less diligent safety culture mindset would provide the greatest challenge from a LOCA perspective. It was thought that these outlier plants may not affect the mean trends, but could strongly influence the bounds. The Davis-Besse experience was frequently cited as an example of this effect. The panel also expressed the opinion that the industry and regulatory safety culture are highly positive correlated. Therefore, regulatory and industry changes are expected to occur virtually simultaneous.

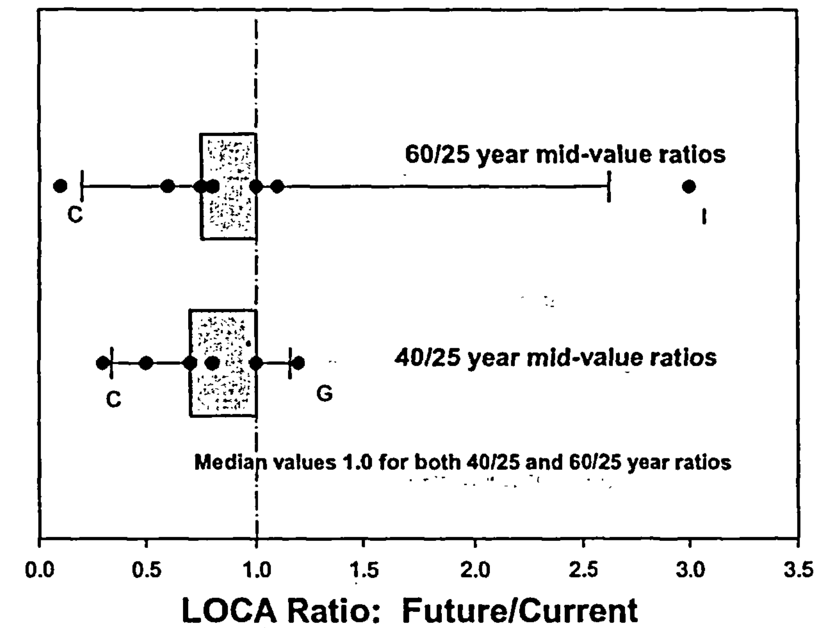


Figure L.2 Effect of Utility Safety Culture on Category 1 LOCAs

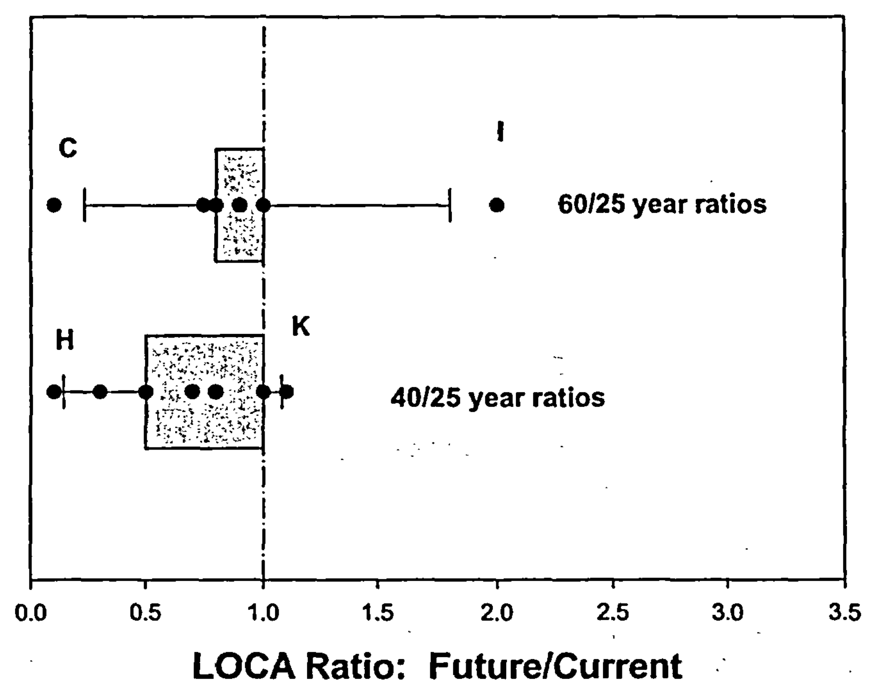


Figure L.3 Effect of Regulatory Safety Culture on Category 1 LOCAs

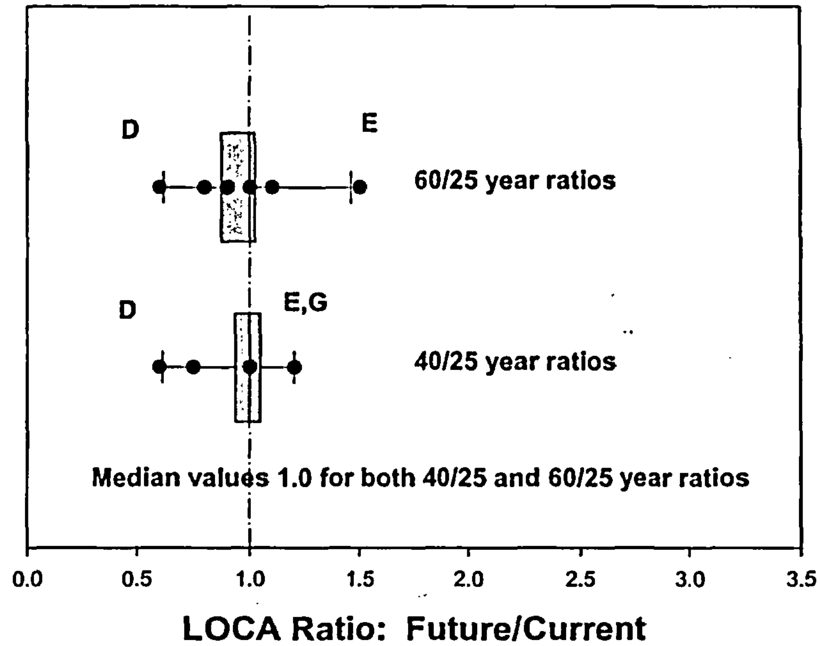


Figure L.4 Effect of Utility Safety Culture on Category 4 LOCAs

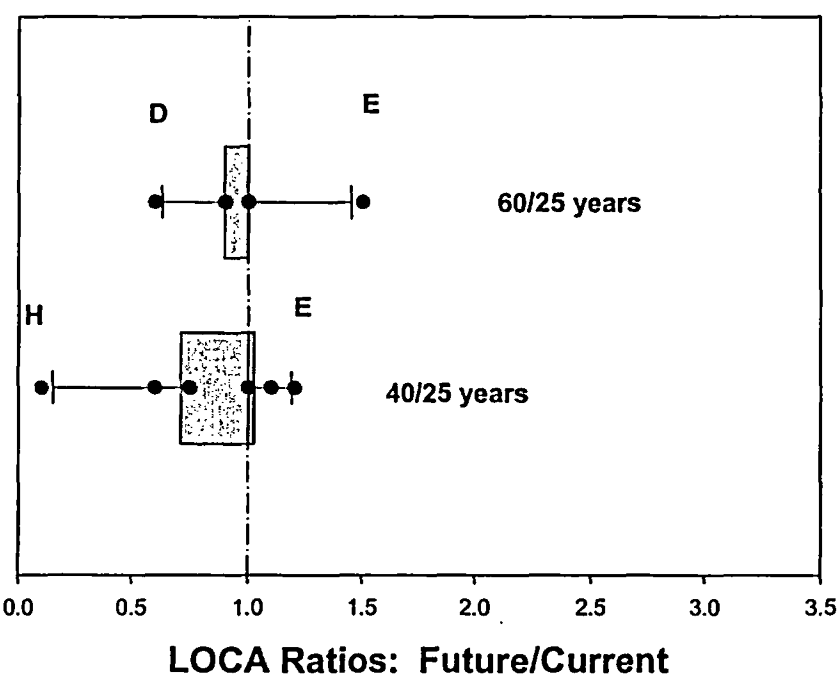


Figure L.5 Effect of Regulatory Safety Culture on Category 4 LOCAs

As can be seen in comparing Figures L.2 with L.4 and L.3 with L.5, the panel members felt that any improvements in safety culture would be more beneficial for the smaller LOCA categories than their larger counterparts because the smaller LOCA categories constitute the bulk of the experience base. The frequency of the larger LOCA categories due to safety culture effects is expected to remain relatively constant over time.

The bottom line from this discussion is that because the panel members felt that the effect of safety culture was relatively minor, the LOCA frequencies developed during this exercise were not modified to account for this effect. The main caveat to this general conclusion is the previously mentioned concern that the LOCA frequencies developed through the elicitation process could be significantly degraded by a safety-deficient plant operating philosophy. The other concern frequently expressed was that the industry safety culture mindset may deteriorate near the end of a plant's operating life as management tries to "squeeze out" the final few years of operations without investing in the necessary maintenance activities. Also, near the end of the plant's life there was a concern expressed that the morale of the plant's operating staff may begin to erode as they foresee a potential loss of employment. These concerns are manifested in the higher LOCA Ratios for the 60/25 year results when compared with the 40/25 year results in Figures L.2 through L.5.

L.2 BWR Piping

The participants generally thought that the important degradation mechanisms for BWR piping were thermal fatigue, flow accelerated corrosion (FAC), and IGSCC. It was argued that BWR plants are more prone to thermal fatigue problems that the PWRs because they experience a greater temperature fluctuation during the normal operating cycle. In BWRs, thermal fatigue is a concern for the feedwater lines, the main steam lines, and the residual heat removal (RHR) system. From a LOCA perspective, thermal fatigue is an important aging mechanism because it does not manifest itself as a single crack, but as a family of cracks over a wide area. As such, it can lead to a large LOCA. Thermal fatigue cracks also tend to propagate rapidly, and since it is not material sensitive (i.e., it can attack a number of materials), it is difficult to prioritize critical areas for inspections.

Only the feedwater piping system is highly susceptible to FAC. The main steam line is the other major carbon-steel piping system which experiences constant fluid flow. However, it is not as susceptible to FAC because the erosion rates associated with two-phase flow are less severe. While FAC caused a serious accident in the secondary side piping at Surry 15 years ago, the panel members generally thought that the industry had inspection programs in place today to prevent the reoccurrence of such an event, especially for the primary side piping systems. However, a number of panel members expressed the concern that the water chemistry improvements which mitigate IGSCC could lead to unexpected FAC problems.

The panel consensus is that the susceptibility to IGSCC is greatly reduced compared to the past. Measures such as improved hydrogen water chemistry, weld overlay repairs, and pipe replacement with more crack resistant materials had essentially reduced the likelihood of IGSCC. However, there is still residual concern about the failure likelihood of the large recirculation piping material that has not been replaced. Furthermore, even for the pipe which has been replaced, the question was raised as to whether the new replaced pipe was immune to this type of degradation, or is the problem simply been move out into the future. The German experience with Type 347 stainless steel was raised in this regard. There was also concern expressed about the effects of increased sulfate levels in the future due to efforts focused at extending the life of some of the filters in the plants.

Another aging mechanism of concern is mechanical fatigue. This is primarily a problem in smaller diameter piping, especially those with socket welds, and is caused by an adjacent vibration source. From a LOCA perspective, it was noted that locations susceptible to mechanical fatigue damage were not always obvious. It is impossible to eliminate all plant vibrations, and furthermore, changing the configuration of the plant can result in newly susceptible areas.

As part of this elicitation exercise a total of 14 LOCA-susceptible piping systems were considered for the BWR plants. Of these, however, most of the participants focused on a few common systems as being the important LOCA contributors. Figure L.6 shows the Category 1 LOCA frequencies for each of these piping systems at 25 years of plant operation (present day). Note, the results for the high pressure core spray and low pressure core spray systems are combined as a single entry in Figure L.6 (HPCS/LPCS). For these smaller category LOCAs, the main concern is with the smaller diameter lines, such as the instrument and drain lines. Most of the participants believe that it is more likely to have a complete break of a smaller diameter line than a comparable size opening in a larger diameter pipe. One reason for this is that for a given crack size, the crack is a larger percentage of the pipe circumference in the smaller diameter pipes, and it was thought that a small diameter pipe was just as likely to have a crack of a certain length as a larger diameter pipe. Furthermore, smaller diameter lines are often fabricated from socket welded pipe which has a history of mechanical fatigue damage from plant vibrations. These lines may also be susceptible to external failure mechanisms arising from human error (e.g., damaging with equipment, such as fork trucks). Finally, these smaller diameter lines are often subject to fabrication flaws and they are typically more difficult to inspect, if they are inspected at all. In-service inspection (ISI) is not routinely performed on these lines. Conversely, the larger diameter lines are inspected more rigorously and routinely.

Besides the instrument and drain lines, the recirculation and, to a slightly lesser extent, the Control Rod Drive (CRD) and Residual Heat Removal (RHR) lines are also of concern, primarily as a result of stress corrosion cracking susceptibility.

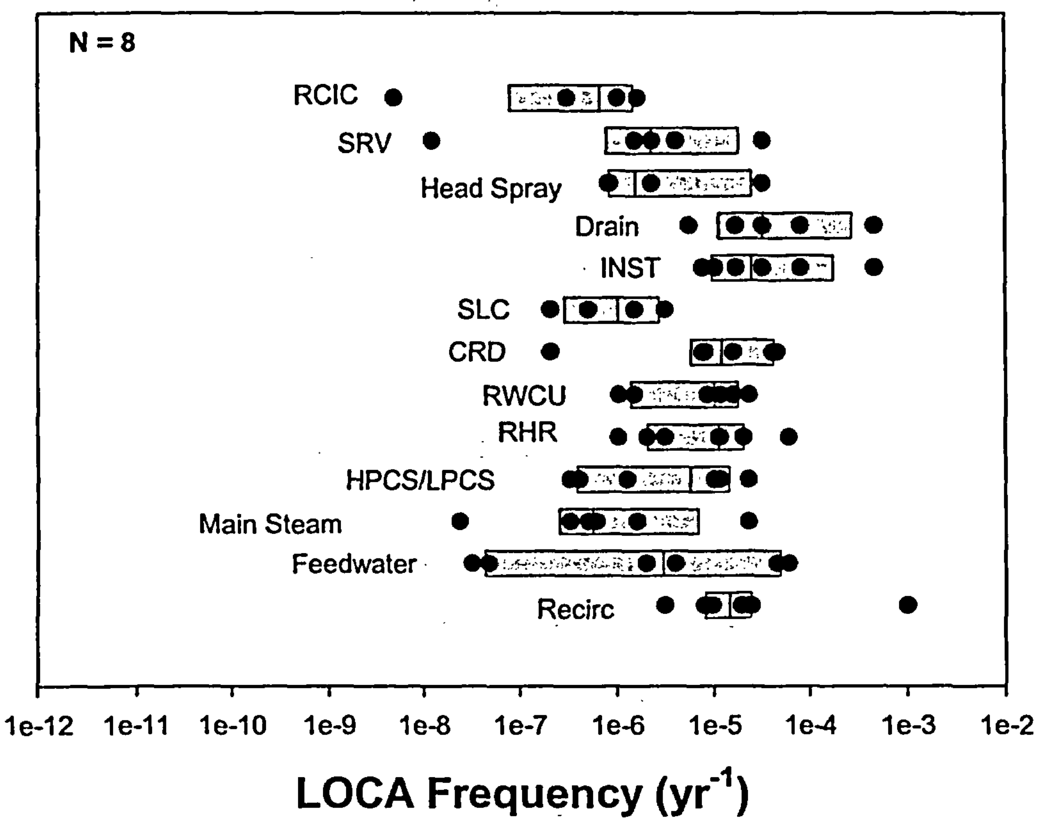


Figure L.6 Category 1 LOCA Frequencies for BWR Piping Systems at 25 Years of Plant Operation

For larger Category 3 LOCAs, the recirculation system was the largest contributor to the overall LOCA frequencies, see Figure L.7. (Note in this figure that the instrument and drain lines, as well as the control rod drive [CRD] lines, are no longer shown in that these smaller diameter lines cannot support a Category 3 LOCA.) The fact that the recirculation system is the largest contributor is a slight departure from the PWR estimates where the smallest diameter piping system that can support a particular LOCA category consistently had the highest LOCA frequencies. The main concern with the recirculation system piping continues to be SCC, even when considering the effective mitigation programs in place today. Of secondary importance were the feedwater, residual heat removal (RHR), reactor water clean-up (RWCU), core spray, and safety relief valve (SRV) systems. There was wide variability expressed for the feedwater system. Several participants thought that its susceptibility was similar to that of the recirculation system while others thought that it would make an inconsequential contribution. This latter group generally thought that the mitigation programs in place for the feedwater system were overall effective. The RHR system was deemed important by some panel members due to the relatively larger number of precursor events reported and the relatively high number of welds. A number of the participants used the weld census data provided to differentiate the relative contributions between systems for those systems that have similar operating experience. The SRV lines were judged to be potentially problematic by four of the eight respondents who addressed the question of BWR piping. They pointed out that the SRV lines are subject to high dynamic loads during the relatively common SRV discharge events, however, only a short section of these lines are actually susceptible to a LOCA event. Overall, in comparing Figure L.6 with Figure L.7, one can see approximately a one order of magnitude reduction in the LOCA frequency between the Category 1 and 3 LOCAs for most of the BWR piping systems considered.

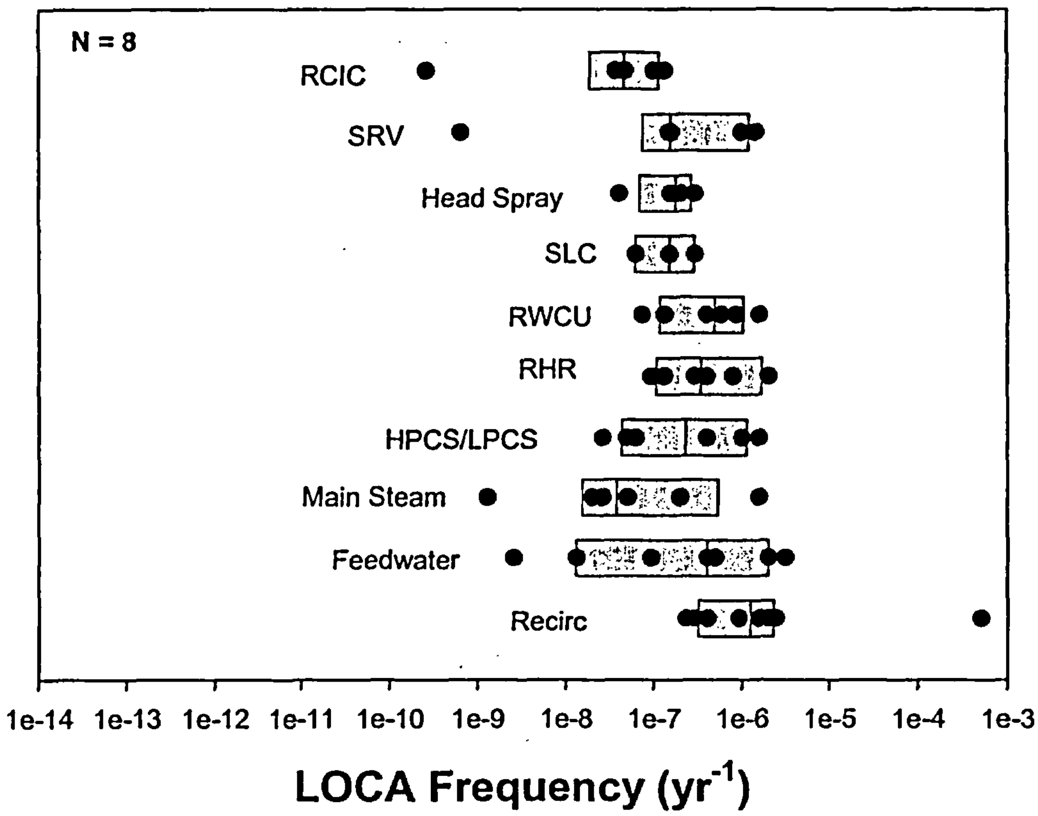


Figure L.7 Category 3 LOCA Frequencies for BWR Piping Systems at 25 Years of Plant Operation

For the largest category BWR piping LOCAs (Category 5), the recirculation system remains the main contributor to the overall LOCA frequencies, see Figure L.8. The RWCU system had about the same median value, however, there was a question expressed as to whether the RWCU system could actually sustain such a high flow rate LOCA. One of the participants thought that the maximum diameter for this system was only 6-inches, not 24-inches as specified in the development of the elicitation questions. Besides the recirculation, and RWCU systems, the next two largest contributors to the BWR Category 5 LOCA frequencies were the feedwater and RHR systems. As for the Category 3 LOCAs, the RHR system was deemed important due to the large number of precursor events reported and the large number of potentially susceptible welds. Several of the participants indicated that these lines are susceptible to stress corrosion cracking.

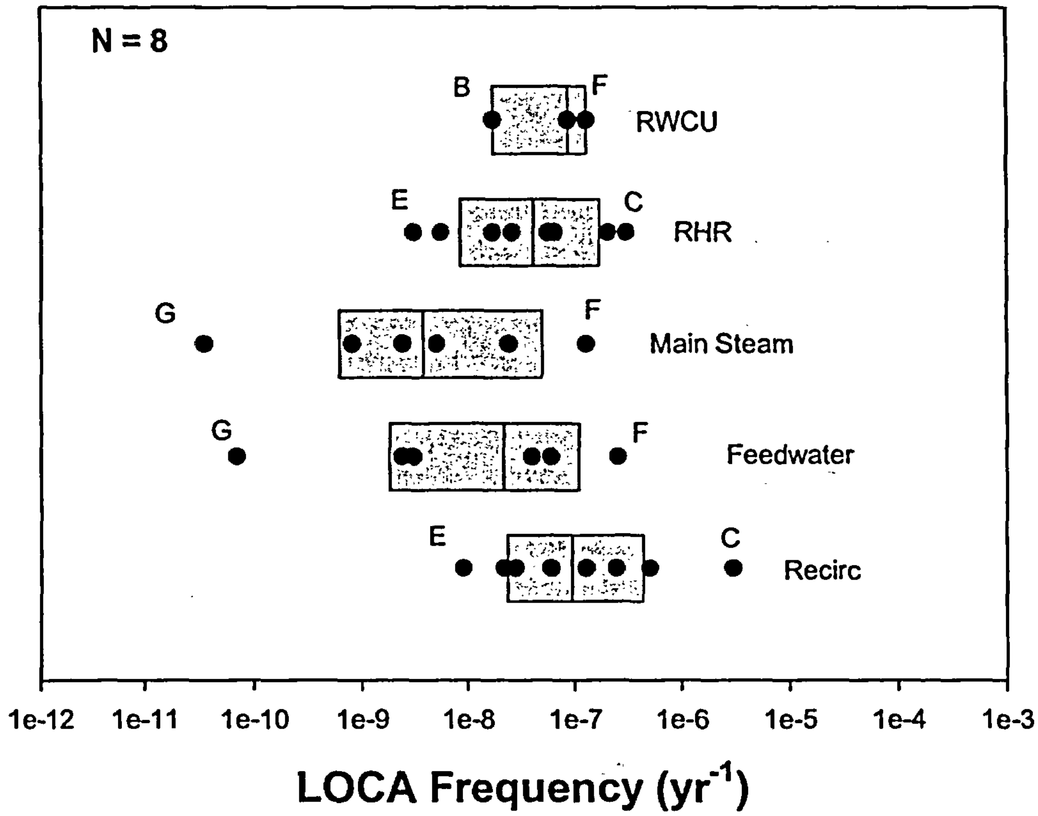


Figure L.8 Category 5 LOCA Frequencies for BWR Piping Systems at 25 Years of Plant Operation

Figure L.9 is a plot of the cumulative BWR piping LOCA frequencies (including contributions from all of the piping systems) for Category 1 through 5 LOCAs. The BWR piping LOCA frequency decrease with LOCA size is relatively shallow, i.e., approximately ½ order of magnitude per LOCA category. The results tend to be governed by the results from the recirculation system. It was noted that for the recirculation system that the mitigation programs in place for controlling IGSCC promote a more uniform residual stress field which can in turn promote longer cracks which are more likely to cause a LOCA. This effect will potentially offset the overall reduction in crack growth due to the mitigation program. It is also of note from Figure L.9 that the variability in the results as expressed by the interquartile range and the difference between the minimum and maximum values does not vary much with LOCA size. It is also of note that the expert ranking is relatively consistent with LOCA size, i.e., Participant C always predicted the highest LOCA frequencies and Participants E and G consistently predicted the lowest LOCA frequencies.

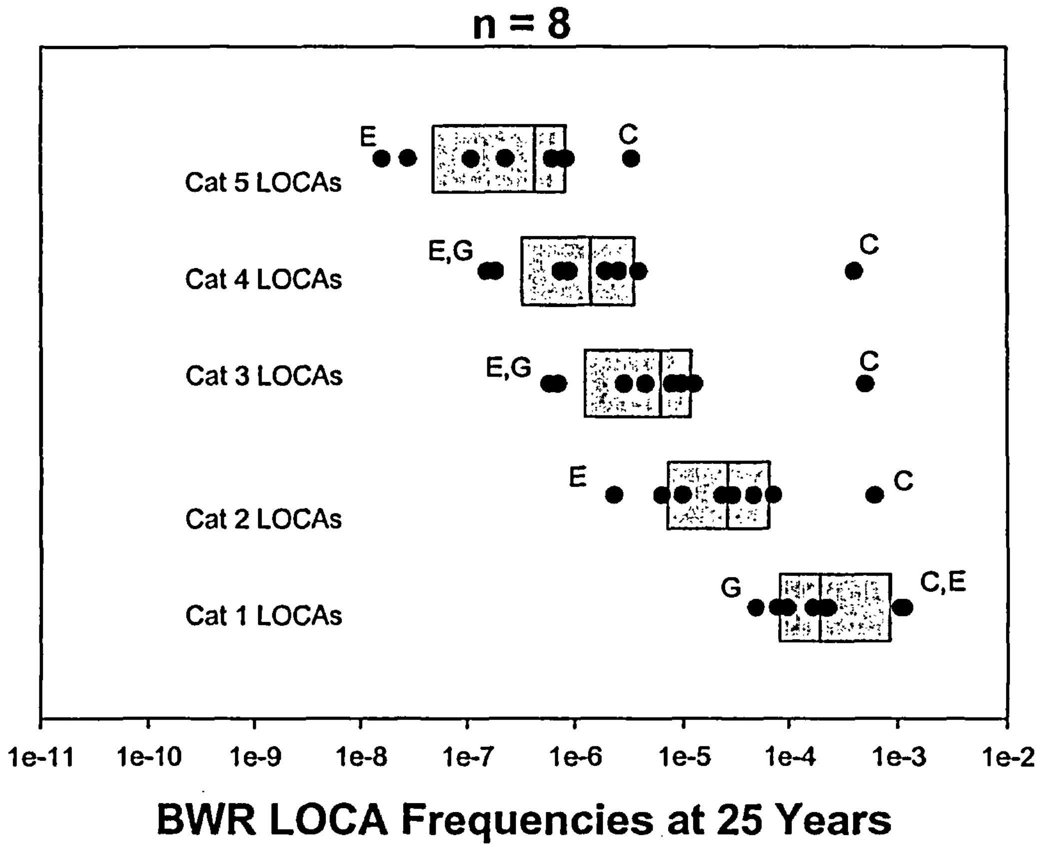


Figure L.9 Cumulative BWR LOCA Frequencies at 25 Years of Plant Operations

Figure L.10 shows the effect of operating time on the cumulative Category 1 LOCA frequencies for BWR piping systems. As can be seen in Figure L.10, there is not much of an effect of operating time on the cumulative Category 1 frequency. Similar findings were evident for the larger Category 3 and 5 LOCAs. Obviously, any unabated degradation mechanism would cause an increase in the overall LOCA frequencies. However, it was generally assumed by the panel members that any new degradation mechanism that came on the scene would be aggressively met by the industry and NRC, just like the IGSCC problem in BWRs was met in the past and the PWSCC problem in PWRs is being met today. The minimal changes in LOCA frequencies with time evident in Figure L.10 were the result of a number of compensating factors considered by the panel members. From the perspective of potential decreases in the LOCA frequencies, the recirculation lines should see a decrease in the LOCA frequencies with respect to the current day estimates that are based on an analysis of service history data due to improved mitigation strategies that have been put in place. The experts generally felt that the IGSCC issue for BWRs had been effectively mitigated for the foreseeable future. In addition, the core spray systems may see a decrease in the LOCA frequencies with time as the segments of stainless steel piping potentially susceptible to IGSCC are replaced with carbon steel piping. Finally, future inspection and mitigation programs are expected to lead to additional decreases in the predicted LOCA frequencies. In this regard, having the industry focus its inspection resources on the more important systems through risk-informed ISI should help reduce the propensity for LOCAs. Counteracting these potential decreases are potential increases due to bigger thermal fatigue and FAC concerns in the future. Concern was expressed about the high usage factors that will exist near the end of plant life. Also, there is the concern with new, previously unknown degradation mechanisms that may arise in the future. In this regard, the inspection methods of today may not be reliable for these new mechanisms. Furthermore, these new mechanisms may not manifest themselves in the same locations of concern today. Finally, while timely and proper maintenance programs are always beneficial, there are instances in which they may prove counterproductive. The frequent opening and closing of systems for inspections increases the likelihood for human error such as having tools and other debris left behind or bolts not being torqued properly. Also, improper service of active components (e.g., valves) can lead to passive system failures.

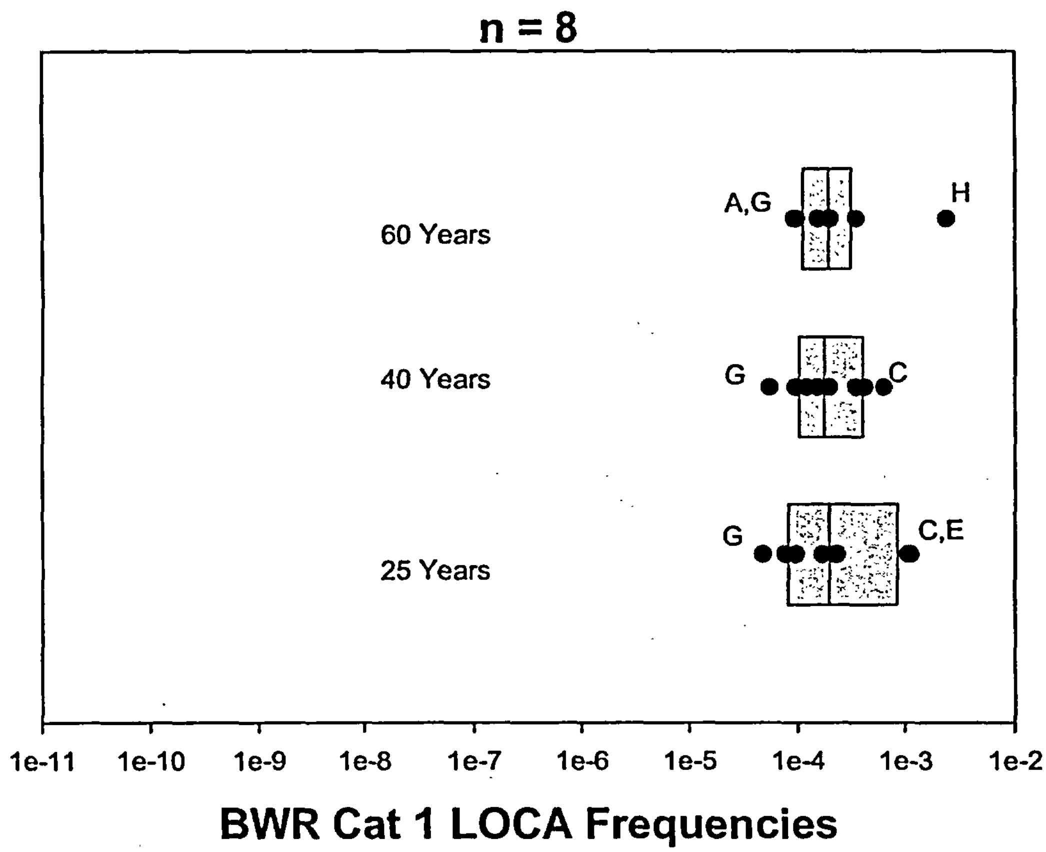


Figure L.10 Effect of Operating Time on the Cumulative Category 1 LOCA Frequencies for BWR Piping Systems

Figures L.11 and L.12 show the cumulative mid-value estimates, along with the 5% and 95% bound values for the various participants for the Category 1 and 3 LOCAs, respectively. The uncertainty range (difference between 5% lower bound and 95% upper bound values) for the Category 3 LOCAs are comparable (or slightly greater than) for the Category 1 LOCAs. Only participants A, E, and F expressed considerably more uncertainty for the Category 3 LOCAs than they did for the Category 1 LOCAs. Similar findings were found when comparing the Category 5 results with the Category 3 results. Overall, the experts appeared more confident about their BWR estimates than they did for the corresponding PWR estimates. They had less uncertainty about future and bigger size LOCA frequencies compared with their PWR predictions. There was also less uncertainty among the experts about the magnitude of the dominant contributing factors. In addition, the panel members used more consistent approaches and more consistent base case estimates for the BWR estimates than they did when making their estimates for PWRs.

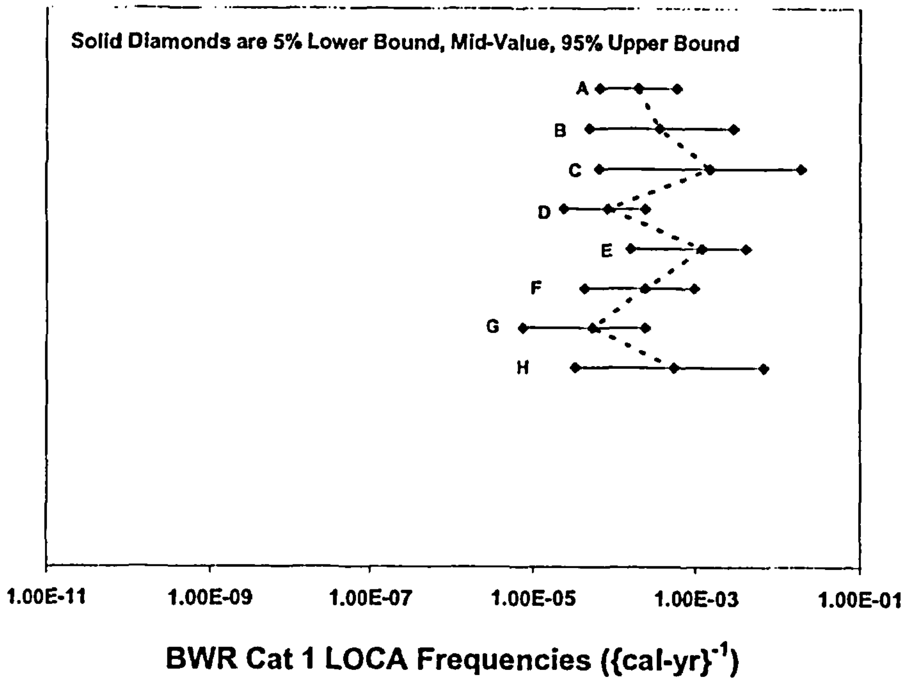


Figure L.11 BWR Category 1 LOCA Frequencies Showing Mid-Values, 5% Lower Bound, and 95% Upper Bound Values for All Participants Who Responded to the BWR Piping Questions

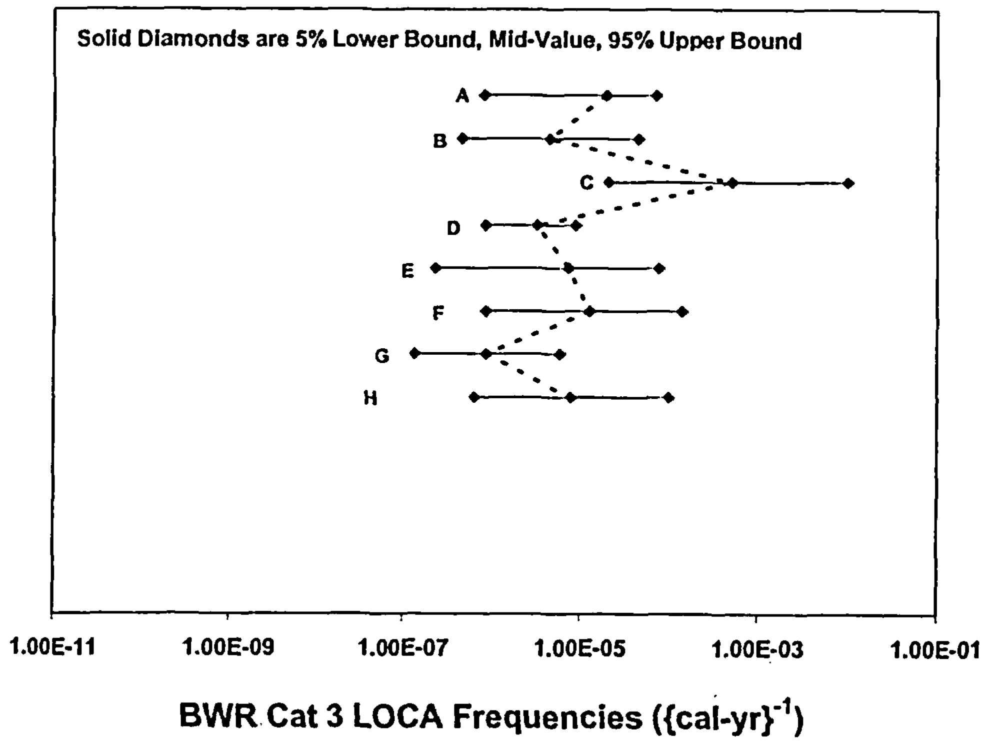


Figure L.12 BWR Category 3 LOCA Frequencies Showing Mid-Values, 5% Lower Bound, and 95% Upper Bound Values for All Participants Who Responded to the BWR Piping Questions

L.3 PWR Piping

The primary aging mechanisms identified by the participants for PWR plants are thermal fatigue, primary water stress corrosion cracking (PWSCC), and mechanical fatigue. The concerns associated with thermal and mechanical fatigue in PWR plants are similar to those in BWR plants. In PWRs, the surge line is thought to be susceptible to thermal fatigue due to cyclic thermal stratification stresses. Also, the direct volume injection (DVI) and chemical volume and control system (CVCS) lines were thought to be susceptible due to periodic testing which imposes additional thermal cycles. In addition, the DVI and Accumulator Safety Injection System (SIS) lines have experienced some thermal fatigue cracking issues due to leakage of cold water past the check valves in these systems.

PWSCC is a relatively new mechanism that has manifested itself in this country over the last 5 years. It has many similar characteristics to the IGSCC problem experienced in BWR reactors in the past. It is a temperature dependent mechanism that attacks Alloy 600 type materials such as bimetallic Inconel 82/182 welds. Many panel members believe that PWSCC problems will be resolved (i.e., mitigated) over the next 15 years. Therefore, its contribution to the overall LOCA frequencies may peak between the 25 and 40 year time period, but then decrease in the future. Today, instances of PWSCC have been observed in surge lines at the surge line to pressurizer weld in the United States at Three Mile Island, in Belgium and Japan, and in hot legs at the hot leg to reactor pressure vessel weld in the United States at V.C. Summer and Sweden. Other lines in which PWSCC may become an issue in the future are the cold leg and the pressurizer spray lines. However, since the cold leg operates at lower temperatures than the hot legs and surge lines, any problems that may materialize in the cold leg will be delayed until later in the operating life.

As part of this elicitation a total of 12 LOCA sensitive piping systems were considered for the PWR plants. However, as was the case for BWR piping, most of the participants focused on a few common systems as being important LOCA contributors. Figure L.13 shows the Category 1 LOCA frequencies at 25 years for the 12 PWR LOCA sensitive piping systems. As was the case for the BWR piping systems, the Category 1 LOCA frequencies are dominated by the small diameter instrument and drain lines. The estimated Category 1 LOCA frequencies for these lines are two orders of magnitude higher than the hot leg and surge lines. This again reflects the belief that a complete break of a smaller pipe is more likely than a partial break of comparable size of a larger pipe. The reasoning provided by the panel members for why these small diameter lines are so susceptible to these smaller category LOCAs are the same as provided for the case of BWRs. These 1- and 2-inch socket welded lines are susceptible to vibration fatigue concerns. Also, they are more difficult to inspect, if they are inspected at all, than their larger diameter counterparts. Generally speaking, the benefits of leak detection and ISI decrease with decreasing pipe size. Finally, they are more susceptible to other, non-aging related failure mechanisms, e.g., an externally applied overload.

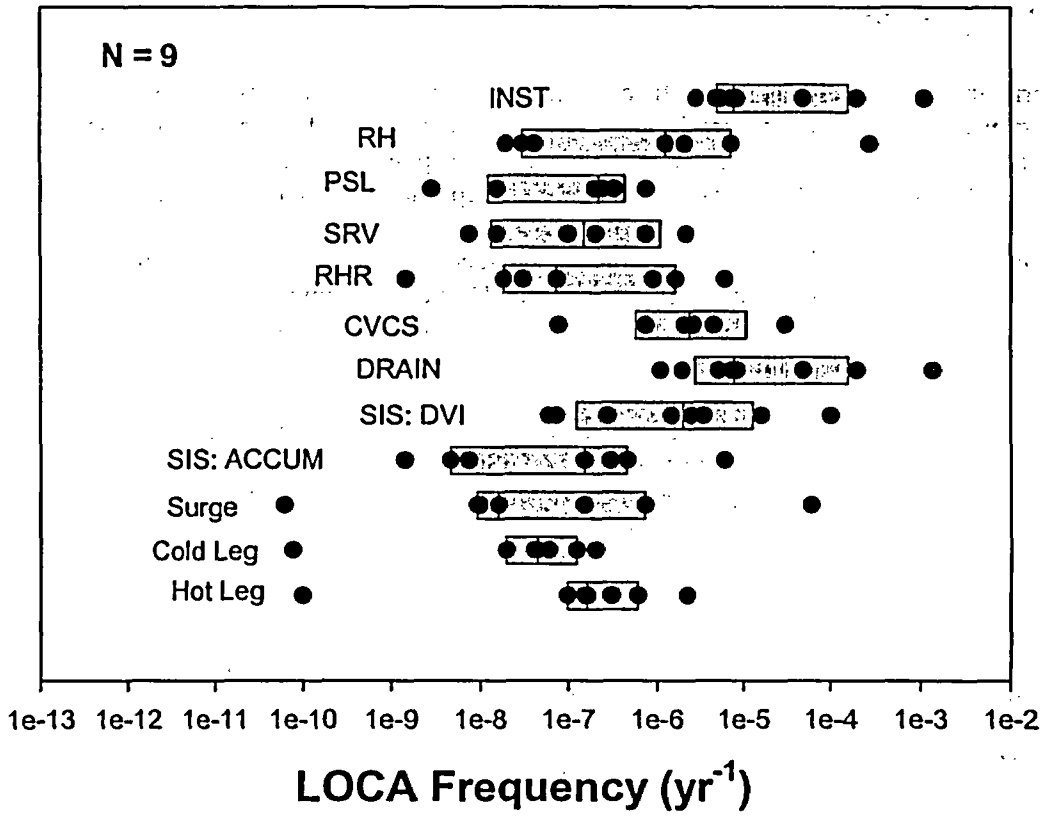


Figure L.13 Category 1 LOCA Frequencies for PWR Piping Systems at 25 Years of Plant Operation

Figure L.14 shows the breakdown of PWR Category 3 LOCA frequencies by piping system at 25 years of plant operations (present day). The small diameter instrument and drain lines, as well as the reactor head (RH) lines, do not appear on this figure in that they are of such size that they could not sustain a Category 3 LOCA. Again, as was the case for the PWR Category 1 LOCAs, the smallest diameter lines that can sustain this size (i.e., category) of LOCA are the dominant contributors. These include the CVCS, SIS-DVI, RHR, surge, and pressurizer spray lines (PSL). This is different than what was observed for the BWR Category 3 LOCAs where the larger recirculation system was the dominant contributor, primarily due to its susceptibility to IGSCC. The two most listed systems as being major contributors to this category of LOCA for PWR piping were the CVCS and SIS-DVI lines. For both, the primary concern was fatigue. One participant commented that the CVCS line was one of the most fatigue sensitive locations in the entire plant. Another commented that they were concerned with environmentally-assisted fatigue for this system. With regard to the SIS-DVI (and the SIS-Accumulator lines for that matter), several participants indicated that both lines had experienced thermal fatigue cracking in the past due to cold water leaking past the check valves. Another line that a number of participants thought would be a major contributor to this category of LOCA was the RHR lines. The concern with these lines was with environmental attack due to the stagnant nature of the flow in these lines. The pressurizer spray lines were of a concern due to the chance for PWSCC at one of the bimetal welds.

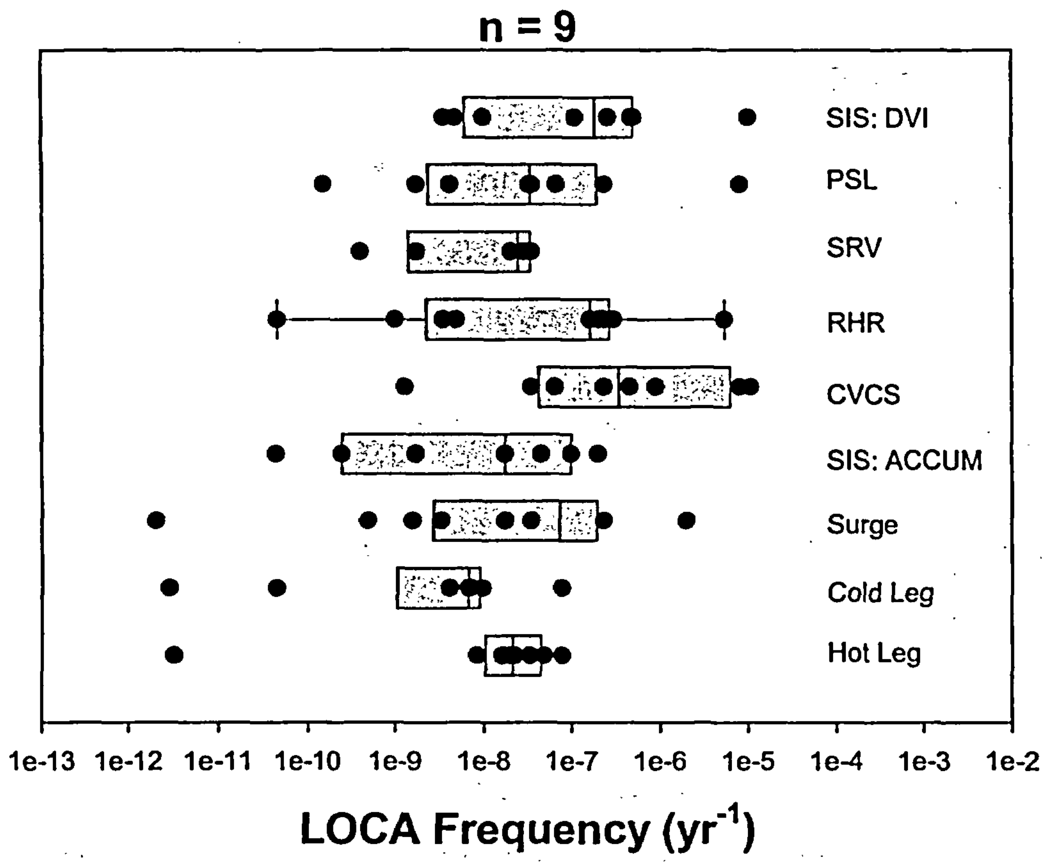


Figure L.14 Category 3 LOCA Frequencies for PWR Piping Systems at 25 Years of Plant Operation

For the largest categories of PWR piping LOCAs (Categories 5 and 6), the hot leg, cold leg, surge line, and RHR lines all contribute to the overall LOCA frequencies, see Figure L.15 for the Category 5 LOCAs. Of these, the median value of the LOCA frequency for the cold leg is about a half order of magnitude less than the median values for the other three piping systems. This slight reduction is primarily due to the fact that the cold leg is less susceptible to PWSCC than either the hot leg or surge line at this time (25 years of plant operations) due to the fact that it operates at a slightly lower temperature. Somewhat surprisingly in examining Figure L.15, a number of the participants felt that the hot leg would have a greater propensity for a Category 5 LOCA than the surge line. Both lines are susceptible to PWSCC due to the presence of bimetallic welds and the high operating temperatures, but the surge line was also judged to be susceptible to thermal fatigue due to thermal stratification and thermal striping stresses. Also, the surge line is smaller diameter, which based on the thought that smaller diameter lines are more prone to LOCAs than their larger counterparts, would imply that the Category 5 LOCA frequencies for the surge line should be higher. Finally, at least one participant argued that the surge line to pressurizer bimetallic weld was one of their biggest concerns in the entire plant due to its susceptibility to PWSCC and the fact that it is a very difficult weld to inspect. Counteracting these arguments, however, is the fact raised by a number of the participants that there are more hot leg to reactor pressure vessel (RPV) bimetal welds (2 to 4 depending on the number of loops) in a plant than there are surge line to pressurizer bimetal welds (one).

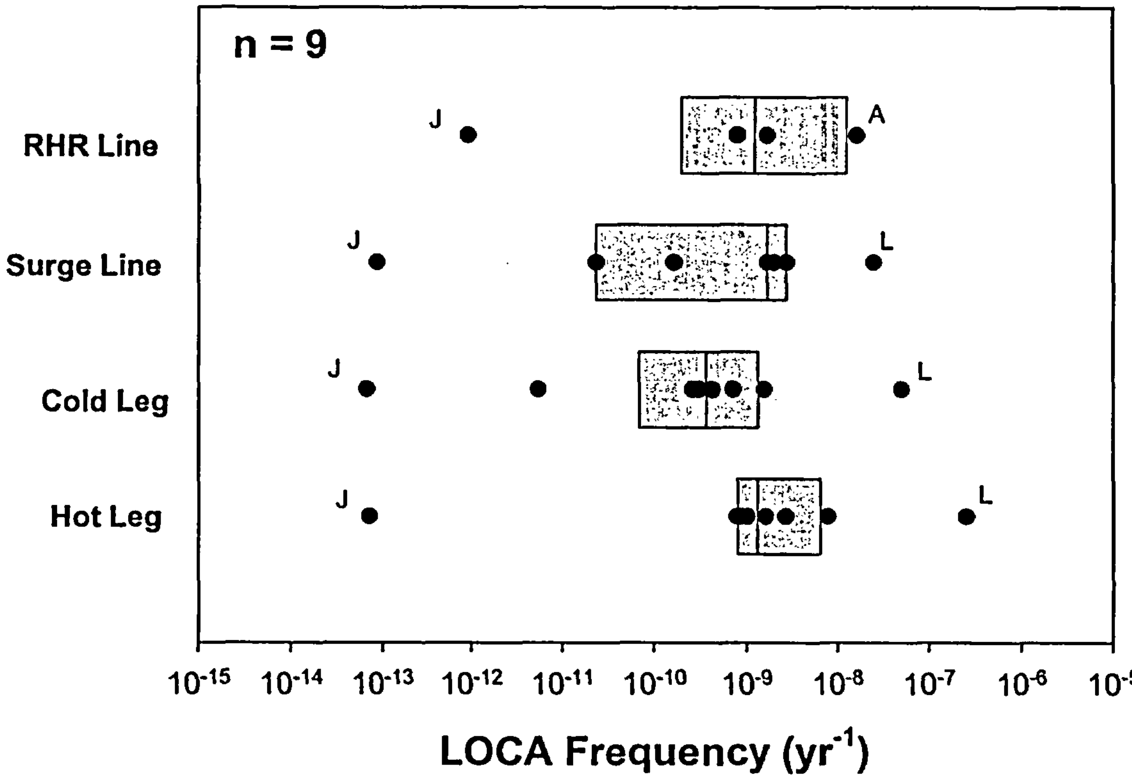


Figure L.15 Category 5 LOCA Frequencies for PWR Piping Systems at 25 Years of Plant Operation

Figure L.16 is a plot of the cumulative PWR LOCA frequencies at 25 years of plant operation. Cumulative frequencies are shown for Category 1, 3, and 6 LOCAs. Based on a review of Figure L.16 there appears to be approximately a one order of magnitude reduction in LOCA frequency between each successive LOCA category.

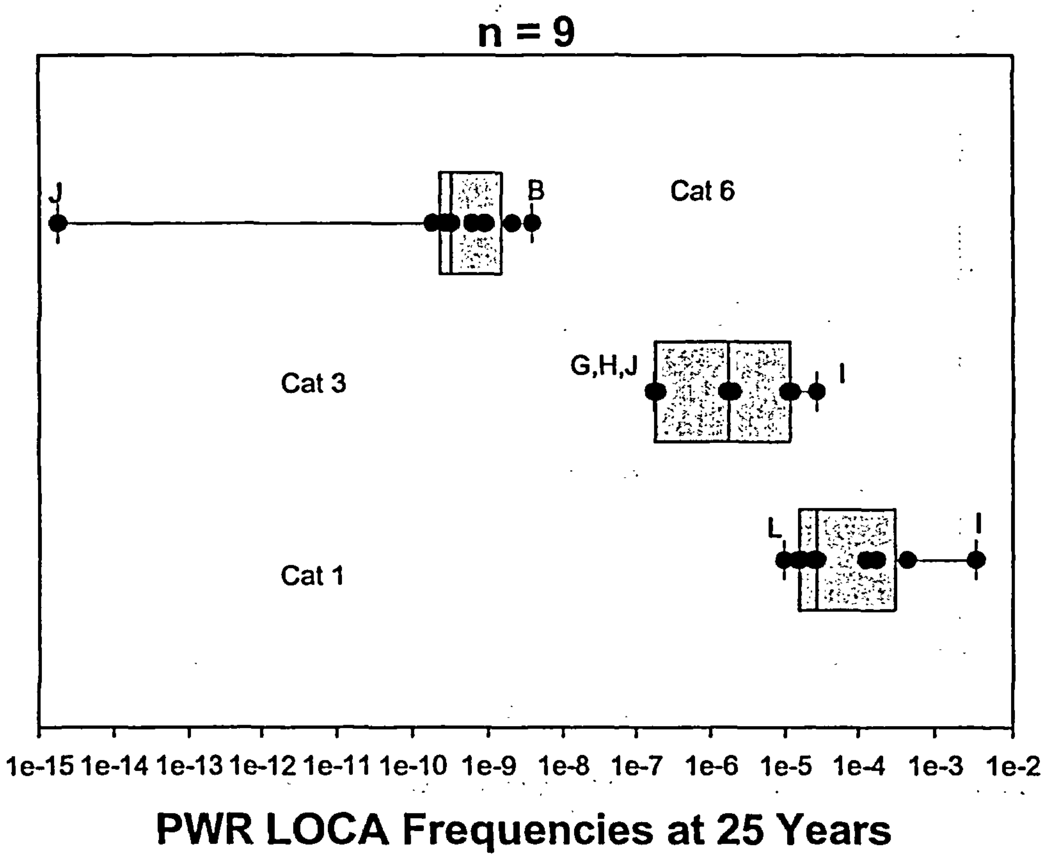


Figure L.16 Cumulative PWR LOCA Frequencies at 25 Years of Plant Operations

Figure L.17 shows the effect of operating time on the cumulative Category 1 LOCA frequencies for PWR piping systems. Several participants felt that the service experience is sufficient to expect the frequencies to remain relatively constant out to 60 years of life. Degradation and aging will naturally continue to occur. However, the inspection and mitigation strategies will effectively identify and temper the frequency increases caused by this aging. Some experts expected a short term frequency increase due to PWSCC before effective mitigation is developed. This trend is consistent with the historical response to evidence of emerging degradation by the industry. Also, at least one participant expressed a concern about the high usage factors that will exist at 60 years at many locations. All of these concerns are reflected in the results showing the effects of operating time and aging in Figures L.17 and L.18 for Category 1 and 3 LOCAs, respectively. As can be seen in Figures L.17 and L.18, there is a slight increase in the cumulative Category 1 and 3 LOCA frequencies between 25 and 40 years, but not much of an effect between 40 and 60 years. The median LOCA frequencies for the Category 1 and 3 LOCAs at 40 years are an order of magnitude higher than the median LOCA frequencies for the Category 1 and 3 LOCAs at 25 years. Similar findings were evident for the larger Category 6 LOCAs. The rationale behind this is that this size of LOCA (and associated pipe size) is most affected by aging. These pipes are not as easily inspected, or as leak sensitive, as their larger counterparts and these pipes have not experienced the infant mortality as their smaller counterparts.

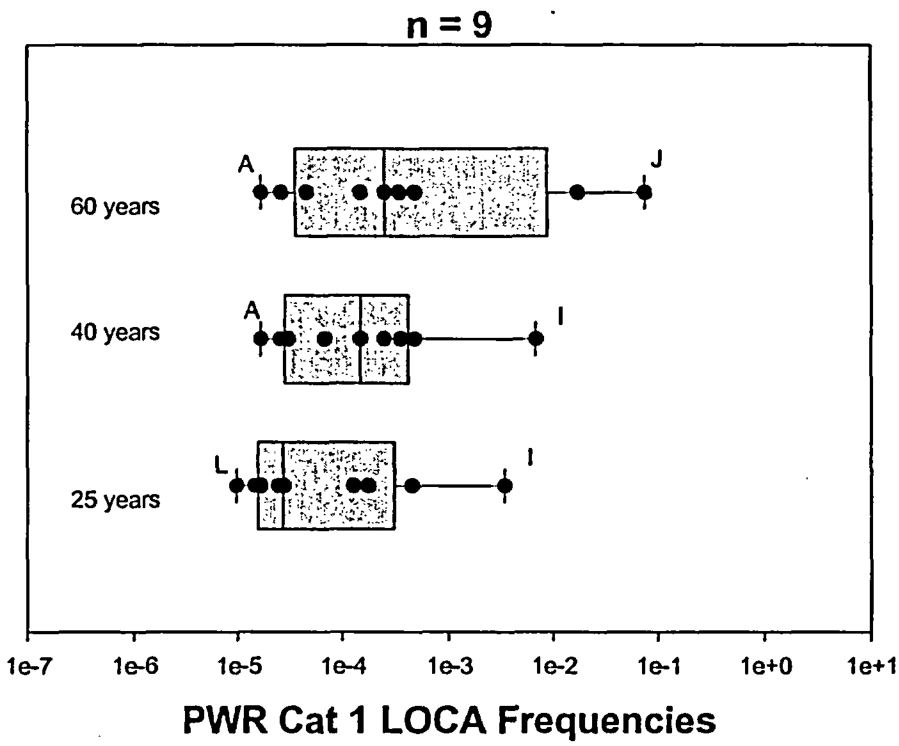


Figure L.17 Effect of Operating Time on the Cumulative Category 1 LOCA Frequencies for PWR Piping Systems

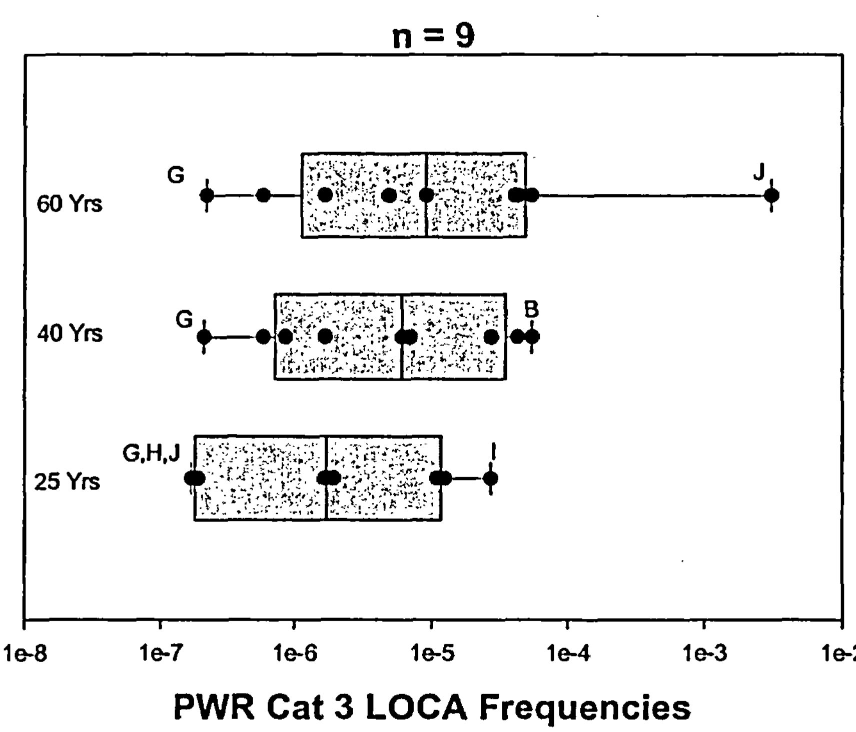


Figure L.18 Effect of Operating Time on the Cumulative Category 3 LOCA Frequencies for PWR Piping Systems

Figures L.19 and L.20 show the cumulative mid-value estimates, along with the 5% and 95% bound values for the various participants for the Category 1 and 3 LOCAs, respectively. The uncertainty range (difference between 5% lower bound and 95% upper bound values) for the Category 3 LOCAs are typically greater than for the Category 1 LOCAs for most of the participants. In a similar vein, the level of uncertainty for the Category 6 estimates were much greater than for the Category 1 or 3 estimates, see Figure L.21. All of the experts had at least two orders of magnitude difference between the lower and upper bound values for their Category 6 estimates, and some of the experts (C, E, and J) had greater than four orders of magnitude difference.

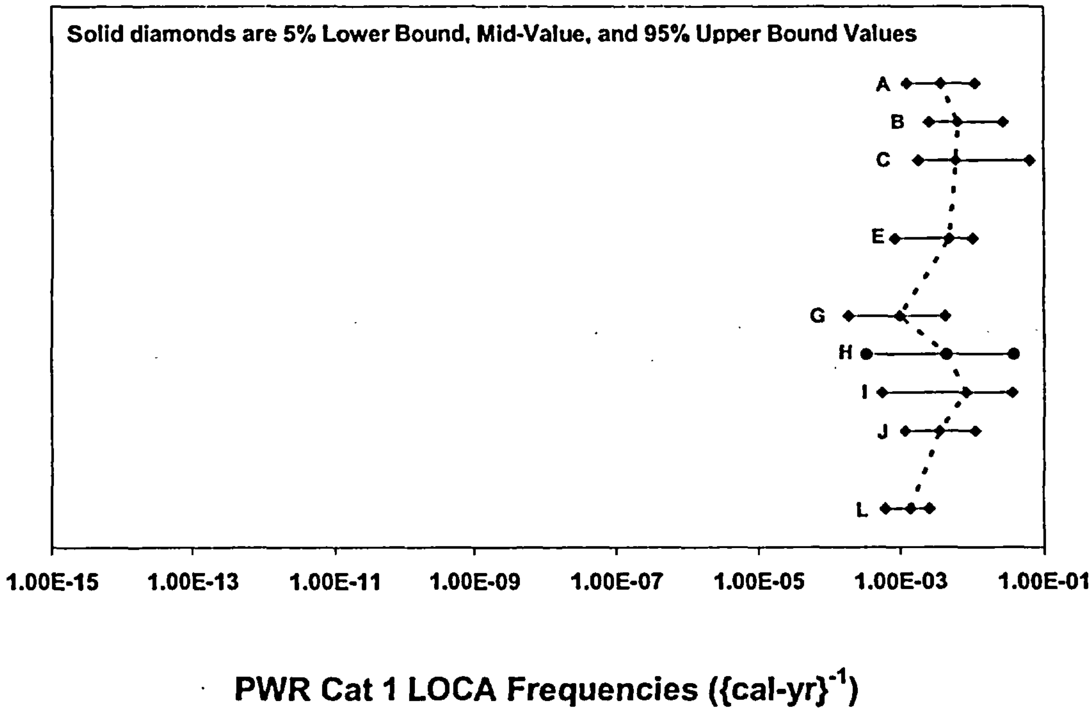


Figure L.19 PWR Category 1 LOCA Frequencies Showing Mid-Values, 5% Lower Bound, and 95% Upper Bound Values for All Participants Who Responded to the PWR Piping Questions

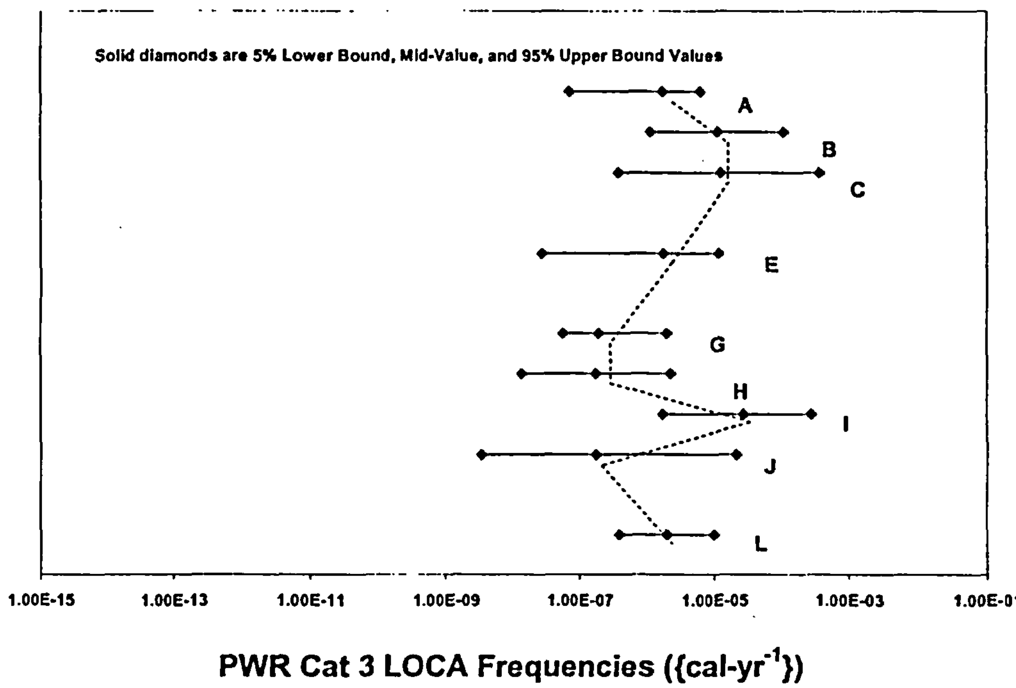


Figure L.20 PWR Category 3 LOCA Frequencies Showing Mid-Values, 5% Lower Bound, and 95% Upper Bound Values for All Participants Who Responded to the PWR Piping Questions

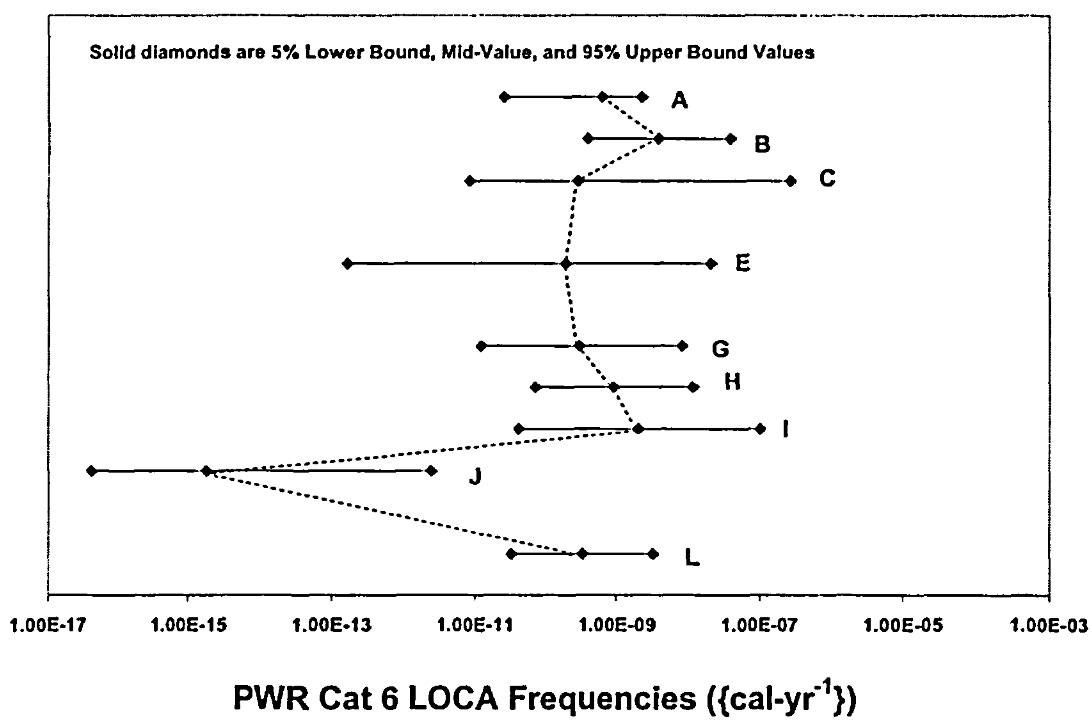


Figure L.21 PWR Category 6 LOCA Frequencies Showing Mid-Values, 5% Lower Bound, and 95% Upper Bound Values for All Participants Who Responded to the PWR Piping Questions

In general, the results for PWR piping appear consistent. The quantitative results and the qualitative opinions and rationale were for the most part consistent. The variability between participants stems from the different approaches used and the basis for their estimates. Several different approaches with different anchoring points were used by the different experts. The variability between the participants seems reasonable given the frequency magnitudes being computed.

L.4 BWR Non-Piping

Generally speaking making estimates of LOCA frequencies for non-piping components is more challenging than making estimates for piping systems. There are multiple components to consider, each having different operating requirements, design margins, materials, and inspectability. There are also widely varying failure modes and scales to consider. For PWRs for the smaller category LOCAs, one must consider steam generator tube ruptures and small penetration failures. For the larger category LOCAs, common cause bolting failures and component shell failures need to be considered. For the larger components, the bigger design margins (compared to those for piping) are somewhat offset by the decreased inspection quantity and quality. Compounding all of this is the fact that there is generally not as much precursor information available for the non-piping components as there is for piping.

For the BWR plants, the three major non-piping components that were considered were the reactor pressure vessel, the pumps, and the valves. In general, many of the same degradation mechanisms that are of concern for BWR piping are also a concern for BWR non-piping components. Stress corrosion cracking (specifically PWSCC) is a concern for many of the smaller Alloy 600 components, such as the control rod drive mechanisms (CRDMs) and other penetrations. As with piping, multiple cracks and fast propagation rates could lead to LOCAs. While the mechanism (PWSCC) is more severe at higher temperatures associated with PWRs, this mechanism could become a more significant issue later in the life of the BWRs. Thermal fatigue is another degradation mechanism associated with BWR non-piping components that is common with BWR piping. Thermal fatigue is especially of concern at inlet nozzles and other locations that experience thermal stratification, especially at the feedwater nozzles. For the same reasons as highlighted above for BWR piping, thermal fatigue can possible lead to larger leaks or LOCAs.

Other mechanisms for non-piping components that were not of concern for BWR piping are radiation embrittlement, common cause bolting failures, and thermal aging of cast stainless steel components, such as pump and valve casings. Radiation embrittlement reduces the base metal toughness of the RPV. Fortunately, for BWRs, it is not as of much concern as it is for PWRs due to the increased shielding available with the BWRs. Common cause bolting failures are important for manways and bolted valves. The common cause mechanisms may possibly include: improper installation or maintenance of the bolts, i.e., improper torque, external corrosion of multiple bolts, and steam cutting of bolts. One participant thought that these common cause failures will cause the greatest risk. Thermal aging of cast stainless steels can cause a significant reduction in the fracture toughness of these materials, however, fortunately to date no cracking mechanisms have emerged for these materials.

Figure L.22 shows the Category 1 LOCA frequencies for the RPV, pumps, and valves at 25 years. The RPV shows the biggest expected Category 1 LOCA frequencies. The Category 1 RPV LOCA frequencies are driven by the CRDM penetration failures. However, the severity of the

CRDM failures for BWRs was reduced by about one order of magnitude with respect to PWR CRDM failures due to the BWR heads operating at a lower temperature. Other than these head penetrations, nozzle and body cracking were mentioned as possible sources of failures. A number of precursor cracking incidents have been seen in service. The valves and pumps contribute to a lesser extent. Most of the experts generally treated these components the same. At least one expert (F) had some experience with manufacturing defects in valve bodies which led to some increased concern with valves. Other issues with valves, and pumps, included the potential for bolt failures for the reasons outlined above, and the fact that the material they are made of (cast stainless steel) is notoriously difficult to inspect and is subject to toughness degradation due to thermal aging.

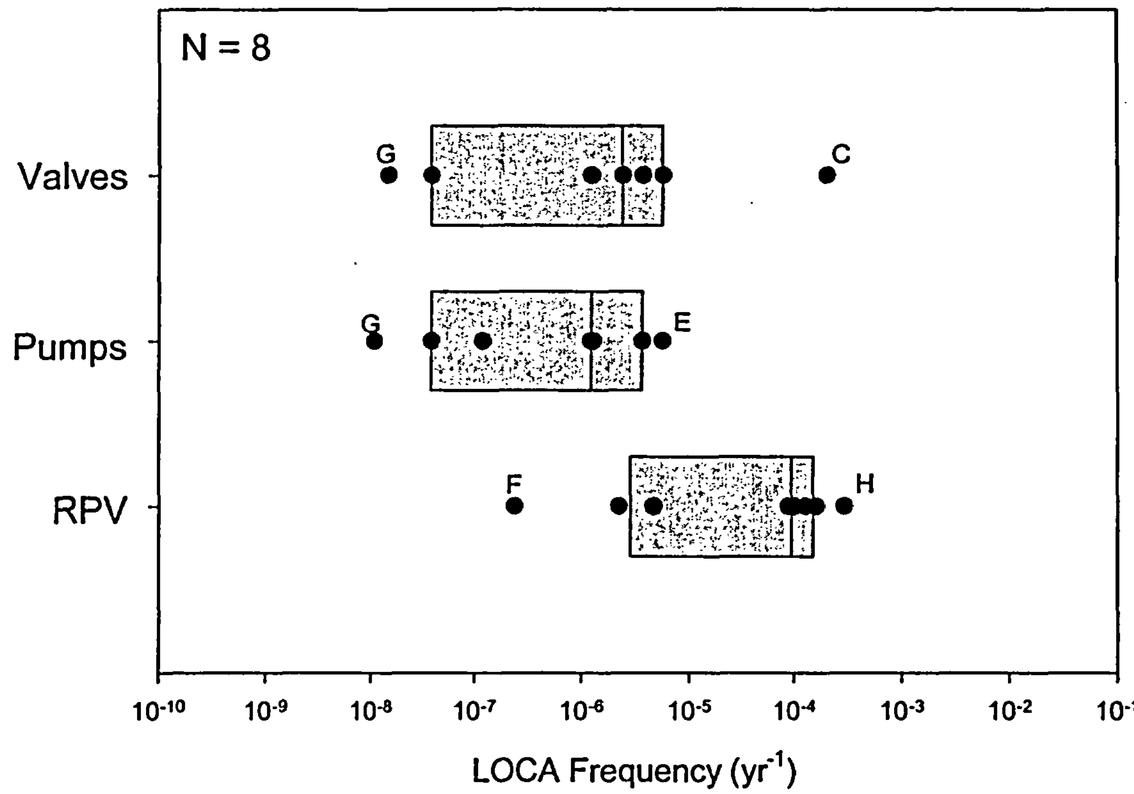


Figure L.22 BWR Category 1 Non-Piping LOCA Frequencies by Major Component at 25 Years of Plant Operations

Figure L.23 shows the Category 3 non-piping LOCA frequencies at 25 years of plant operations. The most noticeable difference between the Category 3 and Category 1 LOCA frequencies is the three orders of magnitude reduction in the median value of the estimated LOCA frequencies for the RPV as the CRDM concerns disappear. A single CRDM failure cannot support a Category 3 LOCA. Only about half of the participants were concerned about RPV nozzle failures, but those that were assigned comparatively high frequencies to them. For the pumps and valves, the corresponding decrease in LOCA frequency is only about one order of magnitude. Consequently, the pumps, valves, and vessel now contribute about equally to the overall LOCA frequency.

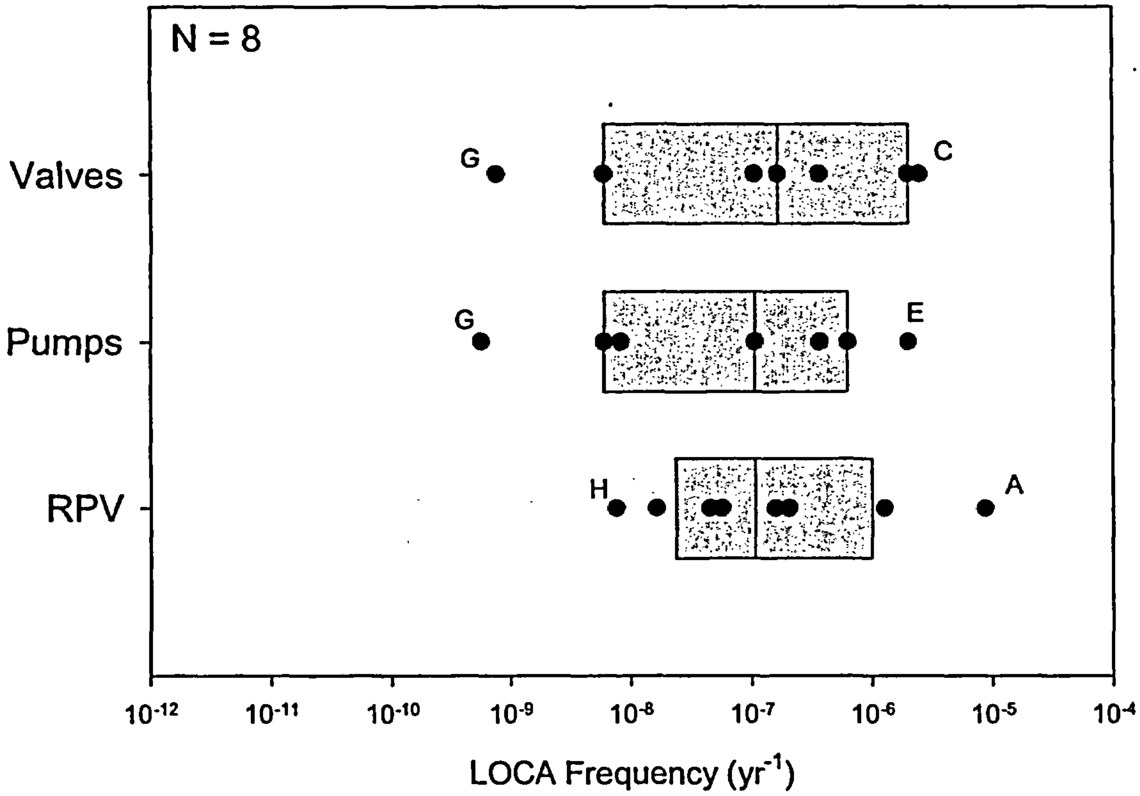


Figure L.23 BWR Category 3 Non-Piping LOCA Frequencies by Major Component at 25 Years of Plant Operations

Figure L.24 shows the Category 5 non-piping LOCA frequencies at 25 years for the BWR non-piping components. As was the case for the Category 3 LOCAs, the pumps, valves, and vessel are all now of about equal importance. For these large LOCAs, the experts felt that the valve, pump, and vessel bodies were the most likely subcomponents to fail. For the vessel body, the concern was with low temperature overpressure (LTOP) while for the valve and pump bodies, the concern was with fatigue and stress corrosion cracking.

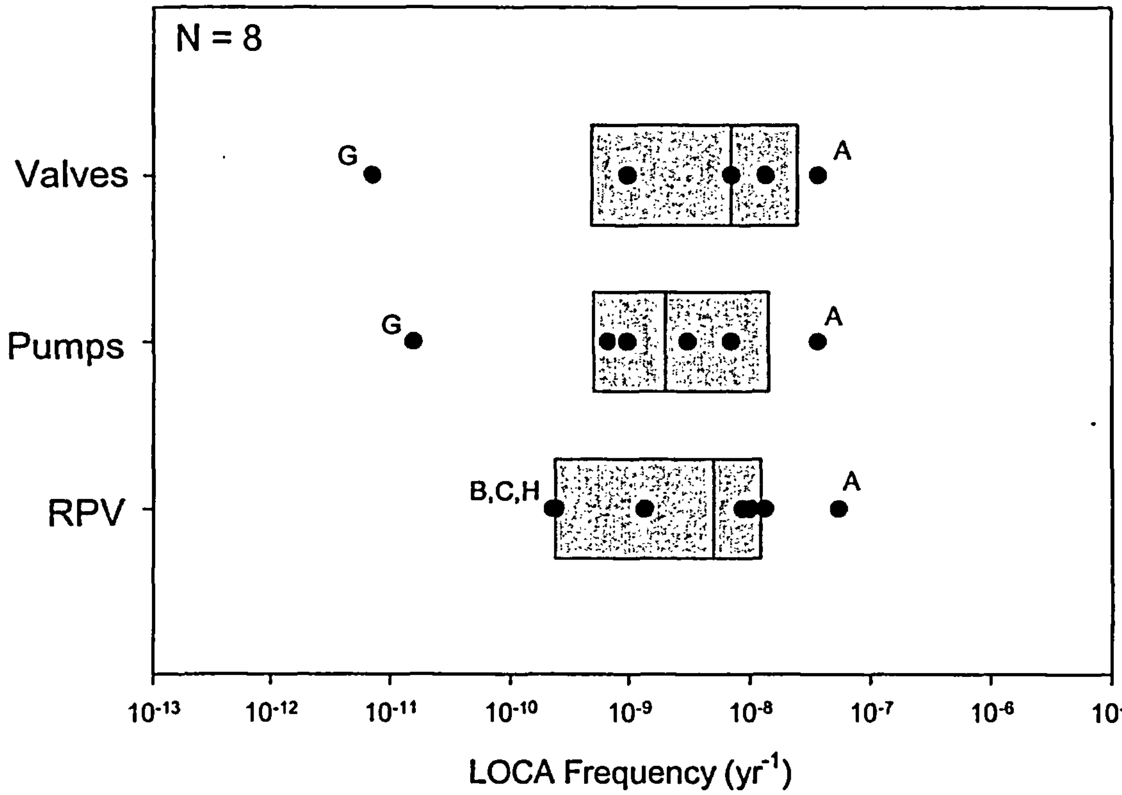


Figure L.24 BWR Category 5 Non-Piping LOCA Frequencies by Major Component at 25 Years of Plant Operations

Figure L.25 shows the cumulative LOCA frequencies for the BWR non-piping components at 25 years of plant operations. On average there is about a one order of magnitude shift in the cumulative LOCA frequency between each successive LOCA category. The median value for the estimate of the Category 1 LOCA frequency is approximately 10^{-4} while the median value for the Category 6 LOCA frequency is about 10^{-9}.

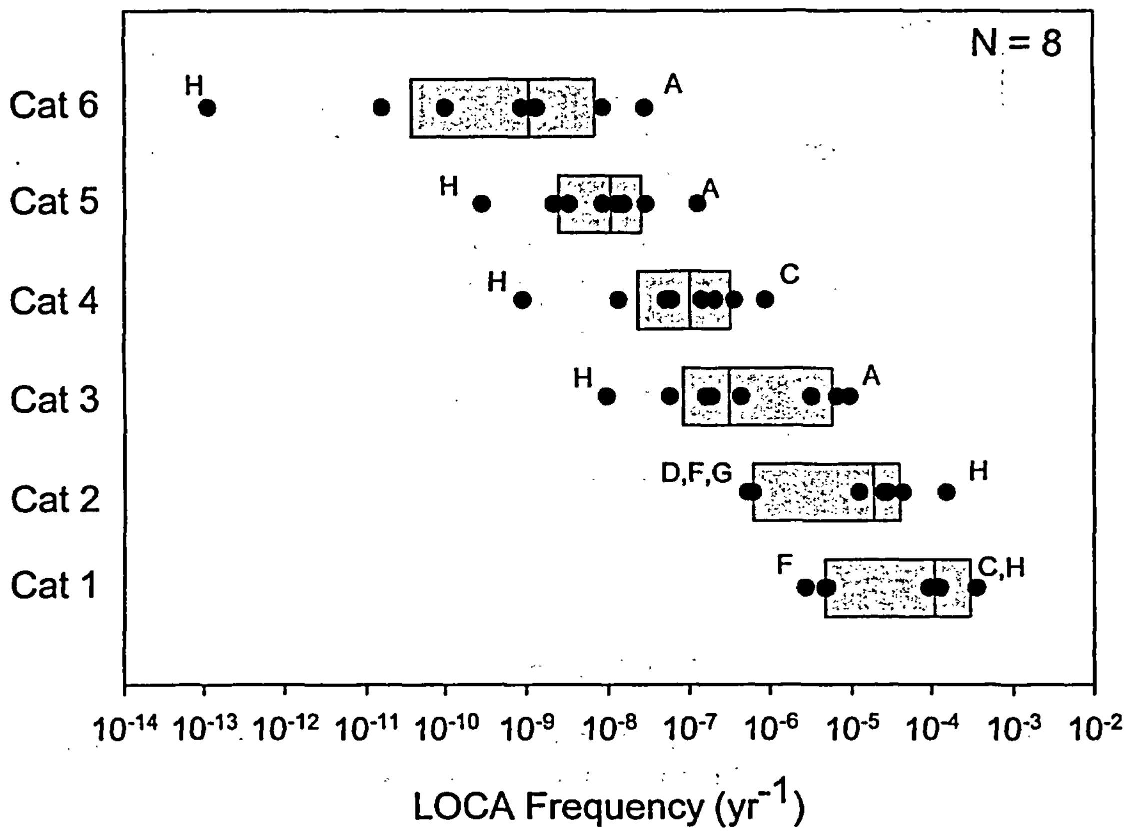


Figure L.25 Cumulative BWR Non-Piping LOCA Frequencies at 25 Years of Plant Operations

Figures L.26 and L.27 show the effect of time on the Category 1 and 3 cumulative LOCA frequencies, respectively, for the BWR non-piping components. For all intents and purposes there is almost no effect of time on the predicted LOCA frequencies. The median values do not change nor does the variability, i.e., the interquartile ranges remain the same. Non-piping components are affected by similar partially compensating factors as the piping components. In addition, a number of participants expressed the belief that the maintenance and mitigation issues raised for piping also apply for non-piping components. The only thing that changes is the minimum value predicted by Participant H for LOCA Category 3. Participant H foresees the non-piping LOCA frequencies increasing at both the 40 and 60 year time interval. Figure L.28 shows the effect of time on the Category 5 frequencies. In this case the median values do not vary with time, nor does the maximum values, however, a number of participants started to see the LOCA frequencies increasing near the end of plant life extension (60 years) such that the lower end of the IQR (i.e., the 25th percentile) increased an order of magnitude at 60 years over what it was at 40 and 25 years. This increase in the Category 5 predictions was driven by aging concerns of a few of the experts at 60 years.

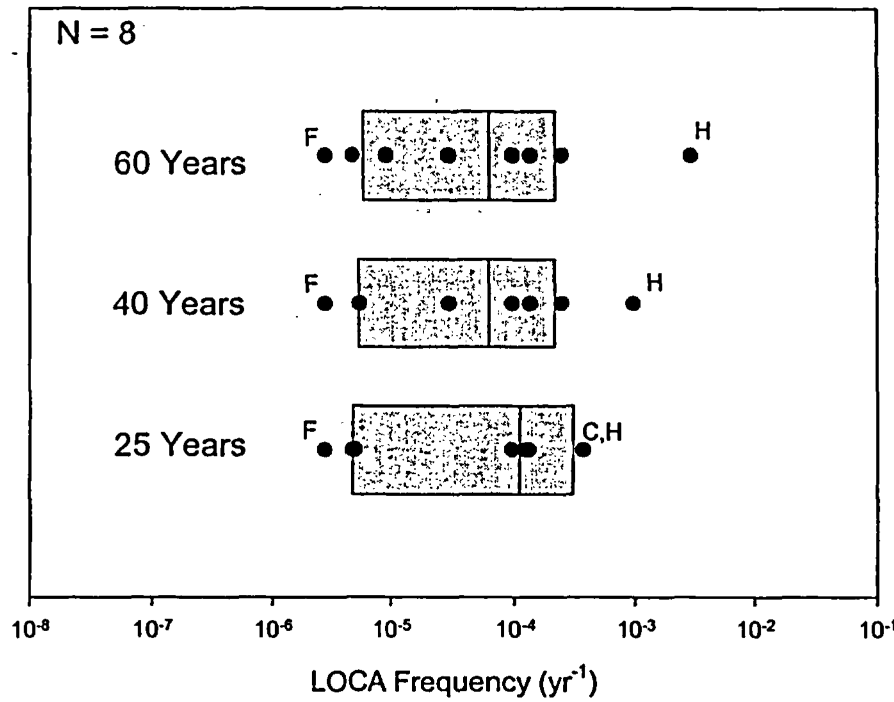


Figure L.26 Effect of Operating Time on the Cumulative Category 1 LOCA Frequencies for BWR Non-Piping Components

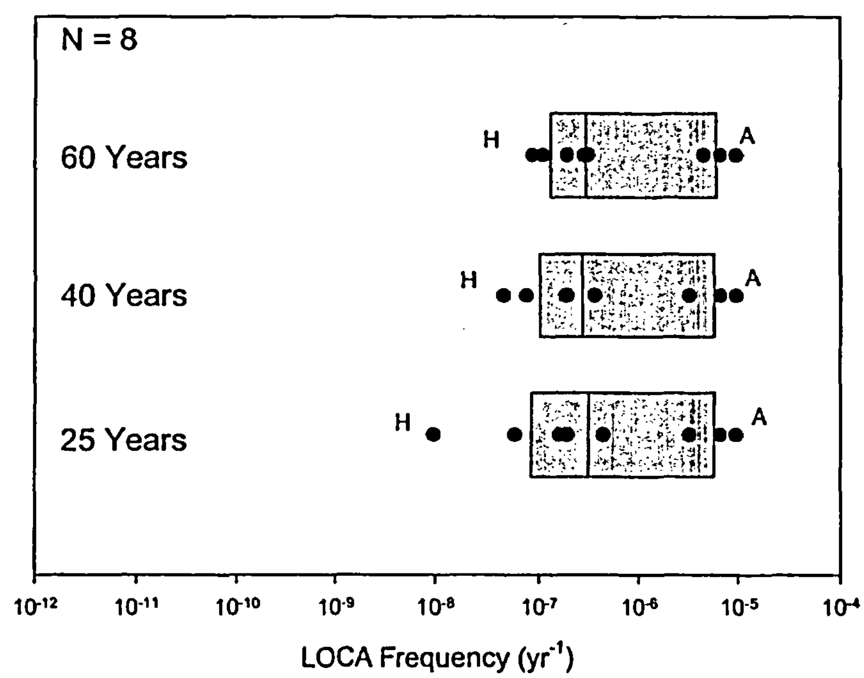


Figure L.27 Effect of Operating Time on the Cumulative Category 3 LOCA Frequencies for BWR Non-Piping Components

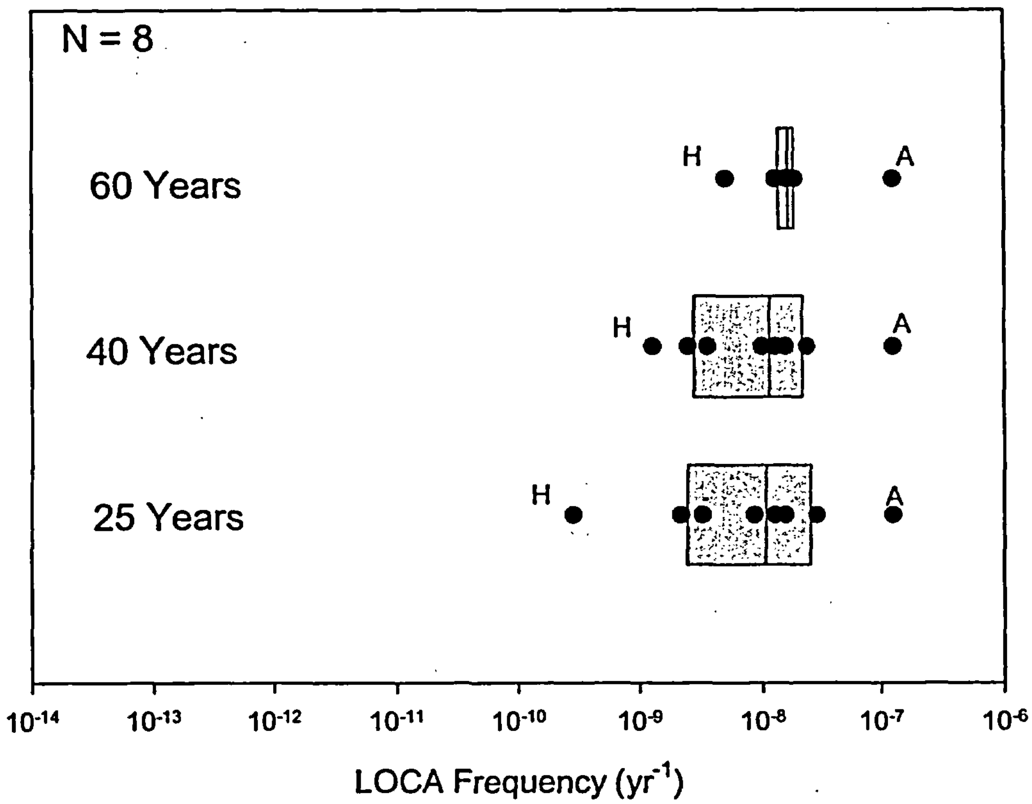


Figure L.28 Effect of Operating Time on the Cumulative Category 5 LOCA Frequencies for BWR Non-Piping Components

Figures L.29 and L.30 show the cumulative mid-value estimates, along with the 5% and 95% bound values for the various participants for the BWR Category 1 and 3 non-piping LOCA frequency estimates, respectively, at 25 years of plant operating time. Of note from these figures is the fact that a number of the participants (A, E, F, and H) predicted greater uncertainty for the Category 3 LOCAs than they did for the Category 1 LOCAs. This is not unusual in that one would expect the uncertainty to increase for lower frequency events, such as larger LOCAs. It is probably somewhat more surprising that the other four participants predicted comparable uncertainty for the Category 1 and 3 LOCAs. Overall the predictions for BWR non-piping were more consistent than for PWR non-piping discussed next. For the BWRs, there are less components and failure modes to consider and the approaches used to estimate the frequencies were more closely related.

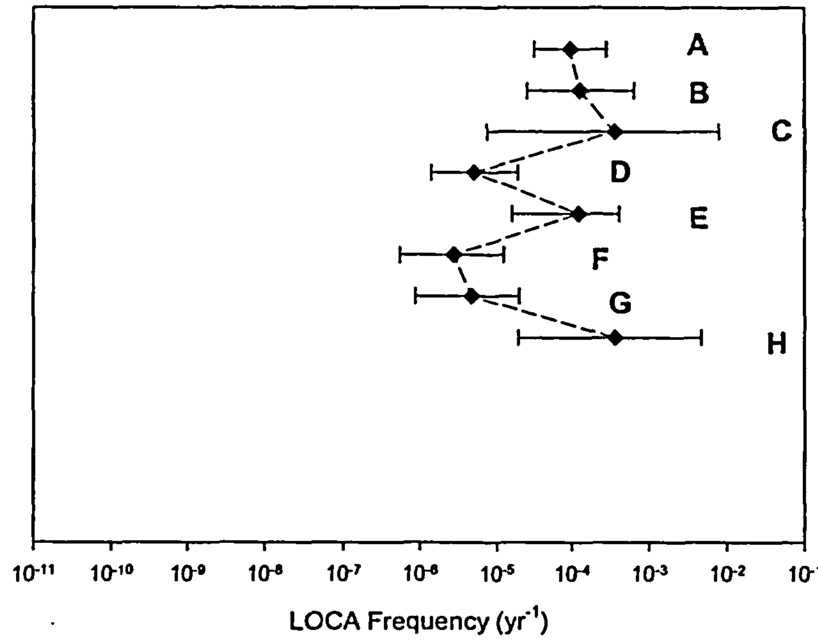


Figure L.29 BWR Non-Piping Category 1 LOCA Frequencies Showing Mid-Values, 5% Lower Bound, and 95% Upper Bound Values for All Participants Who Responded to the BWR Non-Piping Questions

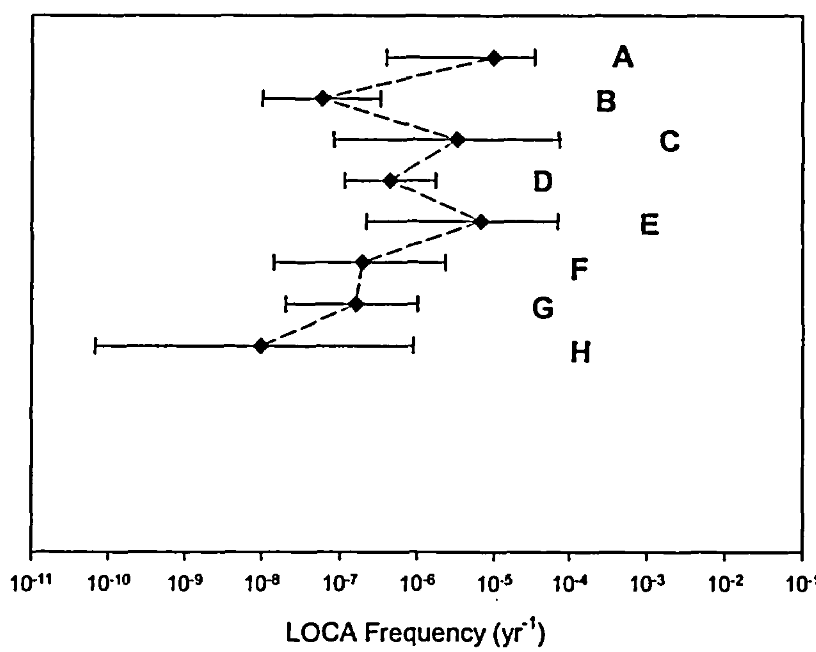


Figure L.30 BWR Non-Piping Category 3 LOCA Frequencies Showing Mid-Values, 5% Lower Bound, and 95% Upper Bound Values for All Participants Who Responded to the BWR Non-Piping Questions

L.5 PWR Non-piping

The same three major non-piping components (RPV, valves, and pumps) as considered for BWRs are considered for PWRs, plus the steam generator and pressurizer are added. One of the service history databases showed an order of magnitude higher incident rate for PWR non-piping than BWR non-piping. This was partially attributed to the fact that there are more PWRs than BWRs. However, this comparison is also biased by the large number of steam generator tube failures reported in the databases. Steam generator tubes are subjected to a host of degradation mechanisms: fatigue, denting, external SCC, PWSCC, and overload failures. It was almost universally accepted that steam generator tube ruptures would be the dominant contributor to the PWR Category 1 non-piping LOCA frequency. In fact the PWR steam generator tube failure frequency is the dominant contributor to the overall PWR small-break LOCA (Category 1 LOCAs) frequency when considering both the piping and non-piping contributions. Even so, it is the expectation of a number of the panel members that the steam generator tube contribution to the small-break LOCA frequency will decrease with time due to steam generator tube replacement programs and improvements made to the secondary side water chemistry.

In general, many of the same degradation mechanisms that are important for PWR piping are important for the non-piping components as well. PWSCC is an important degradation mechanism for many of the smaller Alloy 600 components such as the CRDMs, heater sleeves, steam generator tubes, and other penetrations. As was the case for piping, the likelihood of multiple cracks forming, and possibly coalescing, and the relatively fast propagation rates associated with this type of cracking makes this mechanism a major concern from a LOCA perspective. Also, since this mechanism is more severe at the higher temperatures associated with PWRs, it is considered to be a bigger threat for the PWRs than the BWRs, at least in the short term. As was the case for PWR piping, thermal fatigue is also a concern for PWR non-piping components. It is especially of concern at nozzle inlets and other locations where thermal stratification may exist. Furthermore, for all the reasons highlighted for piping, thermal fatigue is a mechanism that can lead to large leaks, i.e., fast propagation rates, attacks a wide area, and difficult to prioritize inspection protocols due to the fact that it can attack a variety of materials. Mechanical fatigue is another common degradation mechanism to both PWR piping and non-piping components. Mechanical fatigue is most important for smaller components, such as heater sleeves and small penetrations that are subjected to vibratory stresses due to equipment operation.

Another mechanism of special concern to non-piping components is common cause bolting failures. This is especially relevant to manways and bolted valves. The common cause mechanism could be improper installation or maintenance of bolts, e.g., improper torque, external corrosion of multiple bolts, or possibly steam cutting of multiple bolts.

Also, boric acid corrosion of carbon steel components such as RPV and steam generators can be aggressive under certain conditions.

Figure L.31 shows the Category 1 LOCA frequencies for the major PWR non-piping components at 25 years. As expected, the expected failure frequencies are highest for the steam generator. The higher LOCA frequency for the steam generator is driven by the stream generator tube rupture data.

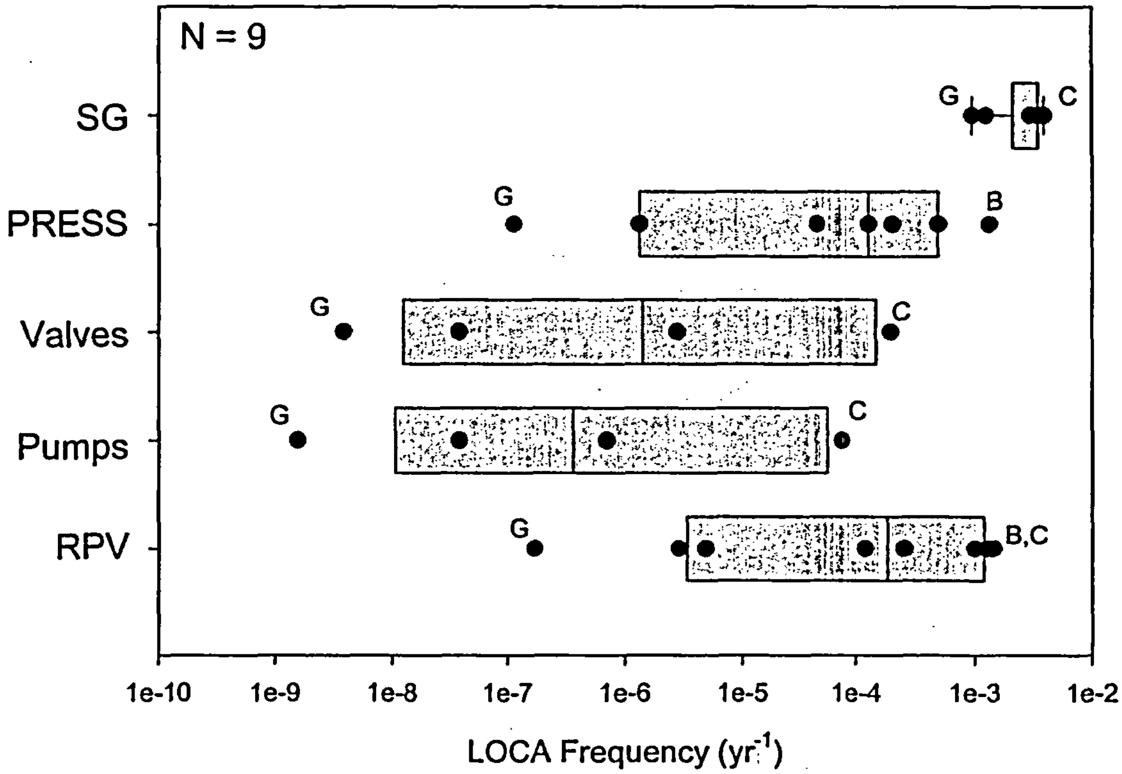


Figure L.31 PWR Category 1 Non-Piping LOCA Frequencies by Major Component at 25 Years of Plant Operations

The other major contributors to the Category 1 LOCA frequencies for PWR non-piping were the RPV and the pressurizer. The main subcomponent contributing to the RPV frequency is the CRDMs while the main subcomponent contribution to the pressurizer frequency is the heater sleeves. For Category 2 LOCAs, a single steam generator tube rupture cannot sustain such a leak. Thus, for the Category 2 LOCAs, the CRDM and pressurizer heater sleeves became the main contributors.

For Category 3 and 4 LOCAs there was no consistent agreement among the experts as to the major contributors. As one can see in Figure L.32, all five major components contribute fairly equally to the Category 4 LOCA frequencies. As such, there is tremendous variability about the frequency associated with each component. This variability was also apparent for the Category 6 LOCAs. This variability reflects the inconsistent opinions and approaches followed by the experts, as well as the difficulty of this type of assessment. As is to be expected, there was a wide array of possible failure modes for dissimilar components to be considered, and the experts tended to gravitate towards a few of the failures that they personally thought were most credible. Given all of this, the level of variability was thought to be reasonable given the event frequencies. This was one reason for adopting the elicitation approach in the first place. The highest LOCA frequencies were for the pressurizer nozzle. In addition, many of the experts considered manway or shell failures important, irrespective of the component type. Thus, they anticipated similar distributions for both the steam generator and pressurizer. There were also major differences of opinion among the experts as to the most important failure modes.

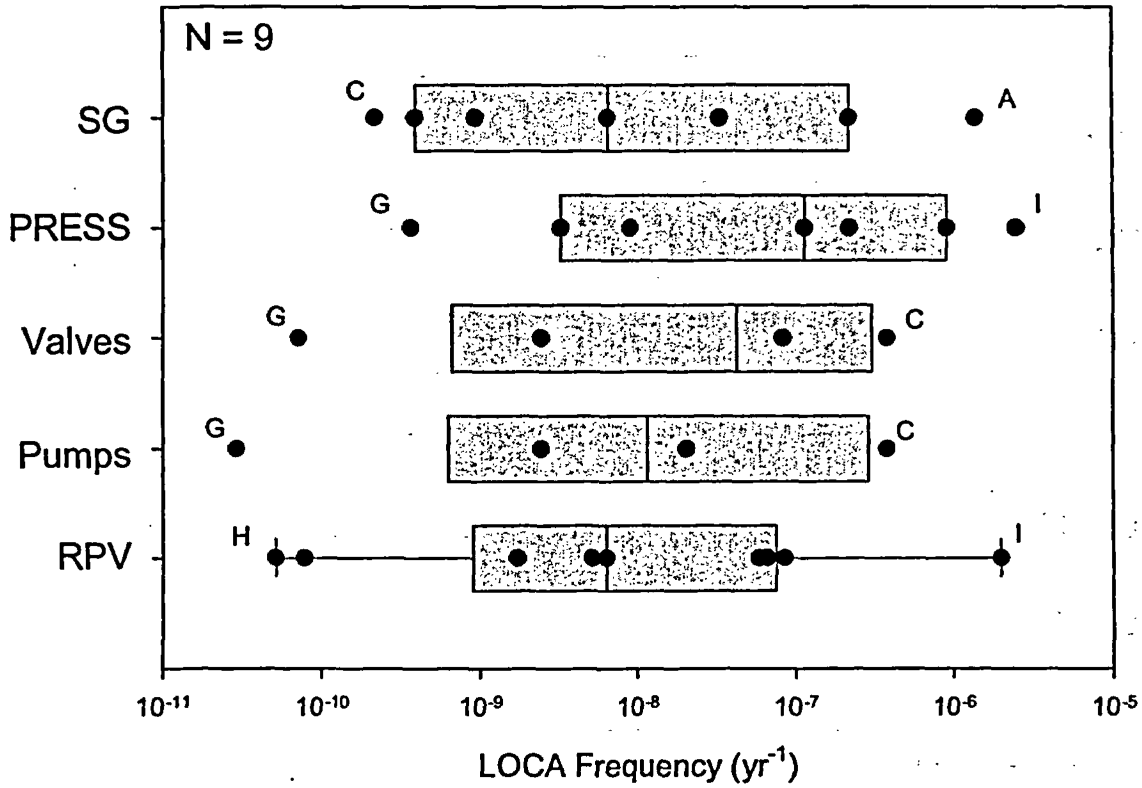


Figure L.32 PWR Category 4 Non-Piping LOCA Frequencies by Major Component at 25 Years of Plant Operations

Figure L.33 shows the cumulative LOCA frequencies for the PWR non-piping components at 25 years of plant operations. The Category 1 LOCA frequencies for PWR non-piping are the highest frequencies estimated by the elicitation panel for piping or non-piping, BWR or PWR. The median frequency is almost 5×10^{-3}. The variability among the experts was very small. The difference between the minimum and maximum predictions was less than an order of magnitude. These high frequencies and low variability were driven by the steam generator tube rupture data for which ample data exist in the service history databases; thus explaining both the high frequencies and excellent agreement between participants. For the Category 2 LOCAs, the agreement, at least on a minimum and maximum basis, is not nearly as good. However, the agreement on the basis of the spread in the IQR is nearly as good as it is for the Category 1 LOCAs. Again, for the Category 2 LOCAs, the major contributors are the CRDMs and the pressurizer heater sleeves. The much wider variability for the Category 3 through 6 LOCAs reflects differences in opinion among the experts as to the important failure modes and their associated frequencies.

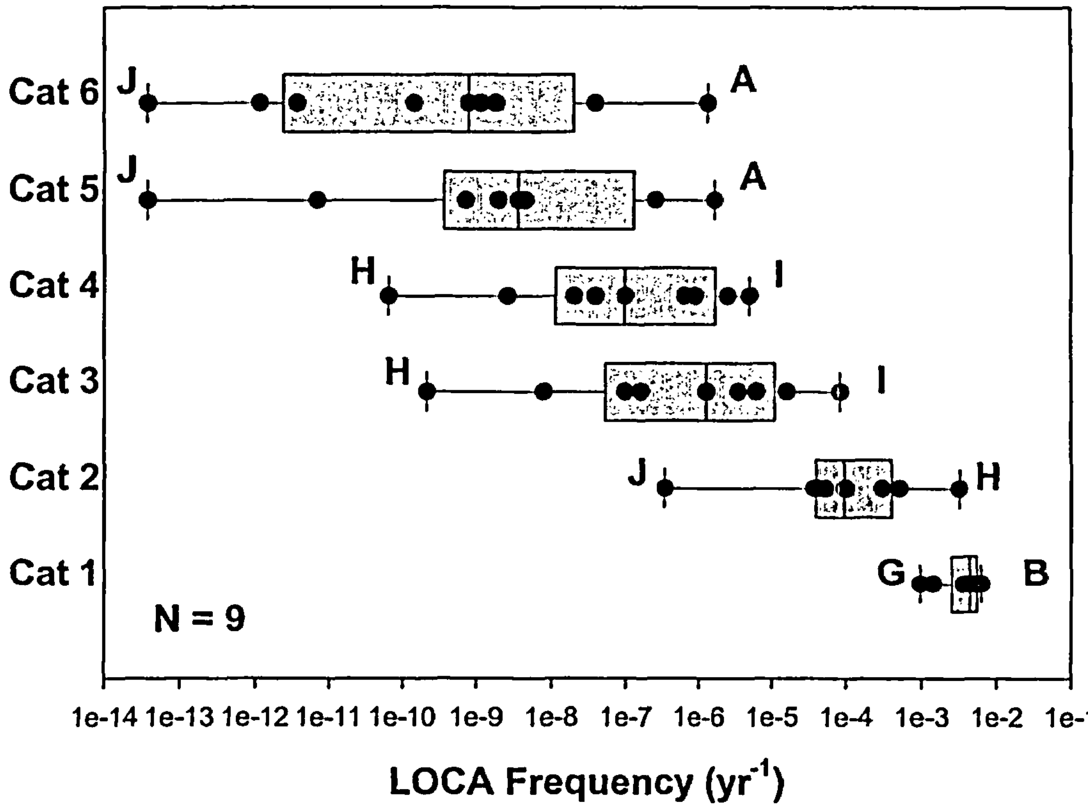


Figure L.33 Cumulative PWR Non-Piping LOCA Frequencies at 25 Years of Plant Operations

Figure L.34 shows the effect of time on the PWR non-piping Category 1 LOCA frequencies. There is a very slight decrease in the frequency between 25 years (present day) and 40 years (end of life) due mostly to steam generator replacement programs and improved inspection and mitigation programs, e.g., improved eddy current inspection programs and improved secondary side water chemistry. There was also an expected decrease in the LOCA frequencies associated with CRDMs due to on-going head replacement programs and improved CRDM inspection programs that may go into effect over the next few years. However, there was some concern expressed that the maintenance and inspection programs for the larger component bodies (pressurizer, steam generator, RPV) may not be as rigorous as for the piping systems. Figure L.35 shows the effect of operating time on the PWR non-piping Category 6 LOCA frequencies. As can be seen in Figure L.35, the median values remain constant with time and the variability among the participants (at least on the basis of the IQR) also remains fairly constant. This tends to indicate that the participants did not foresee any significant aging effects to occur.

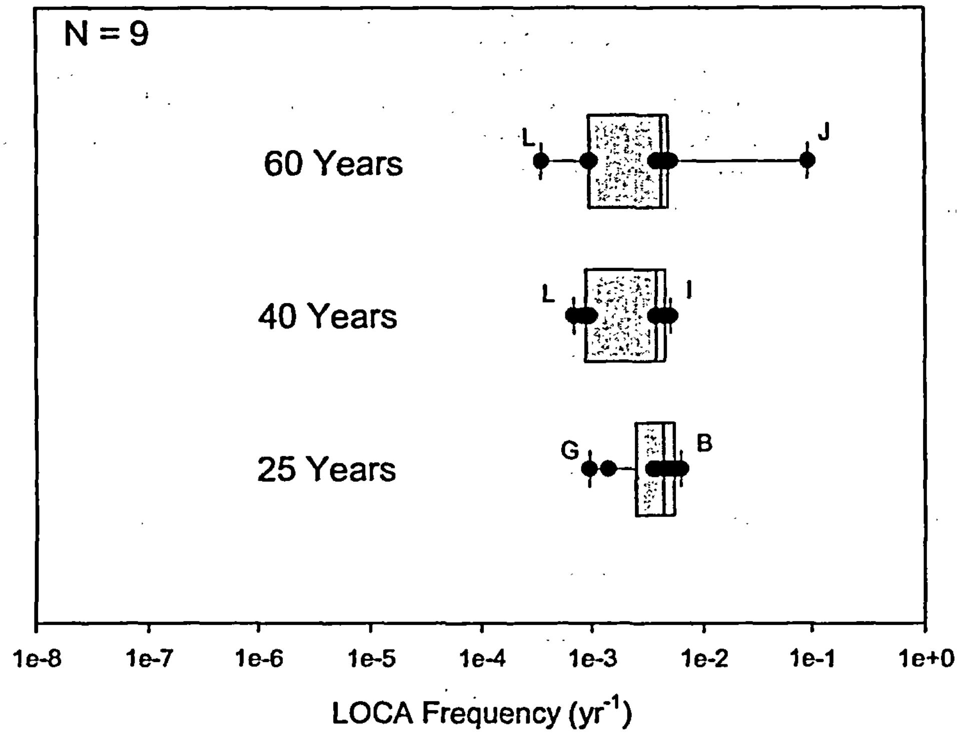


Figure L.34 Effect of Operating Time on the Cumulative Category 1 LOCA Frequencies for PWR Non-Piping Components

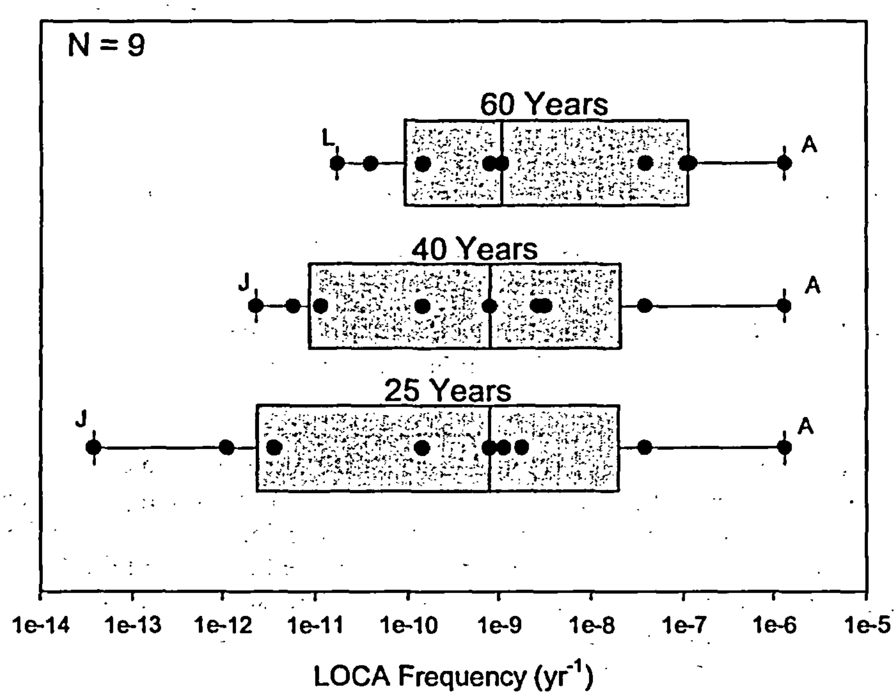


Figure L.35 Effect of Operating Time on the Cumulative Category 6 LOCA Frequencies for PWR Non-Piping Components

Contrary to what was observed for the Category 1 and 6 LOCA frequencies, the median value of the Category 4 LOCA frequencies increases an order of magnitude between 25 and 40 years and then remains constant after that, see Figure L.36. As was the case for PWR piping, aging was thought to have the largest impact on LOCA Categories 3 and 4. It was thought by some that aging could accelerate near the end of the plant life faster than the effects of mitigation and inspection could become effective, especially if the plant operators do not see a return on their investment for such inspection and mitigation programs near the end of the plant's life.

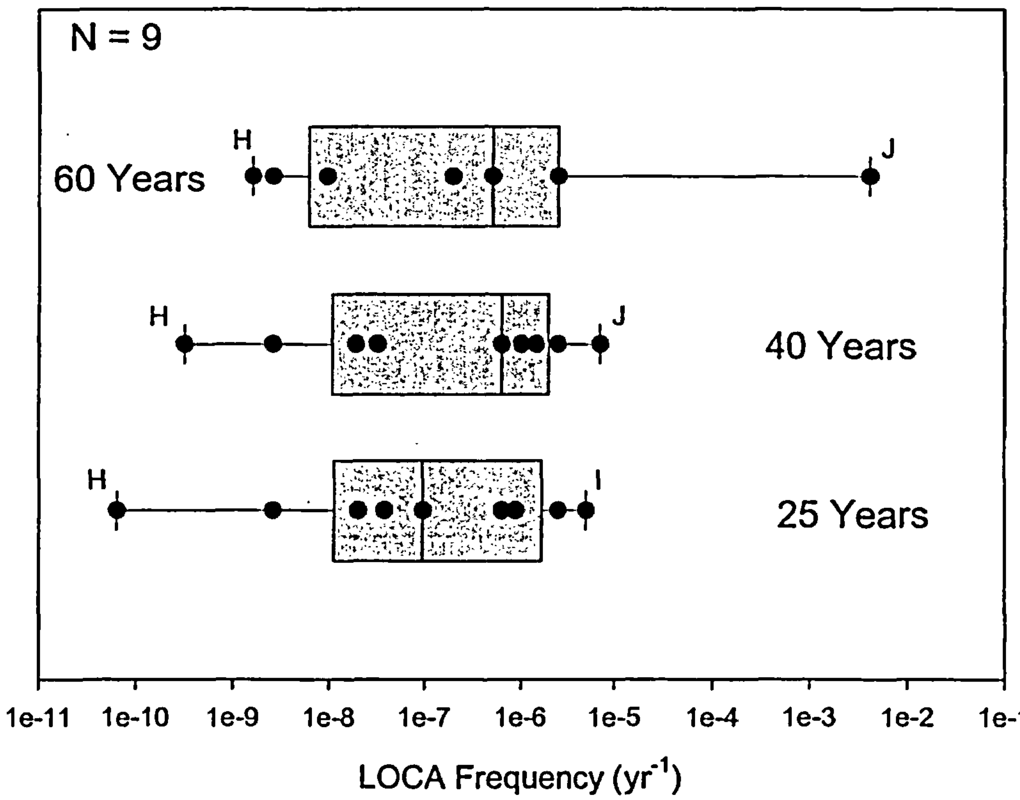


Figure L.36 Effect of Operating Time on the Cumulative Category 4 LOCA Frequencies for PWR Non-Piping Components

Figures L.37 and L.38 show the cumulative mid-value estimates, along with the 5% and 95% bound values for the various participants for the PWR Category 1 and 4 non-piping LOCA frequency estimates, respectively, at 25 years of plant operating time. Of note from these figures is higher uncertainty among almost all of the participants for the Category 4 LOCAs when compared with the Category 1 LOCAs. A number of the experts showed 3 to 4 orders of magnitude of uncertainty for the Category 4 LOCAs while all of the experts had less than approximately 2 orders of magnitude of uncertainty in their Category 1 results. In addition, the variability in the expert's mid-value estimates was within 1 order of magnitude for their Category 1 results while the variability among their results spread over a range of almost five orders of magnitude for their Category 4 results. The fact that the agreement among the experts was so good for the Category 1 predictions plus the low uncertainty of their individual predictions is a reflection that there was near consensus agreement that the single overwhelming dominant contributor to this class of LOCAs was steam generator tube ruptures, for which ample field experience is available in the service history databases.

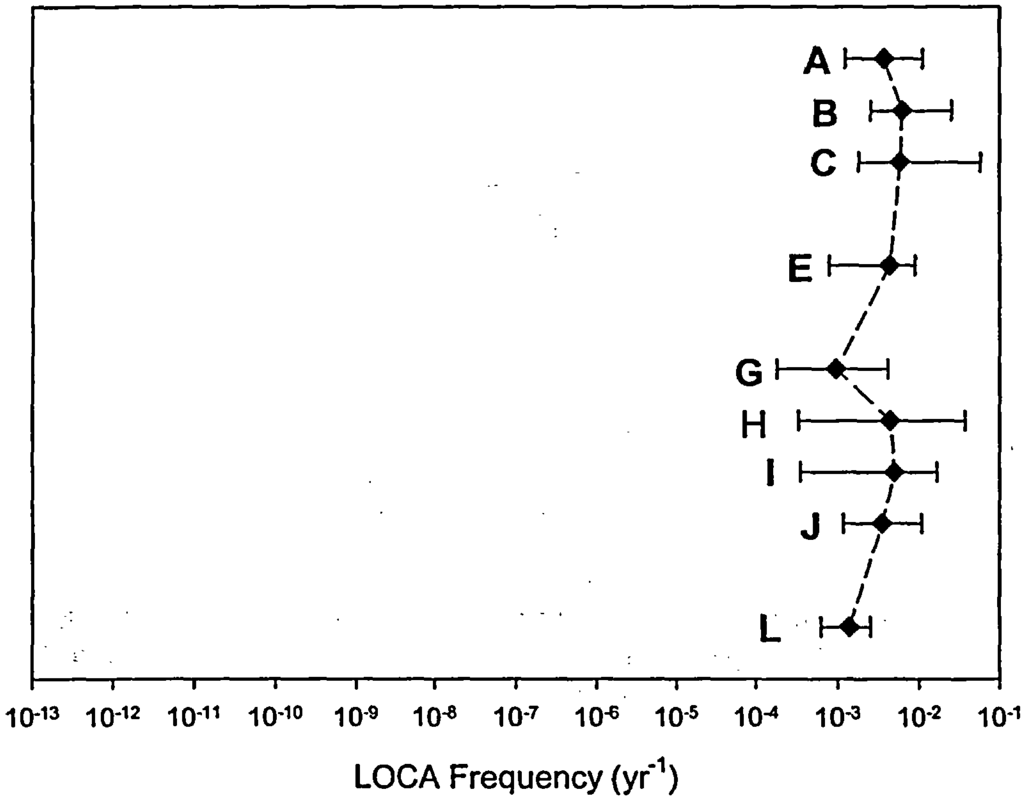


Figure L.37 PWR Non-Piping Category 1 LOCA Frequencies Showing Mid-Values, 5% Lower Bound, and 95% Upper Bound Values for All Participants Who Responded to the PWR Non-Piping Questions

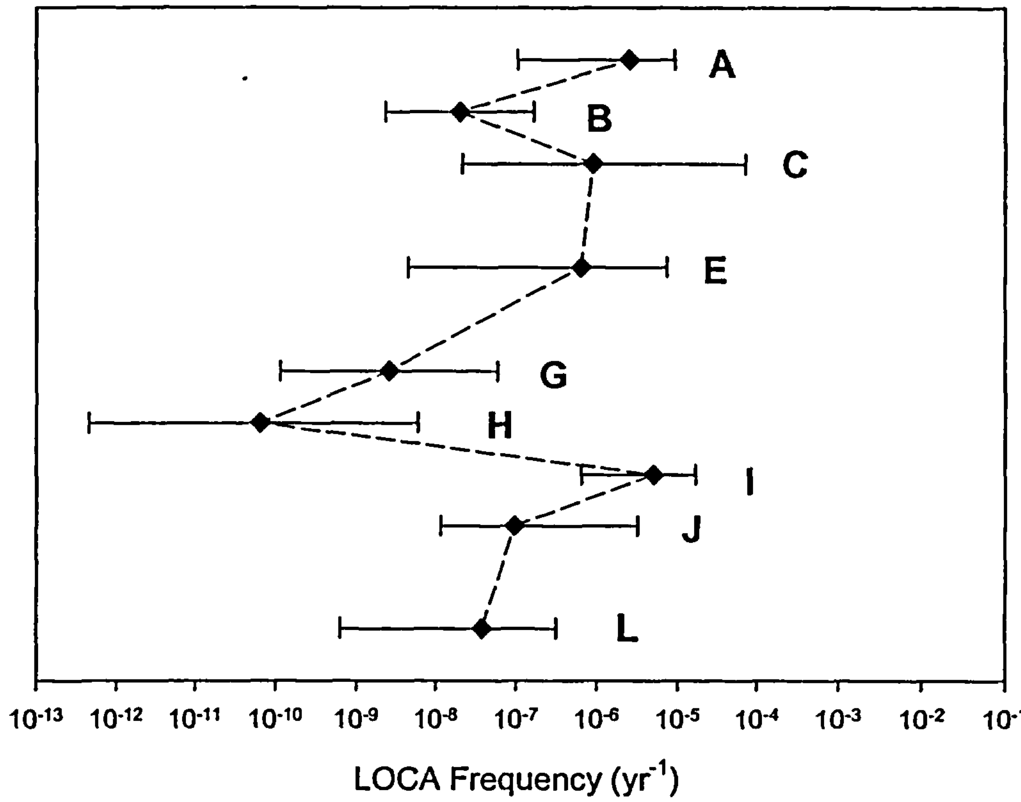


Figure L.38 PWR Non-Piping Category 4 LOCA Frequencies Showing Mid-Values, 5% Lower Bound, and 95% Upper Bound Values for All Participants Who Responded to the PWR Non-Piping Questions

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Charles Greene, NRC Project Manager

11. ABSTRACT *(200 words or less)*

The emergency core cooling system (ECCS) requirements are contained in 10 CFR 50.46, Appendix K to Part 50, and GDC 35. Consideration of an instantaneous break with flow rate equivalent to a double-ended guillotine break (DEGB) of the largest primary system in the plant generally provides the limiting condition in the required ECCS analysis. However, the DEGB is widely recognized as an extremely unlikely event. Therefore, the NRC is developing a risk-informed revision of the design-basis break size requirements for operating commercial nuclear power plants. A central consideration in selecting a risk-informed design-basis break size is an understanding of the loss-of-coolant accident (LOCA) frequency as a function of break size. LOCA frequency estimates have been developed using an expert elicitation process to consolidate service history data and insights from probabilistic fracture mechanics (PFM) studies with knowledge of plant design, operation, and material performance.

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