Questions on LOCA Frequency Analysis

General:

1. How are failures that don't occur at welds considered? These include, for example, Control Rod Drive Mechanism (CRDM) failures, pressurizer heater sleeve failures, Steam Generator (SG) tube ruptures, bottom mounted instrumentation (BMI) nozzle failures, thermal fatigue failures at nozzles , component tees, and other mixing locations?

Response: We plan to address non-pipe related failures and non-weld related failures in 2012 so the focus in 2011 is to consider pipe breaks. Based on our understanding of the prevailing damage mechanisms we believe that LOCA frequencies will be dominated by failures at weld locations including pipe to safe end and nozzle locations. However considering the fact that there are many welds distributed rather uniformly over the pipe runs the weld locations will provide an opportunity to evaluate many detailed pipe break locations. It is not clear that moving the assumed break locations between weld locations will have a significant impact but if it does then we may need to consider such break locations in the LOCA frequency analysis. This is an issue that can be better examined once a number of break locations have been fully analyzed. We acknowledge that we will eventually need to include non-weld effects or demonstrate why adding them will not be risk significant. As far as non-weld and non-pipe locations such as CRDM failures our approach to addressing them has not yet been defined. SGTR are modeled in the PRA but it is not clear how tube failures would generate any debris. Our submittal will include a justification of which locations were considered and those that were screened out and why.

2. How are Loss-coolant Accidents (LOCAs) caused by overpressurization events such as water hammer or other P-T overpressurization events (which may cause vessel failure due to embrittlement) considered?

Response: The potential for water hammer in the Class 1 piping was assessed in the RI-ISI evaluation and determined not be a credible failure mechanism for LOCA sensitive piping. The PRA model explicitly considers over-pressurization during loss of main FW + ATWS conditions and calculates a probability of vessel failure. While such failures are in fact modeled in the PRA, they will not contribute to the CHANGE in CDF or LERF from design or operational changes to address GSI-191 – core damage will be assumed for the current design as well as the revised design if the vessel fails.

3. How are LOCAs caused by pressurized thermal shock (PTS) considered in the analysis?

Response: The PRA model includes a model for PTS induced vessel failure from thermal shock but again, no credit is taken to justify core damage prevention following vessel failure. Our understanding of the NRC research on PTS indicates that PTS induced vessel failure is not risk significant but even if it was when the vessel fails, the issue of debris formation is moot as far as the PRA model is concerned.

NOTE: The model considers vessel integrity following one class of PTS events, namely excess steam flow in response to a transient. For example, the model queries whether the vessel integrity is maintained following a plant trip and failure both of the turbine to trip and the MSIVs to close. PTS or thermal transient impact on the reactor vessel is not included in the medium or large LOCA event models. NUREG 1806 would suggest that a future update of the models should include thermal shock challenges in response to medium and large LOCAs or equivalent.

Please note that vessel integrity is also queried after an ATWS with failure of the MSIVs to isolate, in which case the failure mechanism of concern is overpressurization. This logic is included in the general transient and small LOCA models only.

4. How are non-passive system LOCA frequencies (e.g., Interfacing System LOCAs (ISLOCAs), seal LOCAs, active system LOCAs) considered?

Response: Interfacing LOCAs are explicitly modeled in the PRA. When they occur the pressure boundary failures are outside the containment and not relevant to causing debris induced core damage inside the containment. For ISLOCAs the core damage is due to the diversion of coolant outside the containment, bypassing the sumps, and inability to establish recirculation flow. Seal LOCAs due to active failures are included in the PRA model and the GSI-191 evaluation will evaluate the degree to which such failures can generate debris. If they are shown to have the capability to generate debris, they will be explicitly included in the analysis.

5. When and how is probabilistic fracture mechanism (PFM) being used to help determine LOCA frequency estimates (slide 6)?

Response: We have no plans to perform any new PFM analysis in 2011 but are leaving open to perform such calculations if needed based on the results this year. The focus in 2011 is to evaluate a wide spectrum of cases that span the entire Class 1 pressure boundary. PFM analyses are more appropriate for more focused evaluations at specific and limited number of locations. As the project proceeds and when and if it is determined that some specific locations are especially important and would benefit from a PFM we will include that but at this stage it is too early to tell whether this will be necessary.

Uncertainties:

1. How are aleatory and epistemic uncertainties considered and separated (slide 5)?

Response: The aleatory uncertainties are reflected in the assumption that LOCAs occur as a Poission process so the frequency of a LOCA is a metric for the aleatory uncertainty about whether a LOCA will occur or not. The uncertainty distributions we develop around the LOCA frequencies represent primarily epistemic uncertainties. However we recognize that separating sources of uncertainty into these bins is subject to its own form of uncertainty and can be debated. Our LOCA frequency model of uncertainty assumes that the LOCA frequency is a metric of aleatory uncertainty and the uncertainty distribution we develop that is around that is the epistemic type. We realize this is just a model.

2. l uncertainties discussed in slides are characterized as epistemic uncertainties. What is the basis for this characterization?

Response: See above response to item 1.

3. How is PIPExp database used to resolve uncertainties (slide 6)? It's unclear from the slides.

Response: We mean to say that information contained in the database and insights developed over many years in collecting and analyzing the data helps to reduce the level of uncertainty that we experts inputs in NUREG-1829. In addition an earlier version of the same database provided useful input to the last NRC sponsored project on LOCA frequencies, NUREG/CR-5750. Before such data was collected, for example back in Wash-1400, very little data on nuclear piping systems was either generated or analyzed. The entire effort to implement RI-ISI programs was supported by insights from the service data. An early example of the kind of application that this database has in reducing uncertainties is to guide the judgments on how to define homogeneous weld populations in to groups for failure rate estimation. Knowledge of the damage mechanism responsible for the experienced pipe failures is essential for guiding these judgments (e.g. need to separate bi-metallic welds subject to PWSCC from other welds not subject to this damage mechanism)

4. How are NUREGs 1829 and CR-5750 being used to quantify epistemic uncertainties (slide 6)?

We are preparing a slide presentation to use during our July 7 conference call to walk through examples of how we plan to use these references to incorporate epistemic uncertainties into the STP specific LOCA frequencies. To briefly summarize we plan to incorporate information from NUREG-1829 to establish the uncertainty distribution parameters of our model for the conditional probability of LOCA vs. LOCA category.

5. Why does N_i have uncertainty? Isn't the number of welds known?

There are two reasons for this uncertainty. One is plant to plant variability. Each plant has a different number of welds for a given component, there are different numbers of coolant loops in the PWR population (2,3, and 4), different number of interfacing system connections like ECCS, etc. etc. The second reason is that even though these numbers are known with each plant's organization, there are only publically available counts for some specific plants. Based on a limited sample which will be documented in our submittal, plant to plant variability is responsible for a factor of 2 above and below the "best estimate" for many pipe weld categories.

6. Why is there little uncertainty associated with the number of failures (n_{ik} in slide 9)? Doesn't little uncertainty in this parameter assume that database has complete coverage of all events and that no other "failures" have occurred?

Based on our experience, there is very little uncertainty in pipe failure counts for the Class 1 pressure boundary based on the PIPExp data and certainly much less than the uncertainty we are assigning to the component exposure. Also the whole idea of using a Bayes' method for estimating failure rates is based on the idea of starting with a prior distribution that models a very large uncertainty. In our approach the priors are assumed to be lognormal with range factors of 100.

Calculation of LOCA estimates:

1. How is the integrity management factor (I_{ik}) calculated (slide 8)? Is the Markov model used to determine I_{ik}?

Response: Yes, the Markov model is used to calculate this factor. This calculation approach was worked out for the EPRI RI-ISI and is extensively documented in the attached references. The first report on the Markov model is Reference [1] and the initial pipe failure data developed for use of this model in Reference [2]. EPRI sponsored reviews of this work are documented in Reference [3] which is included as an appendix to Reference [1]. This model and data were developed initially in order to support estimates in the change in CDF and LERF due to changes in weld selections for NDE as part of the EPRI RI-ISI program. The use of the Markov model to calculate inspection factors was first documented in the EPRI RI-ISI Topical Report in Reference [4]. The NRC safety evaluation of the EPRI Topical Report includes findings that approve the use of the model and the supporting data for the EPRI RI-ISI evaluations. This review was supported by an NRC sponsored review of the Markov model and the Bayes' failure rate method very similar to what we plan to use in this project done by LANL in Reference [5]. A peer reviewed journal article with many of the mathematical details of this method are found in Reference [6].

To summarize the Markov model is used to formulate ordinary differential equations which are solved analytically for the time dependent state probabilities. The input parameters for model which are the coefficients of the ODE are defined in terms of pipe failure mechanisms that produce flaws, leaks and ruptures, as well as parameters for the frequency and reliability of programs to detect leaks and inspect for flaws. Then from these solutions, analytical expressions are obtained for the hazard rate, which is kind of a time dependent rate of rupture. Due to the boundary conditions of the equations, the hazard rate increases with plant age (as seen in Appendix D of NUREG-1829). The inspection factor is the hazard rate at 40 years (or 60 years depending on the application) due to some specific integrity management program (combination of leak inspection and NDE) divided by the hazard rate at the same time for the average component with average integrity management.

2. The relationship between the flowchart (slide 12) and equations (1) – (3) (slides 8 and 9) is unclear. Please identify which specific terms in the equations are calculated by specific steps in the flowchart.

We plan to walk through an example in our July 7 presentation which will clarify each step in quantifying the LOCA frequencies.

3. There are many questions related to the flowchart (slide 12).
    a. Why does the number of leaks provide input to both the failure frequency and conditional rupture probability?

    The number of leaks contributes to the numerator of the failure rate estimate. The number of leaks also contributes to the denominator of the conditional probability of rupture estimate.

    b. Where is degradation mechanism (DM) susceptibility in Equations (1) – (3)? How does it factor into those equations?

    Response: We know from the service experience that some failures occur due to some specific damage mechanisms. We can calculate the unconditional failure rate from any damage mechanism simply dividing the number of failures by the total component years in the service data. But after we have a completed RI-ISI program such as the case with STP we know on a weld by weld basis which welds are susceptible to each damage mechanism. Now we need to calculate the conditional failure given we know the applicable damage mechanism. For this we need an estimate of the fraction of welds in the data base that produced so may failures due to each DM how many are susceptible to each DM. Hence the fraction f in the denominator of Equation must be applied to estimate the conditional failure rate. Leaving it out would yield the unconditional failure rate.

    We shall also address this question in the July 7 meeting.

    c. What experts are being used to provide various estimates? Are same experts used to provide each distribution indicated in figure?

    Response: In the case of information we bring in from NUREG-1829, it is the expert panel from that project. We may also need to incorporate expert judgments from our team that will be clearly documented in the submittal.

    d. Why do the DM susceptibility estimates provide input to both the generic prior distribution and in the Bayes update distributions?

    Response: The estimates of the fraction of welds in the generic population that are susceptible to the damage mechanism are used to determine the parameters of the likelihood functions for the Bayes' update of prior distributions which are intended to represent the failure rates for components susceptible to those mechanisms. They need to be consistent. Our July 7 presentation will aim to clarify this.

    e. How, specifically, is the Bayesian update of the prior distribution performed using the three distributions generated to inform the prior?

    Response: The prior are assumed to be lognormal. We use a Poisson likelihood function to update these with one set of data for each hypothesis of weld population and weld susceptibility fraction. This yields several different posteriors that are combined using what is referred to as Bayes' posterior weighting. We will explain this more clearly during July 7 presentation.

f. How is the Beliczy-Schultz correlation used to create the P(R|F) prior distribution?

Response: As you know, this correlation was used by Bengt Lydell as input to this distribution for the base case analyses in Appendix D of NUREG-1829. In the STP we plan to base the priors for this distribution using information from NUREG-1829 which will be explained during the July 7 meeting.

g. How is the P(R|F) prior updated using Bayes?

Response: We perform a Bayes' update for each discrete LOCA category, which is associated to a break size. We use a truncated lognormal distribution to represent the uncertainty in the conditional probability of LOCA at each Category separately. We update it with evidence of normally 0 LOCAs and N failures where N is the number of failures used to calculate the corresponding failure rates. We will show this in the July 7 meeting.

1. The calculation procedure and the application of the Markov model in both the flowchart and equations is unclear. Also, information/documentation on the used of this model for other NRC-approved applications and other nuclear applications should be provided. Are there differences between the model being used for these estimates and what has been approved and used in other applications?

Response: Please see the above question response on the inspection factor and provided references. Any deviations from previously reviewed applications will be fully documented in the submittal and supporting reports.

2. How is the Markov model different than P(R|F)? Is this model used to determine P(R|F)?

Response: P(R|F) is used to calculate rupture frequencies. Failure rates for flaws, leaks, and ruptures are input to the Markov model to develop the integrity management factors.

3. In the Markov model there is no probability of rupture given no detectible damage. Why is this term neglected? Doesn't this presume that ISI is perfect?

Response: This model is only used and will only be applied for ruptures due to degradation mechanisms. There is another version of the model developed in Reference [1] that includes additional transitions for leaks and ruptures absent a detectable flaw. The reason for not including those transitions is that pipe failures due to severe loading conditions are modeled explicitly by non LOCA type initiators in the PRA model.

4. There are a number of questions related to slide 33
    a. What is the basis of the hazard rate {h(t)} equation?

    Response: This is standard reliability engineering theory. The hazard rate is defined as the negative of the rate of change of the reliability function (probability of no rupture) divided by the reliability function. In this Markov model, the reliability is the sum of the state probabilities for success, flaw, and leak. See Reference [7] for the mathematical details.

    b. What is r(t)?

    Response: This is the reliability function referred to in item a.

    c. How is h(t) normalized?

    Response: See the above question/response on the Integrity Management factor

    d. What sensitivity analyses on h(t) are performed??

    Response: This refers to changing the assumptions about: whether or not there is a leak detection program and if there is, how often it is done and what is the effective probability of detection; and whether or not there is an NDE program, and if there is how often it is performed and what is the effective probability of detection. For each combination of leak detection and NDE inspection parameters, a different result is obtained.

Questions on specific slides:

1. Slide 17: What is WH (water hammer)? Yes
2. Slide 19:
    a. How are unconditional failure rates determined? By leaving f out of the denominator in Equation (3)
    b. Are the "conditional" estimated determined through expert elicitation to determine bump-up factors for the "unconditional" estimates (i.e., bump-up of app. 5 for thermal fatigue)? If not, how were the conditional estimates determined and what do they mean? No the results of the RI-ISI DM evaluation are used to resolve deterministically which welds are subject to each DM
3. Slide 22: What are "excessive" LOCAs? Vessel failures and multiple pipe breaks and any LOCA that is beyond the capabilities of the ECCS according to the PRA success criteria.
4. Slide 38: Why does inspection stilly yield a positive ΔCDF for reactor coolant pump (RCP)? While it's a lower number than if no inspection is performed why does it still add risk to the plant? RCP stands for reactor coolant system piping. There is an increase because of the fact that many welds were removed from the RI-ISI program. This is typical in all RI-ISI programs.

References

[1] Piping System Reliability and Failure Rate Estimation Models for Use in Risk-Informed In-Service Inspection Applications. EPRI, Palo Alto, CA: 1998. TR-110161.

[2] Piping System Failure Rates and Rupture Frequencies for Use in Risk-Informed In-Service Inspection Applications. EPRI, Palo Alto, CA: 1999. TR-111880.

[3] Mosleh, A. and F. Groen, “Technical Review of the Methodology of EPRI TR-110161”, University of Maryland report for EPRI, published as an Appendix to EPRI TR-110161 (Reference [6])

[4] Revised Risk-Informed In-Service Inspection Procedure. EPRI, Palo Alto, CA: 1999. TR-112657, Rev. B-A.

[5] U.S. Nuclear Regulatory Commission, Safety Evaluation Report Related to Revised Risk-Informed In-Service Inspection Evaluation Procedure: EPRI TR-112657, Rev. B, July 1999, Washington, D.C., 1999. (published as a forward to TR-112657 (Reference [9])

[6] Martz, H., TSA-1/99-164: Final (Revised) Review of the EPRI-Proposed Markov Modeling/Bayesian Updating Methodology for Use in Risk-Informed In-Service Inspection of Piping in Commercial Nuclear Power Plants,” Los Alamos National Laboratory, June 1999.

[7] Fleming, K. N., “Markov Models for Evaluating Risk Informed In-Service Inspection Strategies for Nuclear Power Plant Piping Systems,” Reliability Engineering and System Safety, Vol. 83, No. 1, pp. 27–45, 2004.