From: George Dick
To: INTERNET:joseph.bauer@exeloncorp.com
Date: Thu, Feb 15, 2001 10:44 AM
Subject: Braidwood Discussion Points for RI ISI Meeting

Joe,

Please see attached.

George

CC: Anthony Mendiola, Lawrence Rossbach, Mahesh Chaw...

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Subject: Braidwood Discussion Points for RI ISI Meeting
Creation Date: Thu, Feb 15, 2001 10:44 AM
From: George Dick

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BRAIDWOOD UNITS 1 AND 2 RI-ISI
Discussion Points

1. In accordance with the guidance provided in RGs 1.174 and 1.178, an engineering analysis of the proposed changes is required using a combination of traditional engineering analysis and supporting insights from the probabilistic risk assessment (PRA). The purpose of the traditional engineering analysis is to ensure that the impact of the proposed ISI changes is consistent with the principles of defense-in-depth. Based on the staff's experience with the review of RI-ISI submittals, the percentage of volumetric inspection of ASME Class 1 welds has ranged from about 7% to 12%. In cases where the original proposal was for less than 10% volumetric inspection of these welds, the staff has been requesting that the sample obtained by the risk-informed process be increased to obtain a 10 % level of inspection sample by selecting elements for inspection to obtain a distribution of inspections among various systems including considerations of various potential degradation mechanisms. This request is based on the staff conclusion that a minimum of 10% volumetric inspection sample of ASME Class 1 welds is needed for the staff to find that an acceptable level of defense-in-depth is being provided. The Braidwood submittal states that 8.9% of the Class 1 welds for Unit 1 will be volumetrically inspected. Please clarify numbers of total category B-F and B-J welds, and numbers of butt welds performing volumetric inspection in each category in the RI-ISI program to ensure that a minimum of 10% stated above is met.

2. Please clarify the following:

   a) In the second page of the transmittal letter, the licensee provided the "start" and "end" dates of the ISI periods. For Period 2 in both units, the year in the start dates are marked 2001. However, the years for the end dates of Period 1 are 2002. Please clarify.

   b) In attachment 1, on page 2 of 4, item c for all dissimilar metal welds in the category B-J, the licensee should indicate that these dissimilar welds include those not covered by the B-F as indicated in the Note (c) of the ASME Code Table IWB-2500-1 for category B-J.

   c) In attachment 1, on page 2 of 4, the licensee discusses the Table IWC 2500-1 requirements for category C-F-1. However, similar discussions for C-F-2 are missing in the submittal RR 12R-39, Revision 0. Please explain.

   d) Is there any recognizable plant experience on piping failures at Braidwood?

   e) What is the minimum pipe diameter included in the RI-ISI evaluation and program?

   f) Both Tables 5 and 6 included the Risk Category 4 in the High Risk columns. Should these be under Medium Risk columns?

3. In accordance with the Section 3.2.3 of the SER to the EPRI topical report, a pipe segment susceptible to a degradation other than flow accelerated corrosion (FAC) and which also has the potential for water hammer receives high pipe failure potential. The licensee has not identified water hammer as a potential degradation mechanism for selected pipe segments.

Clarify if any of the selected system welds are susceptible to water hammer and any other aging mechanism than FAC.

4. Please provide a reference to the version of the PRA used to support the RI-ISI submittal. Please also provide the CDF and the LERF estimates from the PRA version used to support the RI-ISI submittal.

5. Page 6 states that "The potential for synergy between two or more damage mechanisms working on the same location was considered in the estimation of pipe failure rates and rupture frequencies which was reflected in the risk impact assessment." Specifically how was this synergy reflected in the risk impact? Was synergy also reflected in the safety significant categorization and if so how?

6. Page 5 states that, "If no other damage mechanism was identified, the element was removed from the RI-ISI element selection population and retained in the appropriate augmented program." Does "removed from the RI-ISI element selection population" mean that all welds within a medium ranked segment that is included in the FAC program, for example, are excluded from the required 10% and that discontinued Section XI inspections within the segment will not be included in the change is risk calculations? If not please explain what removed from the population means. Does the reported 8.9% and 10.1% of Class 1 butt welded elements inspected include the population of Class 1 HELB and the FAC element welds in the denominator?

7. The licensee has included the essential service water system (SX) within the scope of the RI-ISI program but chose not to subsume the service water inspection program. The licensee has also included the containment purge system (VQ) within the scope of your RI-ISI. Neither SX nor VQ appear in the tables identifying inspection locations selected for RI-ISI. Were there any segments in SX or VQ that had a medium or a high consequence ranking? How many Section XI inspections are currently being performed in VQ and SX?

8. In the note to Table 4 regarding Unit 2 the licensee indicates that the difference in the distribution of welds in the different risk categories is due primarily to the Unit 1's steam generators (SGs) being replaced whereas Unit 2's SGs has not been replaced. Please explain how the replacement of the SGs could cause such a large reduction in the number of Unit 1's Category 3 FW (108) and Category 4 RC (23) locations as compared to Unit 2. Additionally, the total number of welds in the systems seems to vary substantially between the two units. For example Unit 1 has 104 less FW and 27 less RC welds than Unit 2, but 65 more SI welds than Unit 2. Do these difference in total welds reflect actual physical difference between the piping systems in the two units?

9. Page 12 of the submittal discusses a "separate Markov calculation" for the change in LERF for lines connected to the RC that continue outside containment. Normally such lines have an inboard and an outboard isolation valve. A rupture outside containment and failure of the inboard isolation valve will result in an unisolatable LOCA outside of containment. Is this the scenario that is being addressed here? If this is not the scenario, please provide an example to illustrate the scenario. The methodology in EPRI TR-112657 includes a semi-quantitative technique for this situation in Table 3-14. Alternatively, the probability of the inboard isolation valve failing can be factored into the CLERP probability. If the licensee's methodology deviates

from the EPRI TR-112657 for unisolatable LOCAs, please provide a comparison of the licensee's method with the accepted method.

10. If the calculations are performed using the data from the Tables in TR-111880 instead of the updated failure rates, the licensee may identify the Tables used instead of responding to question 11 on the Bayesian update. If results from the bounding evaluations described in the EPRI TR-112657 instead of the Markov calculations are sufficient to illustrate that the suggested change in risk guidelines are not exceeded, please provide a brief description of these evaluations and the results instead of responding to the questions 12 and 13 on the Markov calculations.

11. In Section 3.7 on pages 11, reference is made to the use of updated failure rates and rupture frequencies. The EPRI report, "Piping System Failure rates and Rupture Frequencies for Use in Risk Informed In-Service Inspection Applications," TR-111880 was completed in September, 1999. A copy of the report was submitted to the NRC in support of a RI-ISI relief request at another nuclear power plant. A draft version of the report was submitted to the NRC during the review of the EPRI Topical report TR-112657. As indicated by its tittle, EPRI TR-111880 contains tables of vendor and system specific parameter values to be used to support RI-ISI applications. The evaluation documented in TR-111880 was performed by a team sponsored by EPRI. This team developed required plant characteristics, evaluated individual failure events collected from plant operating experience, interpreted the observed experience, characterized and grouped the observed experienced, and calculated a specific set of suggested failure parameters. EPRI TR-111880 states that the values provided in the Tables includes about 905 years of operating experience for Westinghouse reactors. The licensee's submittal states that the updated parameters include 1000 years of experience. As illustrated in the following Table RAI-11, examination of your new parameters reveals a difference in the grouping of the systems and large differences in the parameter values. These differences do not appear consistent with an increases of 10% in the years of experience for rare events such as failures and ruptures. The differences appear to indicate differences in the judgements interpreting, and the subsequent manipulations of the experience data. The staff will need to fully understand any differences in the evaluation of the experience data, and the justification for these differences, to accept the plant specific data as an improved set of parameters that need to be used instead of the industry data to support the change in risk calculations in the Braidwood submittal.

    a) Please describe how the failure rates were updated; that is, were the rates in the table updated or were the original calculations performed with the new data? Please provide a reference to the equations' numbers in TR-111880 or TR-110161 that were used in the update.

    b) EPRI TR-111880 reported that, "[t]o provide the best possible estimates of pipe failure rates, rupture rates for each failure mechanism are calculated for eight different system groups, for each type of reactor vendors." In the table in the Braidwood submittal, the safety injection (SI) system and the residual heat removal (RH) system (both originally in the RAS group) are individually listed. In EPRI TR-111880 the two systems are assigned the same parameters, but there are very large differences between the systems' parameters in the submittal. The desire to balance resolution with available data reflects the Bayesian update procedure where, as less and less data is available, the result of the update become more and more dependent on the initial judgements and less and less on

the experience data. Please characterize the quantity of data available for each system's update and the impact that data has on the priors developed from judgement. Please justify the development of finer groupings and explain why this finer grouping is applicable to the analysis supporting this submittal but was not applicable for the TR-111880 calculations.

c) What is the range of dates used in the update and how many additional reactor years are in the update?

d) The Braidwood submittal states that the updated failure parameters reflect estimates of "weld population exposure" that were not available when TR-111880 was developed. Was it only exposure information that was collected or were any failures observed and also used to update the parameters? What information sources was used to estimate the extra years of exposure and to identify any failures that might have occurred during these years? Are these the same information sources that were used to develop the original estimates in EPRI TR-111880?

e) Please explain why all systems were not "updated," but rather, some (i.e., CS, SX, FAC, and ST) used the existing values from EPRI TR-111880.

f) Although not illustrated in the Table, the staff notes that the probability of rupture given a failure (P(R/F)) has been changed, in some cases, by almost a factor of 5 reduction. This change implies that there has been additional data collected on the number of observed ruptures and flaws. Both events are infrequent. Is this parameter being calculated as described in EPRI TR-110161? Please describe and summarize the experience data that was used to calculate the change in this parameter.

g) Please explain why there is such a wide variation in the magnitude of the changes when the same calender time was used for the update of all the parameters.

Table RAI-11

The entries give the factor change for the rupture failure frequencies between EPRI TR-111880 and Table 7 in the submittal. A "9 X reduction" means that the failure frequency in the submittal is 9 times smaller than in EPRI TR-11880

Damage mechanism	RC System	SI* System	CAC** System	RH* System
T.F.	25 X reduction	9 X increase	10 X reduction	6 X increase
SC	6 X reduction	negligible change	5 X reduction	5 X reduction
E.C.	2 X reduction	negligible change	10 X increase	60 X increase
DC	7 X reduction	2 X reduction	70 X reduction	4 X increase

*SIR in TR-111880
**RAS in TR-111880

12. Please provide references to all the equations that describe the Markov calculation that the licensee is using to calculate the change in risk. For example Equation 6.1 in TR-110161 refers to multiple failure sizes and multiple conditional core damage probabilities for each segment. Is the licensee using this equation? Please give the values of all the input parameters required by the equations and also provide a references from which the input parameters were developed and justified (except for the conditional core damage, condition large early release probabilities, and weld failure rates). For example, if the licensee is using Equations 3.23 and 3.24 in TR-110161, what values are being used for the parameters? Please provide specific references, e.g. equation numbers, table numbers, page numbers, and report references.

13. It is the staff's understanding that the Markov calculations include calculating an "inspection effectiveness factor" for use in equation 3-9 of EPRI-TR 112657. Please provide the distribution of inspection effectiveness values calculated and a discussion on how these values compare with the direct use of the probability of detection estimates.