From: George Dick
To: INTERNET:Joseph.Bauer@exeloncorp.com
Date: 3/5/01 3:32PM
Subject: RI ISI Questions for Byron

Joe,

Please see attached questions related to the Byron relief requests of November 17, 2000.

Thanks,

George

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(3AA3F7F2.527 : 22 : 21110)

Subject:	RI ISI Questions for Byron
Creation Date:	3/5/01 3:32PM
From:	George Dick
Created By:	GFD@nrc.gov

Recipients	**Action**	**Date & Time**
Joseph Bauer (INTERNET:Joseph.Bauer@exeloncorp.c	Transferred	03/05/01 03:33PM

Post Office	**Delivered**	**Route**
Joseph INTERNET:exeloncorp.com		

Files	**Size**	**Date & Time**
Byron RI-ISI RAIs Final.wpd	40344	03/05/01 02:54PM
MESSAGE	743	03/05/01 03:32PM

Options

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BYRON UNITS 1 AND 2 RI-ISI

1. In accordance with the guidance provided in Regulatory Guides (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 butt 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 butt welds is needed for the staff to find that an acceptable level of defense-in-depth is being provided. The staff has therefore concluded that RI-ISI submittals will not be approved unless this requirement is met. Please clarify numbers of total Category B-F and B-J butt welds performing volumetric inspection and numbers of those butt welds in each category included in the RI-ISI program to ensure that a minimum of 10% stated above is met.

2. Please provide the following information for both units:

   a. When does the current 10-year ISI interval start and end?
   b. When does the current ISI period start and end?
   c. What cumulative percentage of inspections have been completed for the current interval?
   d. When will the next refueling outage start?

3. It is the NRC's position that the RI-ISI program should be consistent with the requirements of the ASME Code, Section XI on the ISI period and interval start and end dates, and the minimum percentage of examination to be completed at the end of each ISI period. Please describe the implementation plan for Byron Units 1 and 2 with respect to the above discussion.

4. For Relief Request 12R-40:

   a. On page 1 of Attachment 1, item c pertains to all dissimilar metal welds for Category B-J. This note should also indicate that these dissimilar metal welds include those not covered by Category B-F as indicated in Note c of ASME Code, Section XI, Table 2500-1 for Category B-J.

   b. On page 2 of Attachment 1, the licensee discusses Table IWC 2500-1 requirements for Category C-F-1. However, similar discussions for Category C-F-2 are missing, please explain.

5. As discussed in Section 3.2.3 of the NRC Safety Evaluation Report (SER) related to EPRI TR-112657 Rev. B-A dated October 28, 1999, a pipe segment susceptible to a degradation other than FAC and which also has the potential for water hammer should receive a high pipe failure potential. The licensee has not identified water hammer as a potential degradation

mechanism for selected pipe segments. Please clarify if any of the selected system welds are susceptible to water hammer and any other aging mechanism other than FAC.

6. Is there any recognizable plant experience regarding piping failures at either Byron Unit?

7. Table 1 of Attachment 2 identifies the reactor coolant (RC) system as one of the systems for RI-ISI implementation.

   a. Footnote 2 of Table 1 clarifies that pressurizer relief piping was included. Are thermowells also included with this system?

   b. Tables 2 through 6 provide the failure potential assessment summary and number of welds and inspections per risk category. Please verify that the pressurizer piping is included under the RC system and the steam generator (SG) piping under the main steam (MS) system for these tables.

8. ASME Code Case N-578 guidelines specify that for those welds not being inspected in the existing plant FAC and IGSCC inspection programs, the number of locations to be volumetrically examined as part of the RI-ISI program is as follows: For piping segments that are in Risk Categories 1, 2, or 3 (i.e., High risk), the number of inspection locations in each risk category should be 25% of the total number of elements in each risk category. For Risk Categories 4 and 5 (i.e., Medium risk), the number of inspection locations in each category should be 10% of the total number of elements in each risk category. Volumetric examinations are not required for those segments determined to be in Risk Categories 6 or 7 (i.e., Low risk). As referred to in Section 3.5 on page 6 of the submittal and in accordance with EPRI TR-112657 Rev. B-A, “Inspection locations are generally selected on a system-by-system basis, so that each system with ‘High’ risk category elements will have approximately 25% of the system’s ‘High’ risk elements selected for inspection and similarly 10% of the elements in systems having ‘Medium’ risk category welds will be inspected.”

   a. Table 3 identifies 160 Risk Category 3 elements for the feedwater (FW) system for Unit 1. However, Table 5 states that only 32 inspections (20%) are to be performed under the RI-ISI program. This number of inspections is less than the 25% required by the code case, please explain.

   b. Table 4 identifies 274 Risk Category 3 elements for the FW system for Unit 2. However, Table 6 states that only 61 inspections (22.2%) are to be performed under the RI-ISI program. This number of inspections is less than the 25% required by the code case, please explain.

   c. As per the note for Table 6, Table 5 provides information for Unit 1, not Unit 2 as stated. Please revise.

   d. The licensee has identified the service water (SX) system as a system to be included in the RI-ISI program (Table 1). For Unit 1, 282 Category 2 elements, and for Unit 2, 293 elements have been identified. As discussed in Section 2.3 (Augmented Programs) and the footnote to Table 5, SX inspections will be in accordance with the Service Water Integrity Program (GL 89-13) and have not been subsumed into the RI-ISI program and will remain unaffected. Please provide additional information on this program to insure that the inspections currently performed on this system meet the minimum requirements of the RI-ISI program.

e. Tables 3 and 4 identify 8 Category 3 (i.e., High risk) elements for the MS system for Units 1 and 2. However, no corresponding inspections are indicated for this category on Tables 5 and 6. Please explain why 25% of these welds are not inspected as required by the code case.

9. Please clarify the examination methods which will be used for Class 1 and Class 2 socket welds under the RI-ISI program, and explain the basis of using these methods.

10. In Section 3.5 (Inspection Location Selection and NDE Selection), the licensee states that longitudinal welds are considered subsumed with examinations of the associated circumferential weld when the circumferential weld is selected for RI-ISI examination. This approach was approved under Code Case N-524. Longitudinal welds are discussed for Category B-J welds (Item Numbers B 9.12 and B 9.22), Category C-F-1 welds (Item Numbers C5.12, C5.22, and C 5.42) and for Category C-F-2 welds (Item Numbers C 5.52, C5.62, and C 5.82). However, these item numbers are not within the scope of proposed relief request 12R-40. The licensee also states in Section 3.6 that the reference to adopting Code Case N-524 ("Alternative Examination Requirements for Longitudinal Welds in Class 1 and 2 Piping, Section XI, Division 1") will be removed from the ISI Plan upon approval of proposed relief request 12R-40. Other than for the areas of intersection between the longitudinal and circumferential welds (i.e., Code Case N-524), it is unclear what other longitudinal welds are covered under this relief request. Please clarify, and discuss how this case will be covered with the deletion of the reference to this code case.

11. Please provide a reference to the version of the PRA used to support this RI-ISI program submittal. Please also provide the core damage frequency (CDF) and the large early release frequency (LERF) estimates from the PRA version used to support this RI-ISI submittal.

12. Section 2.4 on page 4 of the submittal 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 significance categorization and, if so, how?

13. Section 2.3 on page 4 of the submittal addresses the augmented programs and states that the service water integrity program (SWIP), FAC, and HELB augmented programs were not subsumed into the RI-ISI program and remain unaffected. It further states that, "If no other damage mechanism was identified, the element was removed from the RISI element selection population and retained in the appropriate augmented inspection program." Does "...removed from the RISI 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 ASME Section XI inspections within the segment will not be included in the change in risk calculations? If not, please explain what this phrase means.

14. A comparison of the number of segments for Byron Units 1 and 2 for the systems identified in Table 1 indicate that 2 systems have different numbers of segments. The chemical volume and control (CV) system has 2 additional segments identified for Unit 2 as compared to Unit

1 and the RC system has 9 additional segments for Unit 1 as compared to Unit 2. Further, in the note to Table 4 regarding Byron Unit 2, it is stated that the difference in the distribution of welds in the different risk categories is due primarily to the SG replacement project at Byron Unit 1, which has not occurred at Byron Unit 2. Additionally, for some systems the total number of welds in the systems vary considerably between the two units. For example, Byron Unit 1 has 114 less FW welds and 23 less residual heat removal (RH) welds than Byron Unit 2, but 42 more RC welds. Please explain how the replacement of the SGs could cause such a large reduction in the number of Category 3 FW welds (by 114) and increase in the number of Category 4 RC welds (by 27) at Byron Unit 1, as compared to Byron Unit 2. Also, do the differences in the number of system segments and welds reflect actual physical differences between the piping systems in the two units?

15. In Section 3.7 on page 10 of the submittal discusses a "separate Markov calculation" for the change in LERF for lines connected to the RC system 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 loss of coolant accident (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 Rev. B-A 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 conditional large early release probability (CLERP). If the methodology used deviates from the EPRI TR-112657 Rev. B-A method for unisolatable LOCAs, please provide a comparison of the method used with the accepted method.

16. If the calculations are performed using the data from the Tables in EPRI TR-111880, instead of the updated failure rates cited in Section 3.7 on pages 9 and 10 and Table 7 of the submittal, identifying the EPRI TR-111880 Tables that are used may be provided instead of responding to the question on the Bayesian update (RAI 17). If the results from the bounding evaluations described in EPRI TR-112657 Rev. B-A, instead of the Markov calculations, are sufficient to illustrate that the suggested change in risk guidelines are not exceeded, a brief description of these bounding evaluations and results can be provided instead of responding to the questions on the Markov calculations (i.e., RAIs 18 and 19).

17. In Section 3.7 on pages 9 and 10 of the submittal, reference is made to the use of updated failure rates and rupture frequencies. The EPRI TR-111880 was completed in September 1999. A copy of that topical 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 and approval of EPRI TR-112657 Rev. B-A. EPRI TR-111880 contains tables of vendor and system-specific parameter values to be used to support RI-ISI applications. The evaluation documented in EPRI 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 experience, 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 that account for service data in Westinghouse reactors. The submittal states that the updated parameters include over 2000 reactor years of experience and reflect estimates of weld population exposure for Westinghouse Class 1 and 2 systems that were not available when EPRI TR-111880 was developed. Examination of the new parameters in the submittal reveals a difference in the grouping of the systems and large differences in the parameter values. These differences

do not appear consistent with an increase of 10% in the years of experience for rare events such as failures and ruptures. The differences appear to indicate differences in the judgements used in 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 your submittal.

a. Please describe how the failure rates were updated; that is, were the rates in Table A-9 of EPRI TR-111880 updated (i.e., used as the “prior” in performing the Bayesian update) or were the original calculations performed with the new data? Please provide a reference to the equation numbers in EPRI TR-111880 or EPRI 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 Table 7 of the submittal, the safety injection (SI) system and the RH system, both originally in the EPRI TR-111880 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 EPRI TR-111880 calculations.

c. The values provided in EPRI TR-111880 are based on an update of the Swedish Nuclear Power Inspectorate (SKI) database described in SKI Report 96:20 (now the EPRI ‘97 database), which is stated as including over 2000 reactor years of operating experience from US commercial nuclear power plants of which Westinghouse reactors accounted for about 905 operating years. This database includes piping and piping component failures that were reported from December 1961 through October 1995. What is the range of dates used in the Tier 2 documented update and how many additional reactor years are in the update?

d. The submittal states that the updated failure parameters reflect estimates of “weld population exposure” that were not available when EPRI 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 were 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, FWC, and ST) used the existing values from EPRI TR-111880.

f. It is noted that the probability of rupture given a failure (P(R|F)) has been changed, by almost a factor of 6 reduction for CV system design and construction defects and by almost a factor of 5 reduction for RC, SI, and RH system elements susceptible to stress corrosion cracking. This change implies that there has been additional data collected on the number of observed ruptures and flaws (failures). 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. There is a great amount of variation in many of the values provided in Table 7 of the submittal, when compared with the values in EPRI TR-111880 Table A-9. The greatest relative increase in rupture frequency from the EPRI TR-111880 values is nearly a factor of 60 for RH system elements susceptible to erosion-cavitation. The greatest reduction in rupture frequency from the EPRI TR-111880 values is over a factor of 70 for CV system design and construction defects. Table 2 (page 16) indicates that there is no erosion-cavitation degradation mechanism for any of the identified systems at either Byron unit. However, a large Bayesian update change would indicate that there is experience with erosion-cavitation in the RH system. This appears to be inconsistent with there being no erosion-cavitation identified in the Byron RHRS. A similar concern can be raised for the large reduction in CV system design and construction defects updated values and the large changes for many of the other updated values. Please explain why there is such a wide variation in the magnitude of the changes for the updated values from the EPRI TR-111880 values.

h. The Braidwood RI-ISI submittal indicates the Westinghouse reactor experience covers only 1000 reactor years, not 2000 as stated in the Byron RI-ISI submittal. Please explain this discrepancy.

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

19. It is our understanding that the Markov calculations include calculating an “inspection effectiveness factor” for use in equation 3-9 of EPRI TR-112657 Rev. B-A. Please provide the distribution of inspection effectiveness values calculated and a discussion of how these values compare with the direct use of the probability of detection estimates.

20. The SX system is included in the scope of the RI-ISI program, though the SWIP was not subsumed into the RI-ISI Program. Table 6-2 of EPRI TR-112657 Rev. B-A indicates that the SWIP may be subsumed into the RI-ISI program and addressed by the evaluation of localized corrosion that is part of the degradation assessment for RI-ISI, but at Byron it was not subsumed. How many welds are being inspected in the SX system under the current ASME Section XI program? If there are any welds in the SX system that are currently being inspected under the ASME Section XI program, what happens to these inspections under the RI-ISI program and if they are not inspected under the RI-ISI program why is the change in

risk zero? It is noted that the loss of essential service water, as an initiating event, is a major contributor to the Byron CDF and there are 282 Category 2 (i.e., High risk) welds at Byron Unit 1 and 293 Category 2 welds at Byron Unit 2. To be in Category 2 indicates that there is a degradation mechanism in these segments of piping (i.e., medium potential for pipe rupture). Since the SWIP is not subsumed into the RI-ISI program, the degradation mechanisms addressed by this program should not be considered in the risk categorization process. Are there any degradation mechanisms in these segments of piping that are not addressed by the current SWIP? If not, then these segments should be identified as having a low potential for pipe rupture and should be categorized as Category 4 (i.e., Medium risk). Even as a medium risk, the EPRI TR-112657 Rev. B-A methodology would require that 10% of these welds be inspected under the RI-ISI program. Please explain how the SX system welds are being addressed under the RI-ISI program.

21. Section 3.3 of EPRI TR-112657 Rev. B-A requires the consideration of external events (e.g., seismic events) and operation modes outside the scope of the PRA (e.g., shutdown) in the categorization of segments. Were external events and operation modes outside the scope of the PRA systematically considered and was the plant expert/review panel involved in this evaluation?