May 23, 2001

Mr. Oliver D. Kingsley, President
Exelon Nuclear
Exelon Generation Company, LLC
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION REGARDING INSERVICE INSPECTION RELIEF REQUESTS FOR BRAIDWOOD STATION, UNITS 1 AND 2, AND BYRON STATION, UNITS 1 AND 2 (TAC NOS. MB0507, MB0506, MB0567, AND MB0568)

Dear Mr. Kingsley:

By letters dated October 16, 2000, and November 17, 2000, Commonwealth Edison Company (ComEd), submitted requests for alternatives to the ASME Boiler and Pressure Vessel Code Section XI for Braidwood Station and Byron Station, respectively. The submittals specifically requested alternatives to the Section XI requirements for Class 1 and Class 2 piping welds. The requests were based on the Electric Power Research Institute (EPRI) Topical Report 112657, Rev. B-A, “Revised Risk-Informed Inservice Inspection Evaluation Procedure.” Subsequent to the date of the relief requests, ComEd was merged into Exelon Generation Company, LLC (Exelon). By letter dated February 7, 2001, Exelon informed the Nuclear Regulatory Commission (NRC) that it assumed responsibility for all pending actions that were requested by ComEd.

During our review, we have identified that additional information is needed in order to complete the review. Specific questions are in the enclosed Request for Additional Information (RAI). The questions regarding the Braidwood submittal were previously provided to your staff and were discussed in a meeting which was held on February 27, 2001. An electronic copy of the Byron questions was transmitted to Exelon on March 5, 2001. Based on discussions with your staff, we request Exelon’s response to the RAI within 60 days of receipt of this letter.

Sincerely,

/RA/

George F. Dick, Jr., Project Manager, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. STN 50-456, STN 50-457,
STN 50-454, and STN 50-455

Enclosure: As stated

cc w/encl: See next page

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Mr. Oliver D. Kingsley, President May 23, 2001
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SUBJECT: REQUEST FOR ADDITIONAL INFORMATION REGARDING INSERVICE INSPECTION RELIEF REQUESTS FOR BRAIDWOOD STATION, UNITS 1 AND 2, AND BYRON STATION, UNITS 1 AND 2 (TAC NOS. MB0507, MB0506, MB0567, AND MB0568)

Dear Mr. Kingsley:

By letters dated October 16, 2000, and November 17, 2000, Commonwealth Edison Company (ComEd), submitted requests for alternatives to the ASME Boiler and Pressure Vessel Code Section XI for Braidwood Station and Byron Station, respectively. The submittals specifically requested alternatives to the Section XI requirements for Class 1 and Class 2 piping welds. The requests were based on the Electric Power Research Institute (EPRI) Topical Report 112657, Rev. B-A, “Revised Risk-Informed Inservice Inspection Evaluation Procedure.” Subsequent to the date of the relief requests, ComEd was merged into Exelon Generation Company, LLC (Exelon). By letter dated February 7, 2001, Exelon informed the Nuclear Regulatory Commission (NRC) that it assumed responsibility for all pending actions that were requested by ComEd.

During our review, we have identified that additional information is needed in order to complete the review. Specific questions are in the enclosed Request for Additional Information (RAI). The questions regarding the Braidwood submittal were previously provided to your staff and were discussed in a meeting which was held on February 27, 2001. An electronic copy of the Byron questions was transmitted to Exelon on March 5, 2001. Based on discussions with your staff, we request Exelon’s response to the RAI within 60 days of receipt of this letter.

Sincerely,
/RA/
George F. Dick, Jr., Project Manager, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. STN 50-456, STN 50-457,
STN 50-454, and STN 50-455

Enclosure: As stated

cc w/encl: See next page

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REQUEST FOR ADDITIONAL INFORMATION RELATED TO

INSERVICE INSPECTION RELIEF REQUESTS

EXELON GENERATION COMPANY, LLC

BRAIDWOOD STATION, UNITS 1 AND 2

BYRON STATION, UNITS 1 AND 2

DOCKET NOS. STN 50-456, STN 50-457, STN 50-454 AND STN 50-455

By letters dated October 16, 2000, and November 17, 2000, Exelon Generation Company, LLC (the licensee, formerly Commonwealth Edison Company) submitted requests for alternatives to the ASME Boiler and Pressure Vessel Code Section XI for Braidwood Station and Byron Station, respectively. The submittals specifically requested alternatives to the Section XI requirements for Class 1 and Class 2 piping welds. The requests were based on the Electric Power Research Institute (EPRI) Topical Report 112657, Rev. B-A, "Revised Risk-Informed Inservice Inspection Evaluation Procedure."

During its review, the staff has identified that it needs additional information in order to complete its review. Specific questions follow. Questions regarding the Braidwood submittal were previously provided to the licensee and were discussed in a meeting which was held on February 27, 2001. An electronic copy of the Byron questions was transmitted to the licensee on March 5, 2001.

BRAIDWOOD UNITS 1 AND 2

Br.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 inservice inspection (ISI) changes is consistent with the principles of defense-in-depth. Based on the staff's experience with the review of risk informed-in service inspection (RI-ISI) submittals, the percentage of volumetric inspection of ASME Class 1 welds has ranged from about 7 percent to 12 percent. In cases where the original proposal was for less than 10 percent 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 percent 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's conclusion that a minimum of 10 percent 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 percent 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 percent is met as stated above.

Br.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 for RR I2R-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?

Br.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.

Br.4 Please provide a reference to the version of the PRA used to support the RI-ISI submittal. Please also provide the core damage frequency (CDF) and the large early release frequency (LERF) estimates from the PRA version used to support the RI-ISI submittal.

Br.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?

Br.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 percent 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 percent and 10.1 percent of Class 1 butt welded elements inspected include the population of Class 1 high energy line break (HELB) and the FAC element welds in the denominator?

Br.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 the RI-ISI program. 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?

Br.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's 1 steam generators (SGs) being replaced whereas Unit's 2 SGs has not been replaced. Please explain how the replacement of the SGs could cause such a large reduction in the number of Unit's 1 Category 3 main feedwater system (FW) (108) and Category 4 reactor coolant system (RC) (23) locations as compared to Unit 2. Additionally, the total number of welds in the systems seem 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 safety injection system (SI) welds than Unit 2. Do these differences in total welds reflect actual physical differences between the piping systems in the two units?

Br.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 conditional large early release probability (CLERP). 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.

Br.10 The EPRI methodology for development of RI-ISI programs that was approved by the staff incorporated a data base of observed pipe failures (EPRI '97), a methodology to estimate failure parameters from the data base, and the results of the application of the estimation methodology applied to the EPRI '97 data base. The estimation methodology description was submitted as EPRI TR-110161. TR-110161 also included a detailed sample application of the methodology to a specific system at a specific plant.

The failure parameter estimation methodology was applied to the EPRI ‘97 data base to estimate probabilistic pipe failure parameters for all reactor systems and types. The data base development and the failure parameter estimates were documented in the final draft of EPRI TR-111880 that was also submitted to support the EPRI RI-ISI methodology review. TR-110161 and TR-111880 were reviewed by the staff coincident with the RI-ISI methodology review. The approved EPRI RI-ISI Topical (TR-112657 Rev. B-A) references the failure parameter database in TR-111880 as the supporting parameter data base for the Markov methodology. A RI-ISI submittal in December 2000 used failure parameters from TR-111880. On request, the licensee submitted proprietary and non-proprietary versions of the final version of TR-111880, and use of the appropriate failure parameters in the submittal was accepted by the staff.

The Braidwood submittal states that, for some systems, a new set of failure parameters has been developed and used. Additional information on the development of these failure parameters was obtained from the licensee at a public meeting on February 27, 2001. The observed pipe failure data base supporting these parameters is different from that used in TR-111880. The new data base was apparently developed by revising the EPRI ‘97 data base and includes more observed failure data from additional sources, both domestic and foreign. Some of the assumptions and input parameters used in the methodology to estimate the probabilistic parameters from the observed data have also been changed from the original methodology discussed in TR-110161 and TR-111880. System groupings selected in TR-111880 to allow reasonable use of very limited data have also been changed. Finally, new failure parameters were only developed for some of the systems within the scope of the submittals, while original failure parameters from TR-111880 were used for the remaining systems. The methodology and data base changes resulted in changes to estimated failure frequencies ranging from a factor of 60 increase to a factor of 70 decrease. During the meeting on February 27, 2001, the licensee indicated that the use of the original failure parameters as opposed to the new parameters would yield results that do not meet the quantitative risk change criteria included in EPRI- TR-112657 Rev. B-A.

The staff finds that the re-evaluation of observed data and the use of new assumptions and input parameters are a substantive change to the methodology reviewed during the approval of the EPRI methodology for development of RI-ISI programs. The use of new failure parameters for some systems and not others raises issues of consistency and completeness that were not relevant in the industry wide, EPRI sponsored estimates in TR-111880. Furthermore, the magnitude of the quantitative changes in the failure parameters indicate that these changes could have a major impact on information used to judge, in part, the acceptability of the proposed change. Therefore, the use of these new failure parameters is a deviation from the approved EPRI methodology.

The staff finds that acceptance of new failure parameters for use in RI-ISI evaluations requires the submittal of a complete and integrated evaluation describing the guidance used to develop the data base, the assumptions used to develop the failure parameter estimates, and the complete set of quantitative results (e.g., a submittal of up-dated versions of TR-110161 and TR111880). Staff review of such a submittal would require significant additional resources and, given the current resources required to support the timely review of a large number of RI-ISI relief requests, would require more calendar time than planned for review of individual plant licensing actions. Therefore, the staff has determined that review of up-dated versions of TR-110161 and TR-111880 (or an equivalent) is more properly performed as a Topical Report review rather than within a routine RI-ISI relief request review. Any such Topical Report submitted should address, as a minimum, all systems of one reactor type to ensure consistent reflection of the current data base and current assumptions in all calculations supporting a RI-ISI submittal. Review resources would be optimized if the Topical Report also included all reactor types as does TR-111880. Use of new methods, data basis, and quantitative results will not be accepted without prior staff review. Please indicate if the licensee intends to modify the RI-ISI evaluation to utilize the original pipe failure parameters or if a new data base Topical report(s) will be submitted for staff review before review of the Byron RI-ISI program will be completed.

Br.11 Please provide a brief description of these evaluations and the results from the change in risk bounding evaluations described in EPRI TR-112657. 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, the licensee may choose to rely on the bounding results to support the acceptability of your proposed program and need not respond to questions 12 and 13 on the Markov calculations.

Br.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 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.

Br.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.

BYRON UNITS 1 AND 2

By.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 percent to 12 percent. In cases where the original proposal was for less than 10 percent 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 percent 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 percent 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 percent stated above is met.

By.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?

By.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.

By.4 For Relief Request I2R-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.

By.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.

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

By.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.

By.8 ASME Code Case N-578 guidelines specify that for those welds not being inspected in the existing plant FAC and intergranular stress-corrosion cracking (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 percent 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 percent 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 percent of the system’s ‘High’ risk elements selected for inspection and similarly 10 percent 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 percent) are to be performed under the RI-ISI program. This number of inspections is less than the 25 percent 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 percent) are to be performed under the RI-ISI program. This number of inspections is less than the 25 percent 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 explain or revise as needed.

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 ensure 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 main steam system (MS) for Units 1 and 2. However, no corresponding inspections are indicated for this category on Tables 5 and 6. Please explain why 25 percent of these welds are not inspected as required by the code case.

By.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.

By.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 B9.12 and B9.22), Category C-F-1 welds (Item Numbers C5.12, C5.22, and C5.42) and for Category C-F-2 welds (Item Numbers C5.52, C5.62, and C5.82). However, these item numbers are not within the scope of proposed relief request I2R-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.

By.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.

By.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?

By.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 percent 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.

By.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 result in 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?

By.15 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.

By.16 The EPRI methodology for development of RI-ISI programs that was approved by the staff incorporated a data base of observed pipe failures (EPRI '97), a methodology to estimate failure parameters from the data base, and the results of the application of the

estimation methodology applied to the EPRI ‘97 data base. The estimation methodology description was submitted as EPRI TR-110161. TR-110161 also included a detailed sample application of the methodology to a specific system at a specific plant. The failure parameter estimation methodology was applied to the EPRI ‘97 data base to estimate probabilistic pipe failure parameters for all reactor systems and types. The data base development and the failure parameter estimates were documented in the final draft of EPRI TR-111880 that was also submitted to support the EPRI RI-ISI methodology review. TR-110161 and TR-111880 were reviewed by the staff coincident with the RI-ISI methodology review. The approved EPRI RI-ISI Topical (TR-112657 Rev. B-A) references the failure parameter database in TR-111880 as the supporting parameter data base for the Markov methodology. A RI-ISI submittal in December 2000, used failure parameters from TR-111880. On request, the licensee submitted proprietary and non-proprietary versions of the final version of TR-111880, and use of the appropriate failure parameters in the submittal was accepted by the staff.

The Byron submittal states that, for some systems, a new set of failure parameters have been developed and used. Additional information on the development of these failure parameters was obtained from the licensee at a public meeting on February 27, 2001. The observed pipe failure data base supporting these parameters is different from that used in TR-111880. The new data base was apparently developed by revising the EPRI ‘97 data base and includes more observed failure data from additional sources, both domestic and foreign. Some of the assumptions and input parameters used in the methodology to estimate the probabilistic parameters from the observed data have also been changed from the original methodology discussed in TR-110161 and TR-111880. System groupings selected in TR-111880 to allow reasonable use of very limited data have also been changed. Finally, new failure parameters were only developed for some of the systems within the scope of the submittals, while original failure parameters from TR-111880 were used for the remaining systems. The methodology and data base changes resulted in changes to estimated failure frequencies ranging from a factor of 60 increase to a factor of 70 decrease. During the meeting on February 27, 2001, the licensee indicated that the use of the original failure parameters as opposed to the new parameters would yield results that do not meet the quantitative risk change criteria included in EPRI- TR-112657 Rev. B-A.

The staff finds that the re-evaluation of observed data and the use of new assumptions and input parameters are a substantive change to the methodology reviewed during the approval of the EPRI methodology for development of RI-ISI programs. The use of new failure parameters for some systems and not others raises issues of consistency and completeness that were not relevant in the industry wide, EPRI sponsored, estimates in TR-111880. Furthermore, the magnitude of the quantitative changes in the failure parameters indicate that these changes could have a major impact on information used to judge, in part, the acceptability of the proposed change. Therefore, the use of these new failure parameters is a deviation from the approved EPRI methodology.

The staff finds that acceptance of new failure parameters for use in RI-ISI evaluations requires the submittal of a complete and integrated evaluation describing the guidance

used to develop the data base, the assumptions used to develop the failure parameter estimates, and the complete set of quantitative results (e.g., a submittal of up-dated versions of TR-110161 and TR111880). Staff review of such a submittal would require significant additional resources and, given the current resources required to support the timely review of a large number of RI-ISI relief requests, would require more calendar time than planned for review of individual plant licensing actions. Therefore, the staff has determined that review of up-dated versions of TR-110161 and TR-111880 (or an equivalent) is more properly performed as a Topical report review and not within a routine RI-ISI relief request review. Any such Topical report submitted should address, as a minimum, all systems of one reactor type to ensure consistent reflection of the current data base and current assumptions in all calculations supporting a RI-ISI submittal. Review resources would be optimized if the Topical report also included all reactor types as does TR-111880. Use of new methods, data basis, and quantitative results will not be accepted without prior staff review. Please indicate how the licensee intends to modify the RI-ISI evaluation to utilize the original pipe failure parameters or if a new data base Topical report will be submitted for staff review before review of the Byron RI-ISI program will be completed.

By.17 Please provide a brief description of the evaluations and the results from the change in risk bounding evaluations described in EPRI TR-112657. 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, you may choose to rely on the bounding results to support the acceptability of your proposed program and need not respond to questions 18 and 19 on the Markov calculations.

By.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).

By.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.

By.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 percent 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.

By.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?