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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington. D.C. 20555 - 0001

Subject: Response to Request for Additional Information Regarding Risk Informed Inservice Inspection Relief Requests for Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2

Braidwood Station, Units 1 and 2
Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. STN 50-456 and STN 50-457

Byron Station, Units 1 and 2
Facility Operating License Nos. NPF-37 and NPF-66
NRC Docket Nos. STN 50-454 and STN 50-455

References:

(1) Letter from K. A. Ainger, (Exelon Generation Company, LLC) to NRC, "Response to Request for Additional Information Regarding Risk Informed Inservice Inspection Relief Requests for Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2," dated September 5, 2001

(2) Letter from G. F. Dick, Jr. (U.S. NRC) to O. D. Kingsley (Exelon Generation Company, LLC), "Request for Additional Information Regarding Inservice Inspection Relief Requests for Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2," dated May 23, 2001

(3) Letter from T. J. Tulon (Commonwealth Edison Company) to U.S. NRC, "Braidwood Station Interval 2 Inservice Inspection Program: Relief Request I2R-39, Alternative to the ASME Boiler and Pressure Vessel Code, Section XI, Requirements for Class 1 and Class 2 Piping Welds", dated October 16, 2000

(4) Letter from William Levis (Commonwealth Edison Company) to U.S. NRC, "Byron Station Interval 2 Inservice Inspection Program, Relief Request I2R-40, Alternative to the ASME Boiler and Pressure Vessel Code, Section XI, Requirements for Class 1 and Class 2 Piping Welds," dated November 17, 2000

A047

In References 3 and 4, Commonwealth Edison Company, now Exelon Generation Company (EGC), LLC, requested approval of an alternative to the existing 1989 edition of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," requirements for the selection and examination of Class 1 and 2 piping welds. This alternative utilizes the "risk-informed" inservice inspection program methodology discussed in Electric Power Research Institute (EPRI) Topical Report (TR) 112657, "Revised Risk-Informed Inservice Inspection Evaluation Procedure," Revision B-A, December 1999.

In Reference 2, the NRC requested additional information regarding our Reference 3 and 4 submittals. We responded to this request in Reference 1. After the NRC reviewed the information provided in Reference 1, a telephone conference call was held on September 19, 2001 between members of the NRC and EGC staffs to further discuss the method of addressing the risk associated with structural elements affected by flow accelerated corrosion (FAC) and microbiologically influenced corrosion (MIC). Attachments A and B provide our revised responses to the Reference 1 questions associated with this issue.

As previously noted, we anticipate implementing the "risk-informed" inservice inspection program methodology during the Byron Station, 2002, Spring refueling outage scheduled to begin on March 9, 2002; therefore, we request that the NRC review and approve the use of this methodology by March 1, 2002.

Please direct any questions you may have regarding this submittal to Mr. J. A. Bauer at (630) 657-2801.

Respectfully,

K. A. Ainger

K. A. Ainger
Director – Licensing
Mid-West Regional Operating Group

Attachment A: Response to Request for Additional Information, Revised Questions 5, 6, 10 and 11, Braidwood Station, Units 1 and 2, Interval 2 Inservice Inspection Program

Attachment B: Response to Request for Additional Information, Revised Questions 12, 13, 16 and 17, Byron Station, Units 1 and 2, Interval 2 Inservice Inspection Program

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Braidwood Station
NRC Senior Resident Inspector – Byron Station

Attachment A

Response to Request for Additional Information
Revised Questions 5, 6, 10 and 11

Braidwood Station Units 1 and 2

Interval 2 Inservice Inspection Program

Relief Request I2R-39, "Alternative to the ASME Boiler and Pressure Vessel Code, Section XI, Requirements for Class 1 and Class 2 Piping Welds"

<u>Question Br. 5:</u>
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?

<u>Braidwood Response to Question Br. 5:</u>
<u>How was this synergy reflected in the risk impact?</u>
For segments with two or more Inservice Inspection (ISI) amenable damage mechanisms, the associated failure rates and rupture frequencies for these and design and construction errors are summed, with the exception that microbiologically influenced corrosion (MIC) and flow accelerated corrosion (FAC) contributions are not added if the weld is part of the associated augmented inspection program for MIC or FAC. These contributions were not added as the associated augmented inspection programs will not change. Only those damage mechanisms whose inspection programs are changed in the Risk-Informed ISI (i.e., RISI) program were included. However, when there are two or more damage mechanisms, including MIC or FAC, the failure rates and rupture frequencies for the applicable ISI amenable damage mechanisms are increased by a factor of three to consider the possible effects of synergy, i.e., to consider the potential that through wall cracks would occur more quickly when two or more mechanisms were present at the same location. Design and construction errors are not considered a separate damage mechanism for the purpose of determining whether or not the synergy factor will be applied.

The above treatment was made because the service data upon which the Electric Power Research Institute (EPRI) methodology for damage mechanism assessment was based does not explicitly address multiple damage mechanisms. The following examples serve to better explain the procedure that was followed.

If a segment was found to be susceptible to both thermal fatigue (i.e., Thermal Transient (TT) and/or Thermal Stratification Cycling and Striping (TASCS)) and FAC; and FAC was not covered in the augmented program for FAC (i.e., a hypothetical case), the failure rates for design and construction errors, thermal fatigue, and FAC from EPRI Topical Report TR-111880, "Piping System Failure Rates and Replacement Frequencies for use in Risk Informed Inservice Inspection Applications," would be summed; then this result would be multiplied by a factor of three for synergy. The rupture frequencies would be determined in the same way. However, if the segment was found susceptible to the same damage mechanisms and FAC was covered in the augmented FAC program, the FAC contribution would not be included in the failure rate or rupture frequency, but its synergy effects would be included by application of the factor of three to the sum of the failure rate or rupture frequency for the design and construction errors and thermal fatigue damage mechanism.

<u>Was synergy also reflected in the safety significant categorization and if so how?</u>
As explained above, the potential for synergy was considered using engineering judgment in the delta risk evaluation and the assignment of failure potential categories in the application of the EPRI RISI risk matrix was not changed as a result of this consideration of synergy. This judgment was based on insights developed by our contractors in estimating failure rates and rupture frequencies for many different damage mechanisms and system categories in preparation of EPRI Topical Report TR-111880. Therefore, if a location was susceptible to two or more ISI amenable damage mechanisms other than FAC, the failure potential category was not increased from medium to high due to consideration of synergy. The judgment of our contractor team was that a factor of three increase in rupture frequency

would provide a conservative upper bound on the possible effects of synergy. The assumption in the risk classification matrix in the EPRI methodology was that the difference in frequency between medium and high failure potential was more than an order of magnitude. In summary, our approach to treatment of synergy effects from two or more damage mechanisms was thought to be both reasonable and beyond the guidance set forth in Regulatory Guide (RG) 1.174, "An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant Specific Changes to the Current Licensing Basis," RG 1.178, "An Approach for Plant Specific, Risk-Informed Decision Making: Inservice Inspection of Piping," and the EPRI RI-ISI Topical Report.

Question 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% 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?

Braidwood Response to Question Br. 6:

Welds identified as having FAC as the only degradation mechanism are removed from the RISI population for element selection and the percentages for selecting high and medium risk welds are not applied to the FAC-only welds. FAC-only welds currently inspected under American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," will not be selected for inspection under the RISI program, but will continue to be addressed by the FAC program. The FAC-only welds that are not selected for the RISI program are all included in the delta risk calculations. Those examinations eliminated at any of these welds would result in a slight change in risk for those specific welds and contribute to the overall delta risk that was quantified for the system.

The reported percentages of Class 1 butt-welded elements inspected does not include the population of High Energy Line Break (HELB) and the FAC element welds in the denominator as all lines in the HELB and FAC programs are classified as ASME Class 2, ASME Class 3, or non-class.

Question 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 database 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 database 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 database supporting these parameters is different from that used in TR-111880. The new database was apparently developed by revising the EPRI '97 database 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.

Braidwood Response to Question Br. 10:

This question raises several issues with the treatment of failure rates and rupture frequencies in the Braidwood RISI evaluations that bear on the acceptability of the element selections that were made in implementing the EPRI RISI methodology.

The NRC position reflected in this question is that since the failure rates from EPRI TR-111880 were not used for all systems, the treatment of failure rates represents a departure from the "Standard EPRI method" and hence additional time would be required to complete a review of updated failure rates. The updated failure rates and rupture frequencies in question were used for the reactor coolant system (RCS), safety injection (SI) system, chemical and volume control system (CVCS), and residual heat removal (RHR) system which capture most of the segments in which elements were removed and fully encompass the segments with significant Conditional Core Damage Probability (CCDP) values.

After review of this question, we have elected to amend our Relief Request to base the Risk Impact Evaluations on the EPRI Pipe Ruptures Frequencies provided in EPRI TR-111880. When these frequencies were applied to the RCS, the delta core damage frequency (i.e., ΔCDF) calculations failed to meet the system level success criterion of 1E-7/year. As a result, additional inspections were added to the Braidwood Station RISI program. These additional inspections are identified in Tables Br-10-A and Br-10-B.

The revised element selection was made with the goal of providing a 10% margin below the system level success criterion. The ΔCDF and delta large early release frequency (i.e., ΔLERF) calculations using the revised element selection, the EPRI TR-111880 pipe failure frequencies and the Markov Calculations[1] are provided in Tables BR-10-C and Br-10-D.

Table RAI Br-10-A: Impact of Revised ISI Element Selection and Failure Rate Assumptions on RCS Delta CDF Results at Braidwood Units 1 and 2

REACTOR UNIT	ISI ELEMENT SELECTION	ASSUMED FAILURE RATES	EPRI RISK CATEGORY			TOTAL EXAMS	EXAMS ADDED TO REDUCE RISK
			HIGH	MEDIUM	LOW		
Braidwood 1	Current Section XI	N/A	117	122	0	239	-
	RISI per Relief Request	Revised per Relief Request	49	54	0	103	0
	Revised RISI Selection	EPRI TR 111880	89	54	0	143	+40
Braidwood 2	Current Section XI	N/A	87	113	5	205	-
	RISI per Relief Request	Revised per Relief Request	50	56	0	106	0
	Revised RISI Selection	EPRI TR 111880	91	56	0	147	+41

1 See the response to question 11 for a discussion of the differences between the bounding and Markov calculations.

Table Br-10-B: Revised Element Selection for Braidwood RCS

BRAIDWOOD UNIT 1			BRAIDWOOD UNIT 2		
WELD ID	ADD EXAM	DELETE EXAM	WELD ID	ADD EXAM	DELETE EXAM
1RC-16-01(1)	X		2PZR-01-SE-05(1)	X	
1PZR-01-SE-02(1)	X		2PZR-01-SE-02(1)	X	
1RC-32-07(1)	X		2PZR-01-SE-03(1)	X	
1PZR-01-SE-04(1)	X		2PZR-01-SE-04(1)	X	
1RC-32-13(1)	X		2PZR-01-SE-06(1)	X	
1PZR-01-SE-06(1)	X		2RC-36-06	X	
1RC-35-01(1)	X		2RC-36-07	X	
1RC-36-09	X		2RC-36-08.01	X	
1RC-36-06	X		2RC-36-09	X	
1RC-36-08	X		2RC-31-12.01	X	
1RC-29-01-04	X		2RC-42-08	X	
1RC-29-06-04	X		2RC-42-09	X	
1RC-31-04	X		2RC-37-01	X	
1RC-31-05	X		2RC-37-02	X	
1RC-31-06	X		2RC-37-03	X	
1RC-37-03	X		2RC-37-04	X	
1RC-37-04	X		2RC-37-05	X	
1RC-37-06	X		2RC-37-06	X	
1RC-37-08	X		2RC-37-07	X	
1RC-29-01-03	X		2RC-37-07A.01	X	
1RC-29-02-03	X		2RC-37-07B.01	X	
1RC-29-03-03	X		2RC-37-07C.01	X	
1RC-29-04-03	X		2RC-37-08	X	
1RC-29-05-03	X		2RC-37-09	X	
1RC-29-06-03	X		2RC-37-10	X	
1RC-42-02	X		2RC-37-11	X	
1RC-42-03	X		2RC-41-03	X	
1RC-42-04	X		2RC-41-04	X	
1RC-42-06	X		2RC-41-05	X	
1RC-42-08	X		2RC-41-06	X	
1RC-41-01AA	X		2RC-41-07	X	
1RC-41-02AA	X		2RC-41-08	X	
1RC-41-03AA	X		2RC-41-11	X	
1RC-41-04AA	X		2RC-41-12	X	
1RC-41-05AA	X		2RC-41-13	X	
1RC-41-06AA	X		2RC-29-11	X	
1RC-41-01AB	X		2RC-29-12	X	
1RC-41-02AB	X		2RC-29-13	X	
1RC-41-03AB	X		2RC-29-14	X	
1RC-41-04AB	X		2RC-29-15	X	
			2RC-29-16	X	
(1) Butt weld					

Table RAI Br-10-C: Revised Risk Impact Results for Braidwood Unit 1

BRAIDWOOD 1 RISK IMPACT REPORT*		
SYSTEM	DELTA CDF MARKOV MODEL	DELTA LERF MARKOV MODEL
CVCS	3E-11	3E-12
CS	2E-09	9E-11
FW	-6E-09	-7E-10
MS	8E-11	1E-11
RCS	9E-08	2E-09
RHR	2E-09	2E-09
SI	6E-10	6E-10
SX	4E-09	2E-10
TOTAL	9E-08	4E-09

* Positive values indicate a risk increase while negative values denote a risk decrease

Table RAI Br-10-D: Revised Risk Impact Results for Braidwood Unit 2

BRAIDWOOD 2 RISK IMPACT REPORT*		
SYSTEM	DELTA CDF MARKOV MODEL	DELTA LERF MARKOV MODEL
CVCS	-3E-09	-3E-10
CS	2E-09	1E-10
FW	-1E-08	-1E-09
MS	9E-11	1E-11
RCS	8E-08	2E-09
RHR	4E-09	4E-09
SI	-5E-08	-5E-08
SX	4E-09	2E-10
TOTAL	2E-08	-5E-08

* Positive values indicate a risk increase while negative values denote a risk decrease

Question 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 chose 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.

Braidwood Response to Question Br. 11:
A simplified and conservative risk impact calculation, not using the Markov model calculation of pipe break frequency, was performed for Braidwood Station Units 1 and 2. This calculation was performed using the same approach as was implemented for a previously approved relief request for South Texas Project. The change in risk for a particular system was calculated using the following:

$$\Delta CDF_j = \sum_i [FR_{i,j} * (SXI_{i,j} - RISI_{i,j}) * CCDP_{i,j}] \qquad (1)$$

where

ΔCDF_j = Change in CDF for system j
$FR_{i,j}$ = Rupture frequency per element for risk segment i of system j
$SXI_{i,j}$ = Number of Section XI inspection elements for risk segment i of system j
$RISI_{i,j}$ = Number of RISI inspection elements for risk segment i of system j
$CCDP_{i,j}$ = Conditional core damage probability given a break in risk segment i of system j

The total change in risk for all systems within the RISI evaluation scope is calculated by summing the changes in risk for each individual system, as follows:

$$\Delta CDF_{TOTAL} = \sum_j \Delta CDF_j \qquad (2)$$

The ΔLERF for each system was calculated as the product of the ΔCDF, and a factor equivalent to the ratio of the conditional large early release probability (CLERP) to the CCDP selected for each system. In addition, the ΔLERF from unisolable loss of coolant accidents (LOCAs) outside containment was added for those systems with piping segments subject to this phenomenon (i.e., SI and RHR). The CLERP/CCDP ratio was chosen for each system as the ratio for the limiting segment for the system. Application of the limiting CLERP/CCDP ratio across all segments of the system results in conservative system ΔLERF calculations. The total change in LERF for all systems within the RISI evaluation scope is calculated by summing the ΔLERF for each individual system.

Using this method to calculate the change in risk requires making several assumptions. Those assumptions are as follows:

- Inspections are 100% successful at finding flaws and preventing ruptures.
- Increased probability of detection (POD) due to inspection for cause is not credited.
- Pipe failure rates and rupture frequencies are constant, not age dependent.

The results of the Braidwood Station Unit 1 risk impact calculation are shown in Table Br 11-A. Using the bounding analysis, the EPRI Pipe Failure Frequencies, and including all of the additional welds that were added in response to Question 10, only the RCS system exceeded the change in CDF criterion of 1.0E-07 per system per year. The total change in CDF was 3E-07, well below the criterion of risk significance from Regulatory Guide 1.174 of 1E-06 for all systems. Similarly, the change in LERF values

were all well below the criterion of 1E-08 per system per year. The total change in LERF was 9E-9, well below the criterion of risk significance from Regulatory Guide 1.174 of 1.0E-07 for all systems.

The results of the Braidwood Station Unit 2 risk impact calculation are shown in Table Br 11-B. Using the bounding analysis, the EPRI Pipe Failure Frequencies, and including the additional welds that were added in response to Question 10, only the RCS system exceeded the CDF criterion of 1E-07 per system per year. The total change in CDF was 2E-07, well below the criterion of 1E-06 for all systems. Similarly, the change in LERF values were all well below the criterion of 1E-08 per system. The total change in LERF was -8E-8, i.e., a decrease in risk associated with LERF.

As the results of the bounding analysis did not meet the system level success criterion for the RCS system, the Markov modeling approach was applied. Using the Markov model, the details of which are discussed in response to Questions 12 and 13 in our September 5, 2001 submittal, all of the systems meet the system level success criterion. A comparison of the results of the bounding analysis versus the Markov analysis is provided in Table Br-11-A for Braidwood Station Unit 1 and Br-11-B for Braidwood Station Unit 2.

Table Br-11-A: Comparison of Risk Impact Results for Braidwood Unit 1

Braidwood 1 Risk Impact Report*				
System	Delta CDF		Delta LERF	
	Bounding	Markov Model	Bounding	Markov Model
CVCS	1E-10	3E-11	1E-11	3E-12
CS	3E-09	2E-09	2E-10	9E-11
FW	-7E-09	-6E-09	-9E-10	-7E-10
MS	1E-10	8E-11	2E-11	1E-11
RCS	3E-07	9E-08	6E-09	2E-09
RHR	3E-09	2E-09	3E-09	2E-09
SI	1E-09	6E-10	1E-09	6E-10
SX	8E-09	4E-09	4E-10	2E-10
TOTAL	3E-07	9E-08	9E-09	4E-09

* Positive values indicate a risk increase while negative values denote a risk decrease

Table Br-11-B: Comparison of Risk Impact Results for Braidwood Unit 2

BRAIDWOOD 2 RISK IMPACT REPORT*				
SYSTEM	DELTA CDF		DELTA LERF	
	BOUNDING	MARKOV MODEL	BOUNDING	MARKOV MODEL
CVCS	-6E-09	-3E-09	-5E-10	-3E-10
CS	4E-09	2E-09	2E-10	1E-10
FW	-2E-08	-1E-08	-2E-09	-1E-09
MS	2E-10	9E-11	2E-11	1E-11
RCS	3E-07	8E-08	5E-09	2E-09
RHR	7E-09	4E-09	7E-09	4E-09
SI	-9E-08	-5E-08	-9E-08	-5E-08
SX	8E-09	4E-09	4E-10	2E-10
TOTAL	2E-07	2E-08	-8E-08	-5E-08

* Positive values indicate a risk increase while negative values denote a risk decrease

Attachment B

Response to Request for Additional Information
Revised Questions 12, 13, 16 and 17

Byron Station Units 1 and 2

Interval 2 Inservice Inspection Program

Relief Request I2R-40, "Alternative to the ASME Boiler and Pressure Vessel Code, Section XI, Requirements for Class 1 and Class 2 Piping Welds"

<u>Question By.12:</u>
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?

<u>Byron Response to Question By.12:</u>
<u>How was this synergy reflected in the risk impact?</u>
For segments with two or more Inservice Inspection (ISI) amenable damage mechanisms, the associated failure rates and rupture frequencies for these and design and construction errors are summed, with the exception that microbiologically influenced corrosion (MIC) and flow accelerated corrosion (FAC) contributions are not added if the weld is part of the associated augmented inspection program for MIC or FAC. These contributions were not added as the associated augmented inspection programs will not change. Only those damage mechanisms whose inspection programs are changed in the Risk-Informed ISI (i.e., RISI) program were included. However, when there are two or more damage mechanisms, including MIC or FAC, the failure rates and rupture frequencies for the applicable ISI amenable damage mechanisms are increased by a factor of three to consider the possible effects of synergy, i.e., to consider the potential that through wall cracks would occur more quickly when two or more mechanisms were present at the same location. Design and construction errors are not considered a separate damage mechanism for the purpose of determining whether or not the synergy factor will be applied.

The above treatment was made because the service data upon which the Electric Power Research Institute (EPRI) methodology for damage mechanism assessment was based does not explicitly address multiple damage mechanisms. The following examples serve to better explain the procedure that was followed.

If a segment was found to be susceptible to both thermal fatigue (i.e., Thermal Transient (TT) and/or Thermal Stratification Cycling and Striping (TASCS)) and FAC; and FAC was not covered in the augmented program for FAC (i.e., a hypothetical case), the failure rates for design and construction errors, thermal fatigue, and FAC from EPRI Topical Report TR-111880, "Piping System Failure Rates and Replacement Frequencies for use in Risk Informed Inservice Inspection Applications," would be summed; then this result would be multiplied by a factor of three for synergy. The rupture frequencies would be determined in the same way. However, if the segment was found susceptible to the same damage mechanisms and FAC was covered in the augmented FAC program, the FAC contribution would not be included in the failure rate or rupture frequency, but its synergy effects would be included by application of the factor of three to the sum of the failure rate or rupture frequency for the design and construction errors and thermal fatigue damage mechanism.

<u>Was synergy also reflected in the safety significant categorization and if so how?</u>
As explained above, the potential for synergy was considered using engineering judgment in the delta risk evaluation and the assignment of failure potential categories in the application of the EPRI RISI risk matrix was not changed as a result of this consideration of synergy. This judgment was based on insights developed by our contractors in estimating failure rates and rupture frequencies for many different damage mechanisms and system categories in preparation of EPRI Topical Report TR-111880. Therefore, if a location was susceptible to two or more ISI amenable damage mechanisms other than FAC, the failure potential category was not increased from medium to high due to consideration of synergy. The judgment of our contractor team was that a factor of three increase in rupture frequency

would provide a conservative upper bound on the possible effects of synergy. The assumption in the risk classification matrix in the EPRI methodology was that the difference in frequency between medium and high failure potential was more than an order of magnitude. In summary, our approach to treatment of synergy effects from two or more damage mechanisms was thought to be both reasonable and beyond the guidance set forth in Regulatory Guide (RG) 1.174, "An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant Specific Changes to the Current Licensing Basis," RG 1.178, "An Approach for Plant Specific, Risk-Informed Decision Making: Inservice Inspection of Piping," and the EPRI RI-ISI Topical Report.

Question 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% 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.

Byron Response to Question By.13:

Welds identified as having FAC as the only degradation mechanism are removed from the RISI population for element selection and the percentages for selecting high and medium risk welds are not applied to the FAC-only welds. FAC-only welds currently inspected under American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," will not be selected for inspection under the RISI program, but will continue to be addressed by the FAC program. The FAC-only welds that are not selected for the RISI program are all included in the delta risk calculations. Those examinations eliminated at any of these welds would result in a slight change in risk for those specific welds and contribute to the overall delta risk that was quantified for the system.

Question 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 database 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 request, 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 database Topical report will be submitted for staff review before review of the Byron RI-ISI program will be completed.

Byron Response to Question By.16:
This question raises several issues with the treatment of failure rates and rupture frequencies in the Byron RISI evaluations that bear on the acceptability of the element selections that were made in implementing the EPRI RISI methodology.

The NRC position reflected in this question is that since the failure rates from EPRI TR-111880 were not used for all systems, the treatment of failure rates represents a departure from the "Standard EPRI method" and hence additional time would be required to complete a review of updated failure rates. The updated failure rates and rupture frequencies in question were used for the reactor coolant system (RCS), safety injection (SI) system, chemical and volume control system (CVCS), and residual heat removal (RHR) system which capture most of the segments in which elements were removed and fully encompass the segments with significant Conditional Core Damage Probability (CCDP) values.

After review of this question, we have elected to amend our Relief Request to base the Risk Impact Evaluations on the EPRI Pipe Ruptures Frequencies provided in EPRI TR-111880. When these frequencies were applied to the RCS, the delta core damage frequency (i.e., ΔCDF) calculations failed to meet the system level success criterion of 1E-7/year. As a result, additional inspections were added to the Byron Station RISI program. These additional inspections are identified in Tables By-16-A and By-16-B.

The revised element selection was made with the goal of providing a 10% margin below the system level success criterion. The ΔCDF and delta large early release frequency (i.e., ΔLERF) calculations using the revised element selection, the EPRI TR-111880 pipe failure frequencies and the Markov Calculations[1] are provided in Tables By-16-C and By-16-D.

Table RAI By-16-A: Impact of Revised ISI Element Selection and Failure Rate Assumptions on RCS Delta CDF Results at Byron Units 1 and 2

REACTOR UNIT	ISI ELEMENT SELECTION	ASSUMED FAILURE RATES	EPRI RISK CATEGORY			TOTAL EXAMS	EXAMS ADDED TO REDUCE RISK
			HIGH	MEDIUM	LOW		
Byron 1	Current Section XI	N/A	77	115	2	194	-
	RISI per Relief Request	Revised per Relief Request	53	49	0	102	0
	Revised RISI Selection	EPRI TR 111880	68	52	0	120	+18
Byron 2	Current Section XI	N/A	69	108	0	177	-
	RISI per Relief Request	Revised per Relief Request	51	48	0	99	0
	Revised RISI Selection	EPRI TR 111880	62	58	0	120	+21

1 See the response to question 11 for a discussion of the differences between the bounding and Markov calculations.

Attachment B
Response to Request for Additional Information
Byron Units 1 and 2

Table By-16-B: Revised Element Selection for Byron RCS

BYRON UNIT 1			BYRON UNIT 2		
WELD ID	ADD EXAM	DELETE EXAM	WELD ID	ADD EXAM	DELETE EXAM
1RY-01-S/PN-02/F2[1]	X		2RC04AA-12/J04[1]		X
1RY-01-S/PN-04/F4[1]	X		2RY-01-S/PN-02/F2[1]	X	
1RY-01-S/PN-05/F5[1]	X		2RY03AC-6/J01[1]	X	
1RY02A-6/J01[1]	X		2RY-01-S/PN-05/F5[1]	X	
1RY03AA-6/J01[1]	X		2RY-01-S/PN-03/F3[1]	X	
1RC14AA-2/W-02	X		2RY18A-2/W-01	X	
1RC14AA-2/W-03	X		2RY18A-2/W-03	X	
1RC14AA-2/W-03A	X		2RY18A-2/W-02	X	
1RC14AA-2/W-03B	X		2RC14AA-2/W-11	X	
1RC14AA-2/W-03C	X		2RC14AA-2/W-01	X	
1RC14AA-2/W-04	X		2RC16AA-2/W-06	X	
1RC14AA-2/W-05	X		2RC16AA-2/W-03	X	
1RC14AA-2/W-10	X		2RC16AA-2/W-07	X	
1RC14AA-2/W-12	X		2RC04AA-12/J02[1]		X
1RC14AA-2/W-13	X		2RC01AA-29/J06	X	
1RC-01-BD/SE-1[1]	X		2RC-01-BA/F1[1]	X	
1RC-01-BD/SE-2[1]	X		2RC-01-BA/F2[1]	X	
1RC21BA-8/J01[1]	X		2RC03AA-27.5/J02A	X	
(1) Butt weld			2RC03AA-27.5/J04	X	
			2RC13AA-2/W-02	X	
			2RC13AA-2/W-03	X	
			2RC13AA-2/W-04	X	
			2RC13AA-2/W-05	X	
			2RC26A-2/W-01	X	
			2RC26A-2/W-02	X	

Table RAI By-16-C: Revised Risk Impact Results for Byron Unit 1

BYRON 1 RISK IMPACT REPORT*		
SYSTEM	DELTA CDF MARKOV MODEL	DELTA LERF MARKOV MODEL
CVCS	-3E-07	-3E-08
CS	-9E-10	-4E-11
FW	-8E-09	-1E-09
MS	1E-10	1E-11
RCS	9E-08	2E-09
RHR	-1E-09	-1E-09
SI	-4E-08	-4E-08
SX	4E-10	2E-11
TOTAL	-3E-07	-7E-08

* Positive values indicate a risk increase while negative values denote a risk decrease

Table RAI By-16-D: Revised Risk Impact Results for Byron Unit 2

BYRON 2 RISK IMPACT REPORT*		
SYSTEM	DELTA CDF MARKOV MODEL	DELTA LERF MARKOV MODEL
CVCS	-1E-08	-1E-09
CS	-7E-10	-4E-11
FW	-1E-08	-2E-09
MS	1E-10	1E-11
RCS	9E-08	2E-09
RHR	-8E-10	-8E-10
SI	-5E-08	-5E-08
SX	4E-10	2E-11
TOTAL	7E-09	-6E-08

* Positive values indicate a risk increase while negative values denote a risk decrease

Attachment B
Response to Request for Additional Information
Byron Station Units 1 and 2

Question By.17:
Please provide a brief description of the evaluation and the results from the change in risk bounding evaluations described in EPRI TR-112657. If 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, 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.

Byron Response to Question By. 17:
A simplified and conservative risk impact calculation, not using the Markov model calculation of pipe break frequency, was performed for Byron Station, Units 1 and 2. This calculation was performed using the same approach as was implemented for the previously approved relief request for South Texas Project. The change in risk for a particular system was calculated using the following:

$$\Delta CDF_j = \sum_i [FR_{i,j} * (SXI_{i,j} - RISI_{i,j}) * CCDP_{i,j}] \qquad (1)$$

where
ΔCDF_j = Change in CDF for system j
$FR_{i,j}$ = Rupture frequency per element for risk segment i of system j
$SXI_{i,j}$ = Number of Section XI inspection elements for risk segment i of system j
$RISI_{i,j}$ = Number of RISI inspection elements for risk segment i of system j
$CCDP_{i,j}$ = Conditional core damage probability given a break in risk segment i of system j

The total change in risk for all systems within the RISI evaluation scope is calculated by summing the changes in risk for each individual system, as follows:

$$\Delta CDF_{TOTAL} = \sum_j \Delta CDF_j \qquad (2)$$

The ΔLERF for each system was calculated as the product of the ΔCDF, and a factor equivalent to the ratio of the conditional large early release probability (CLERP) to the CCDP selected for each system. In addition, the ΔLERF from unisolable loss of coolant accidents (LOCAs) outside containment was added for those systems with piping segments subject to this phenomenon (i.e., SI and RHR). The CLERP/CCDP ratio was chosen for each system as the ratio for the limiting segment for the system. Application of the limiting CLERP/CCDP ratio across all segments of the system results in conservative system ΔLERF calculations. The total change in LERF for all systems within the RISI evaluation scope is calculated by summing the ΔLERF for each individual system.

Using this method to calculate the change in risk requires making several assumptions. Those assumptions are as follows:

- Inspections are 100% successful at finding flaws and preventing ruptures.
- Increased probability of detection (POD) due to inspection for cause is not credited.
- Pipe failure rates and rupture frequencies are constant, not age dependent.

The results of the Byron Station Unit 1 risk impact calculation are shown in Table By-17-A. Using the bounding analysis, the EPRI Pipe Failure Frequencies and including all of the welds that were added in response to Question 16, only the RCS exceeded the change in CDF criterion of 1.0E-07 per system per year. The total change in CDF was -4E-07, actually a decrease in overall risk and well below the criterion of risk significance from Regulatory Guide 1.174 of 1.0E-06 for all systems. Similarly, the

change in LERF values were all well below the criterion of 1.0E-08 per system per year. The total change in LERF was -1E-07, a decrease in risk and hence, below the criterion of risk significance from Regulatory Guide 1.174 1.0E-07 for all systems.

The results of the Byron Station Unit 2 risk impact calculation are shown in Table By-17-B. Using the bounding analysis, the EPRI Pipe Failure Frequencies and including all of the welds that were added in response to Question 16, only the RCS exceeded the change in CDF criterion of 1.0E-07 per system per year. The total change in CDF was 1E-07, well below the criterion of 1.0E-06 for all systems. Similarly, the change in LERF values were all well below the criterion of 1.0E-08 per system. The total change in LERF was -1E-07, a decrease and hence, below the criterion of 1.0E-07 per system per year for all systems.

As the results of the bounding analysis did not meet the system level success criterion for the RCS, the Markov modeling approach was applied. Using the Markov model, the details of which are discussed in response to Questions 18 and 19 in our September 5, 2001 submittal, all of the systems meet the system level success criterion. A comparison of the results of the bounding analysis versus the Markov analysis is provided in Table By-17-A for Byron Unit 1 and By-17-B for Byron Station Unit 2.

Table By-17-A: Comparison of Risk Impact Results for Byron Unit 1

BYRON 1 RISK IMPACT REPORT*				
SYSTEM	DELTA CDF		DELTA LERF	
	BOUNDING	MARKOV MODEL	BOUNDING	MARKOV MODEL
CVCS	-6E-07	-3E-07	-6E-08	-3E-08
CS	-2E-09	-9E-10	-8E-11	-4E-11
FW	-1E-08	-8E-09	-1E-09	-1E-09
MS	2E-10	1E-10	2E-11	1E-11
RCS	3E-07	9E-08	5E-09	2E-09
RHR	-2E-09	-1E-09	-2E-09	-1E-09
SI	-6E-08	-4E-08	-6E-08	-4E-08
SX	8E-10	4E-10	4E-11	2E-11
TOTAL	-4E-07	-3E-07	-1E-07	-7E-08

* Positive values indicate a risk increase while negative values denote a risk decrease

Table By-17-B: Comparison of Risk Impact Results for Byron Unit 2

BYRON 2 RISK IMPACT REPORT*				
SYSTEM	DELTA CDF		DELTA LERF	
	BOUNDING	MARKOV MODEL	BOUNDING	MARKOV MODEL
CVCS	-2E-08	-1E-08	-2E-09	-1E-09
CS	-1E-09	-7E-10	-7E-11	-4E-11
FW	-2E-08	-1E-08	-2E-09	-2E-09
MS	2E-10	1E-10	2E-11	1E-11
RCS	2E-07	9E-08	5E-09	2E-09
RHR	-1E-09	-8E-10	-1E-09	-8E-10
SI	-9E-08	-5E-08	-9E-08	-5E-08
SX	8E-10	4E-10	4E-11	2E-11
TOTAL	1E-07	7E-09	-1E-07	-6E-08

* Positive values indicate a risk increase while negative values denote a risk decrease.