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RS-01-181

September 5, 2001

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 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

(2) 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

(3) 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

In References 2 and 3, Commonwealth Edison Company, now Exelon Generation Company, 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

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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 1, the NRC requested additional information regarding our Reference 2 and 3 submittals. Attachments A, B and C to this letter provide the Braidwood Station and Byron Station responses to this request for additional information. Our response to the request for additional information was due to the NRC by July 30, 2001; however, as agreed during discussions between G. F. Dick (NRC) and J. A. Bauer (Exelon Generation Company, LLC), the due date was extended to September 5, 2001.

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

Attachments: Attachment A, Response to Request for Additional Information, Braidwood Station, Units 1 and 2, Interval 2 Inservice Inspection Program
Attachment B, Response to Request for Additional Information, Braidwood Station, Units 1 and 2, Interval 2 Inservice Inspection Program
Attachment C, Response to Request for Additional Information, Braidwood Station, Units 1 and 2, RAI Question Br. 12, and Byron Station, Units 1 and 2, RAI Question By. 18

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

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"

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

__Braidwood Response to Question Br.1:__
The revised numbers of Class 1 welds selected for volumetric examinations are listed below. This revised population is the result of a re-selection performed in the re-evaluation of the Braidwood Station Risk Informed Inservice Inspection (RI-ISI) program discussed in the response to Request for Additional Information (RAI) question Br. 10.

Table RAI-Br.1: 10% Selection Criteria for Braidwood Unit 1 and Unit 2

BRAIDWOOD CLASS 1 WELD EXAM SELECTIONS BASED ON EPRI TR-11880 DATA				
UNIT	TOTAL CLASS 1 WELDS	TOTAL CLASS 1 BUTT WELDS	NUMBER SELECTED FOR VOLUMETRIC EXAMINATION	PERCENTAGE SELECTED FOR VOLUMETRIC EXAMINATION
1	1624	773	78	10.1%
2	1605	740	80	10.8%
Note: Class 1 population consists of item numbers B5.10, B5.40, B5.70 for Category B-F and B9.11, B9.21, B9.31 and B9.32 for Category B-J. Item number B9.40 (B-J) is excluded from the butt weld counting due to component configurations.				

The totals of Class 1 welds receiving volumetric examination exceed 10% for both Braidwood Station Unit 1 and Unit 2.

*__Question Br.2:__ **Please clarify the following:***

__Question Br 2 (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.

<u>Braidwood Response to Question Br.2(a):</u>
Based on the American Society of Mechanical Engineers (ASME) Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," Section IWB-2412(b)[1] allowance, Braidwood Station had planned to extend the Period 1 for Unit 1 approximately three months from the original end date of July 28, 2001, to include the Fall 2001, Unit 1 refuel outage (A1R09). For Unit 2, Braidwood Station had planned to extend Period 1 approximately 6 months from the original end date of October 16, 2001, to include the Spring 2002, Unit 2 refuel outage (A2R09). These administrative Period 1 extensions were established as contingencies to complete the ASME Section XI Period 1 examinations if required.

<u>Question Br. 2 (b)</u>
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.

<u>Braidwood Response to Question Br.2(b):</u>
Braidwood Station schedules and examines dissimilar metal piping welds in conjunction with Category B-J welds only.

At Braidwood Station, all dissimilar metal welds are included in Category B-F. Currently, there are no piping dissimilar metal welds in the station's Class 1 piping systems. This statement was included for reference to code requirements only. The applicable code edition in use at Braidwood Station is the 1989 Edition. The statement about the "dissimilar metal welds not covered by Category B-F" is in later code editions. With the adoption of RI-ISI, the requirements of Table 2500-1, Category B-J will be superceded by Table 1, Category R-A.

<u>Question Br. 2 (c):</u>
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.

<u>Braidwood Response to Question Br.2(c):</u>
The discussion of the Table IWC 2500-1 requirements also applies to Category C-F-2. The wording of the paragraph on page 2 of 4 of Relief Request I2R-39 should read:

> "Table IWC 2500-1 requires a volumetric and surface examination for items C5.11, C5.21, and C5.51 and a surface examination for items C5.30, C5.41 C5.70, and C5.81 for those welds selected per the following: "

<u>Question Br. 2 (d):</u>
Is there any recognizable plant experience on piping failures at Braidwood?

<u>Braidwood Response to Question Br.2(d):</u>
There have been no recognizable piping failures affecting systems within the scope of the RI-ISI program at Braidwood Station. As part of the Plant Specific Service History review performed during the element selection process, a variety of plant data going back to commercial operation for both Braidwood Station

1 - Section IWA-2430 (d)(3) in the 1996 Addenda to the 1995 Edition states :
"That portion of an inspection interval described as an inspection period may be reduced or extended by as much as one year to enable an inspection to coincide with a plant outage."

Units was reviewed. The information reviewed included: work requests, Engineering Requests (ERs), Action Requests (ARs), Licensee Event Reports (LERs), Nuclear Tracking System (NTS) items, Problem Identification Form (PIFs) items, and Condition Reports (CRs). From this review two events were identified for Braidwood Station, both occurring on Unit 1. One event is associated with the Residual Heat Removal (RHR) system and was a leak in a ¾ inch pipe documented in LER 50-456/90-012. The other event is associated with a one-inch line in the Emergency Core Cooling System (ECCS) and is documented in NRC Information Notice 88-13, "Water Hammer and Possible Piping Damage Caused by Misapplication of Kerotest Packless Metal Diaphragm Globe Valves," dated April 18, 1988. Both of these events involve piping that is smaller than the lower limit of pipe sizes used for developing failure and rupture data (i.e., 1.5" Class 1 and 2" Class 2) and not within the scope of the RI-ISI evaluation and were not included in the Braidwood Station data update performed for the RI-ISI evaluation.

Question Br. 2 (e)
What is the minimum pipe diameter included in the RI-ISI evaluation and program?

Braidwood Response to Question Br.2(e):
The minimum pipe diameter in the RI-ISI evaluation and program is 1.5" for Class 1 piping and 2" for Class 2 piping.

Question Br. 2 (f):
Both Tables 5 and 6 included the Risk Category 4 in the High-Risk columns. Should these be under Medium Risk columns?

Braidwood Response to Question Br.2(f):
Yes, the High Risk heading should be formatted to include only the Category 1, 2, and 3 columns.

Question 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.

Braidwood Response to Question Br.3:
Although water hammer events have occurred previously at Braidwood Station, these events occurred in the balance of plant systems that are outside the scope of the RI-ISI program. Based on the differences in system design and operating conditions associated with these events and those of the systems within the RI-ISI scope, it is judged that water hammer is not credible when related to the Braidwood RI-ISI evaluation.

The 1997 water hammer event at Braidwood Station Unit 2 in the Feedwater (FW) System was the result of an inadequate fill and vent procedure. The waterhammer occurred during plant startup prior to the required mode of applicability for the FW System. Design changes on Unit 1 and planned design changes for Unit 2 coupled with the changes in operating procedures prevent additional water hammer events during FW System startup.

Question Br. 4
Please provide a reference to the version of the PRA used to support the risk informed- inservice inspection (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.

Braidwood Response to Question Br. 4
The PRA models used to support the RI-ISI submittal are documented in the following.

Braidwood Nuclear Station PRA CDF Calculation, BRW-99-0136-N, Rev. 0 (October 11, 1999).
Braidwood Nuclear Station PRA LERF Calculation, BRW-99-0324-N, Rev. 0 (November 2, 1999).

The Braidwood Station Unit 2 model was used for calculations to support the RI-ISI submittal, but the Braidwood Station Unit 1 and Unit 2 models are virtually identical. The base CDF for Braidwood Unit 2 from the above model is 4.86E-05/yr and the base LERF is 4.96E-06/yr.

Question 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?

Braidwood Response to Question Br. 5:
How was this synergy reflected in the risk impact?
For segments with two or more ISI amenable damage mechanisms, the associated failure rates and rupture frequencies for these and design and construction errors are summed, with the exception that Intergranular Stress Corrosion Cracking (IGSCC) and FAC contributions are not added if the weld is part of the associated augmented inspection program for IGSCC 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 RISI program were included. However, when there are two or more damage mechanisms, including IGSCC 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.

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. Two 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 corrosion cracking, and the corrosion cracking is not covered in the augmented program for IGSCC (i.e., a hypothetical case), the failure rates for design and construction errors, thermal fatigue, and stress corrosion cracking 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. But if the segment was found susceptible to the same three damage mechanisms and the stress corrosion cracking was covered in the augmented IGSCC program, the stress corrosion cracking contribution would not be included in the failure rate or rupture frequency, but its synergy effects would be included by the factor of three.

Was synergy also reflected in the safety significant categorization and if so how?
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 RI-ISI 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 Section XI will not be selected for inspection under the RI-ISI program, but will continue to be addressed by the FAC program. The FAC-only welds are listed in the delta risk calculation tables, but no change in risk is calculated for these welds when Section XI examinations are eliminated at any of these welds.

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. The HELB and FAC-only welds are removed from the RI-ISI population for element selection and no RISI inspections are selected for these welds. For Braidwood Station, all lines in the HELB and FAC programs are classified as ASME Class 2, ASME Class 3, or non-class.

<u>Question Br.7:</u>
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?

<u>Braidwood Response to Question Br. 7:</u>
All of the segments in the Essential Service Water (SX) system that were classified under ASME Section XI Category C-F-2[2] were evaluated in the RI-ISI program as Category 2, High Risk segments. There are currently 45 SX system welds selected for Section XI examination[3].

The Containment Purge system (VQ) did not have any elements of high or medium consequence rank and were therefore eliminated from element selection. Because all the VQ system piping within the Section XI boundary has a wall thickness of less than 3/8 inch, all welds[4] were exempt from Section XI examination; however, these VQ welds were included in the total weld count of Categories C-F-1 and C-F-2 to which the 7.5% sampling rate was applied per IWC-2500-1, Tables C-F-1 and C-F-2, Note 2[5].

<u>Question Br.8:</u>
In the note to Table 4 regarding Unit 2, the licensee indicates that the difference in the distribution of welds in the different risk categories is due primarily to the Unit 1's steam generators (SGs) being replaced whereas Unit 2's SGs has not been replaced. Please explain how the replacement of the SGs could cause such a large reduction in the number of Unit 1's Category 3 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 seems to vary substantially between the two units. For example, Unit 1 has 104 less FW and 27 less RC welds than Unit 2, but 65 more 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?

<u>Braidwood Response to Question Br. 8:</u>
For the Reactor Coolant System (RCS) and Safety Injection (SI) systems, the weld number differences between the units are due to differences in the as-built conditions (i.e., principally in the routing of small-bore (<4") piping). For the FW system, the Unit 1 steam generator replacement eliminated all of the auxiliary FW piping in containment. This accounts for approximately 100 fewer welds in the Unit 1 FW system compared to Unit 2.

2 - 288 SX welds in Unit 1, 292 in Unit 2.
3 - 22 SX welds in Unit 1, 23 in Unit 2.
4 - 57 VQ welds, 30 in Unit 1 and 27 in Unit 2.
5 - Note 2 reads, in part: "(Some welds not exempted by IWC-1220 are not required to be nondestructively examined per Examination Category C-F-1," or similarly for C-F-2, "These welds, however, shall be included in the total weld count to which the 7.5% sampling rate is applied.)

Question 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.

Braidwood Response to Question Br. 9:

The "separate Markov calculation" in the original submittal represents the unisolable LOCA outside containment for lines connected to the RCS. However, a simplified approach was taken by assuming the CLERP/Conditional Core Damage Probability (CCDP) ratio, for those systems susceptible to unisolable LOCA outside containment (i.e., RHR and SI), was 1.0, (i.e., ΔLERF = ΔCDF). This is further explained in the response to RAI question Br. 11.

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.

<u>Braidwood Response to Question Br. 10:</u>
This RAI 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 RI-ISI methodology.

The NRC position reflected in this RAI 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 RCS, SI, Chemical and Volume Control System (CVCS), and RHR systems which capture most of the segments in which elements were removed and fully encompass the segments with significant CCDP values.

After review of this RAI 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 system, the delta CDF calculations failed to meet the system level success criterion of 1E-7/year. As a result, additional inspections were added to the Braidwood Station RI-ISI 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 ΔLERF calculations using the revised element selection, the EPRI TR-111880 pipe failure frequencies and the Markov Calculations[6] 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

6 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 Station 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 Station 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	-4E-09	-5E-10
MS	8E-11	1E-11
RCS	9E-08	2E-09
RHR	2E-09	2E-09
SI	6E-10	6E-10
SX	0E+00	0E+00
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 Station 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	-6E-09	-7E-10
MS	9E-11	1E-11
RCS	8E-08	2E-09
RHR	4E-09	4E-09
SI	-5E-08	-5E-08
SX	0E+00	0E+00
TOTAL	3E-08	-5E-08

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:

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

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 RI-ISI 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 RI-ISI evaluation scope is calculated by summing the changes in risk for each individual system, as follows:

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

The ΔLERF for each system was calculated as the product of the ΔCDF, and a factor equivalent to the ratio of the CLERP to the CCDP selected for each system. In addition, the ΔLERF from unisolable LOCAs outside containment was added for those systems with piping segments subject to this phenomenon (SI and RH). 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 RI-ISI 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, 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 Station 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	-5E-09	-4E-09	-6E-10	-5E-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	0E+00	0E+00	0E+00	0E+00
TOTAL	3E-07	9E-08	9E-9	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 Station 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	-6E-09	-6E-09	-7E-10	-7E-10
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	0E+00	0E+00	0E+00	0E+00
TOTAL	2E-07	3E-08	-8E-09	-5E-08

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

<u>Question Br.12:</u>
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, conditional 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.

<u>Braidwood Response to Question Br. 12:</u>
The requested information on parameter values and data sources is provided in the following table.

MODEL/EQUATION	REPORT REFERENCE	PAGE, TABLE, EQUATION REFERENCES
Equations for calculating changes in CDF and LERF	EPRI TR-112657	Equation 3-9 on p. 3-86
Equation for Calculating CDF and LERF	EPRI TR-110161	Equation 3.40 on p. 3-34
Markov Model used for ISI amenable damage mechanisms	EPRI TR-110161	Figure 3-9 on p. 3-24 Equations (3.26) though (3.38) on pp. 3-24 to 3-27
Definition of Inspection Effectiveness Factor for use in delta risk equation	EPRI TR-110161	$I = \frac{h_{40}\{\omega_{NEW}\}}{h_{40}\{\omega_{OLD}\}}$ This is similar to Equation (3.41) on p. 3-37 except that 40 year vs. steady state hazard rates are used. NEW corresponds with RISI and OLD with ASME Sec. XI.
Definition of the flaw inspection repair rate, ω	EPRI TR-110161	Equation (3.23) on p. 3-18
Definition of the leak detection repair rate, μ	EPRI TR-110161	Equation (3.24) on p. 3-18
Failure rates and rupture frequencies	EPRI TR-111880	Table A-9
Plant specific documentation of all other input data needed to quantify above equations	Braidwood Units 1 and 2 RISI Evaluation (Tier-2 Documentation)	Section 7

Attachment C provides the input parameters and contains a more detailed description of the Markov Model.

Question 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.

Braidwood Response to Question Br. 13:

The inspection effectiveness factor is the ratio of the inspected weld rupture frequency to the non-inspected rupture frequency. The EPRI Topical Report, Section 3.7.2, discusses two methods for determining these factors; one based on an application of the Markov model and the other based on an assumption that the factor is proportional to the complement of the probability of detection (POD) of the ISI examination. The POD is the conditional probability of detection of damage in a pipe element, given the existence of a detectable flaw or crack in the pipe element that exceeds the pipe repair criteria. When the effectiveness factor is developed from the Markov model, the following variables impact its numerical value: 1) the POD which may be different whether the examination is done per ASME Section XI or per EPRI RI-ISI examination criteria, 2) the assumed failure rates and rupture frequencies which are taken to be dependent and conditional on the system, 3) pipe size, and 4) applicable ISI amenable damage mechanisms. There are other inputs to the Markov model that are not varied between EPRI and ASME Section XI programs that describe the frequency and effectiveness of pipe leaks when leak before break applies.

A tabulation of all the unique inspection effectiveness factors for all pipe segments evaluated within the scope of the RI-ISI evaluation for Braidwood Station Units 1 and 2 is presented in Table Br-13. For comparison purposes, the corresponding POD values that were used were presented along with their complements that provide the alternative method of computing the inspection effectiveness factor. A plot that compares the two approaches to computing the inspection effectiveness factors is provided in Figure RAI-Br.13 for the RI-ISI exams.

As seen in these exhibits, there is relatively good agreement between these alternative approaches to estimating the inspection effectiveness factors. When the POD values are approximately .50, the Markov model predicts a higher level of inspection effectiveness, as reflected in lower inspection effectiveness factors. For higher POD values, the Markov model predicts a lower level of inspection effectiveness, as reflected in higher inspection effectiveness factors. Details documenting the inputs to computing these factors are discussed in response to RAI-Br. 12 above.

The inspection effectiveness factors developed using the Markov model are considered a more realistic assessment of inspection effectiveness for these reasons.

- The use of the "1-POD" model for inspection effectiveness is simply an assumption and has no real logical or scientific basis, whereas the Markov model does.
- The Markov model is based on an explicit model of the interactions between degradation phenomena and inspection processes. The results of the Markov model are a function of the POD as well as many other parameters that account for the relative frequency of cracks, leaks, and ruptures, the possibility for leak before break and leak detection and repair prior to rupture, the fraction of the weld that is accessible, the possibility for synergy between different damage mechanisms and the time intervals between inspections.

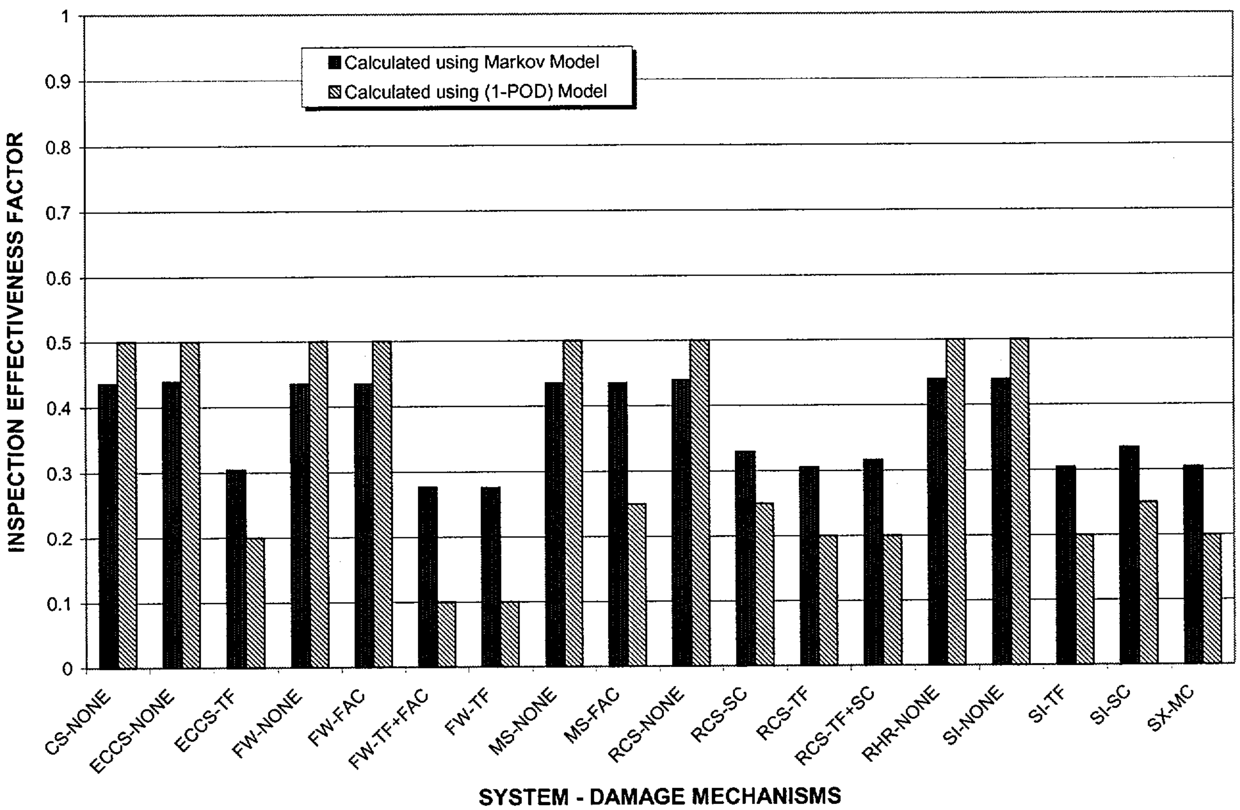
However, it is noted that in the context of developing order of magnitude estimates of risk impacts, both methods provide comparable results as seen in Table RAI-Br.13 and Figure RAI-Br.13.

Table RAI-Br.13: Probability of Detection (POD) and Inspection Effectiveness Factors. Used for PWR Delta Risk Evaluations

SYSTEM	DAMAGE MECHANISM(S)	EPRI RISI EXAMS			ASME SECTION XI EXAMS		
		POD	INSPECTION EFFECTIVENESS FACTOR PER MARKOV MODEL	INSPECTION EFFECTIVENESS FACTOR PER (1-POD)	POD	INSPECTION EFFECTIVENESS FACTOR PER MARKOV MODEL	INSPECTION EFFECTIVENESS FACTOR PER (1-POD)
CS	D&C[1]	0.500	0.436	0.500	0.500	0.436	0.500
CVCS	D&C[1]	0.500	0.439	0.500	0.500	0.439	0.500
	TT	0.800	0.305	0.200	0.500	0.438	0.500
FW	D&C[1]	0.500	0.435	0.500	0.500	0.435	0.500
	FAC	0.500	0.435	0.500	0.500	0.435	0.500
	TT, TASCS, FAC	0.900	0.277	0.100	0.500	0.440	0.500
	TT, FAC	0.900	0.277	0.100	0.500	0.440	0.500
	TT	0.900	0.276	0.100	0.500	0.439	0.500
MS	D&C[1]	0.500	0.435	0.500	0.500	0.435	0.500
	FAC	0.500	0.435	0.500	0.500	0.435	0.500
RCS	D&C[1]	0.500	0.439	0.500	0.500	0.439	0.500
	TT	0.800	0.306	0.200	0.500	0.439	0.500
	TASCS	0.800	0.306	0.200	0.500	0.439	0.500
	IGSCC	0.750	0.329	0.250	0.500	0.439	0.500
	TT, TASCS	0.800	0.306	0.200	0.500	0.439	0.500
	PWSCC	0.750	0.329	0.250	0.500	0.439	0.500
	TT, PWSCC	0.800	0.316	0.200	0.500	0.450	0.500
RHR	D&C[1]	0.500	0.439	0.500	0.500	0.439	0.500
SI	D&C[1]	0.500	0.435	0.500	0.500	0.435	0.500
	IGSCC	0.750	0.334	0.250	0.500	0.450	0.500
	TASCS	0.800	0.305	0.200	0.500	0.438	0.500
	TT	0.800	0.305	0.200	0.500	0.438	0.500
	TT, TASCS	0.800	0.305	0.200	0.500	0.438	0.500
SX	MIC, PIT	0.500	0.435	0.500	0.500	0.435	0.500

1 Design and construction errors were included for all welds and are shown here only for cases with no other damage mechanism present.

Figure RAI-Br.13: Comparison of Inspection Effectiveness Factors for EPRI RISI Exams at Braidwood Units 1 and 2*



* Notes regarding system damage mechanisms plotted in Figure RAI-Br.13:

1) All weld locations are considered susceptible to Design and Construction Errors including welds listed with NONE for damage mechanisms
2) SC refers to stress corrosion cracking mechanisms such as IGSCC and PWSCC
3) TF refers to thermal fatigue and includes Thermal Transients (TT) and Thermal Stratification, Cycling and Striping (TASCS)
4) MC refers to microbiologically influenced corrosion (MIC) and pitting (PIT)

Attachment B

Response to Request for Additional Information

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"

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

<u>RAI Question By.1:</u>
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.

<u>Byron Response to Question By.1:</u>
A review of the element/weld population was performed and the percentage of volumetric examinations was determined to exceed the minimum of 10% of the Class 1 butt welds. See table below.

Table RAI-By.1: 10% Selection Criteria for Byron Unit 1 and Unit 2

BYRON CLASS 1 WELD EXAM SELECTIONS BASED ON EPRI TR-11880 DATA				
UNIT	TOTAL CLASS 1 WELDS	TOTAL CLASS 1 BUTT WELDS	NUMBER SELECTED FOR VOLUMETRIC EXAMINATION	PERCENTAGE SELECTED FOR VOLUMETRIC EXAMINATION
1	1580	792	83	10.48%
2	1534	795	84	10.57%
Note: Class 1 population consists of item numbers B5.10, B5.40, B5.70 for Category B-F and B9.11, B9.21, B9.31 and B9.32 for Category B-J. Item number B9.40 (B-J) is excluded from the butt weld counting due to component configurations.				

The totals of Class 1 welds receiving volumetric examination exceeds 10% for both Byron Station Unit 1 and Unit 2.

<u>RAI Question By.2:</u>
Please provide the following information for both units:

<u>RAI Question By.2(a):</u>
When does the current 10-year ISI interval start and end?

<u>Byron Response to Question By.2(a):</u>
Unit 1: Interval started on July 1, 1996, and will end on September 15, 2005.
Unit 2: Interval started on August 16, 1998, and will end on August 21, 2007.

RAI Question By.2(b):
When does the current ISI period start and end?

Byron Response to Question By.2(b):
Unit 1: Period 2 started September 16, 1999, and will end September 15, 2002.
Unit 2: Period 1 started August 16, 1998, and ended August 21, 2001.
Period 2 started August 22, 2001, and will end August 21, 2004.

RAI Question By.2(c):
What cumulative percentage of inspections have been completed for the current interval?

Byron Response to Question By.2(c):
See table below.

Table RAI-By.2c: Examination Completion Status for Byron Station Unit 1 and Unit 2

CATEGORY	UNIT 1			UNIT 2		
	SELECTED	COMPLETED	% COMPLETE	SELECTED	COMPLETED	% COMPLETE
B-F	22	4	18.18%	22	5	22.73%
B-J	393	117	29.77%	381	116	30.45%
C-F-1	79	26	32.91%	79	25	31.65%
C-F-2	49	13	26.53%	58	14	24.14%
TOTAL	543	160	29.47%	540	160	29.63%
Note: Table identifies status as of the end of the 1st Period for examinations under ASME Section XI for the categories that are to be incorporated into RI-ISI. Unit 1 has completed one of two outages in the 2nd Period in which the selections were based on RI-ISI. Completion status for both units at the end of the 2nd Period will be under RI-ISI requirements.						

RAI Question By.2(d):
When will the next refueling outage start?

Byron Response to Question By.2(d):
Unit 1: Byron Unit 1, Refuel 11 (i.e., B1R11) scheduled to start on March 9, 2002.
Unit 2: Byron Unit 2, Refuel 10 (i.e., B2R10) scheduled to start on September 21, 2002.

RAI Question 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.

Byron Response to Question By.3:
The Risk Informed Inservice Inspection (RI-ISI) Program will start with the 1st Period at 29.47% and 29.63% percentage of volumetric examinations for Unit 1 and 2 respectively (i.e., see Response to Question By.2c). Component selections during the 1st Inspection Period were not subject to the criteria of the RI-ISI Program. The remainder of the selected components will be examined before the end of the current inspection interval with the maximum allowable percentage at the end of the 2nd Period at

67%, as required by Table IWA-2412-1. The equivalent percentage of the current interval's 1st Period will be examined in the 1st Period of the subsequent interval.

This method of RI-ISI incorporation would result in the completion of 100% of the RI-ISI components with in a ten-year time frame as would occur if the RI-ISI program were started at the beginning of the inspection interval. The current period and interval dates will not be altered by this method.

<u>RAI Question By.4:</u>
For Relief Request I2R-40:

<u>RAI Question By.4(a):</u>
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.

<u>Byron Response to Question By.4(a):</u>
At Byron Station, all dissimilar metal welds are included in Category B-F. Currently, there are no piping dissimilar metal welds in the station's Class 1 piping systems. This statement was included for reference to code requirements only. The applicable code edition in use at Byron Station is the 1989 Edition. The statement about the "dissimilar metal welds not covered by Category B-F" is in later code editions. With the adoption of RI-ISI, the requirements of Table 2500-1, Category B-J will be superceded by Table 1, Category R-A.

<u>RAI Question By.4(b):</u>
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.

<u>Byron Response to Question By.4(b):</u>
The discussion of the Table IWC 2500-1 requirements also apply to Category C-F-2. The wording of the paragraph on page 2 of 3 of Relief Request I2R-40 should read:

> "Table IWC 2500-1 requires a volumetric and surface examination for items C5.11, C5.21, and C5.51 and a surface examination for items C5.30, C5.41, C5.70, and C5.81 for those welds selected per the following:"

<u>RAI Question By.5:</u>
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.

Byron Response to Question By.5:
As part of the Electric Power Research Institute (EPRI) methodology for RI-ISI, information was obtained to assess the susceptibility of piping to water hammer. No susceptibility was identified for any piping within the RI-ISI evaluation scope.

Water hammer is not shown as a degradation mechanism, because it is not a degradation mechanism, as discussed below. Water hammer is, however, evaluated during the degradation mechanism assessment. The degradation mechanism assessment (DMA) is documented in Section 4 of the Byron RI-ISI Tier-2 report. The following excerpts are taken from Section 4 of the Tier-2 report:

From Section 4.2.2:

> "Water hammer is an unanticipated, infrequent loading that can produce relatively high loads. Although water hammer is not a degradation mechanism the susceptibility of each piping system to water hammer is a factor in the ranking of examination locations, and is considered in the DMA."

From Section 4.2.3:

> "The medium failure potential in segments is changed to high if the system containing that segment has a history of water hammer."

From Section 4.2.4:

> "Water hammer events have occurred previously at Byron and Braidwood. These events occurred in the balance of plant (BOP) or in materials that are outside the evaluation scope for this study. Based on the differences in systems design and operating conditions associated with the events and those of the evaluation scope for this study, it is judged that water hammer is not credible for purpose of the Byron RI-ISI evaluation."

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

Byron Response to Question By.6:
Within the areas bounded by the RI-ISI Program, the following piping failures have occurred:

During the Emergency Core Cooling System (ECCS) full-flow testing in the Byron Unit 1 Refueling 8 (i.e., B1R08) outage a 1½" Safety Injection (SI) socket weld fractured and leaked (i.e., line number 1SI08JD-1½"). Metallurgical testing showed that the initiating event was lack of fusion at the weld root. This caused a stress-riser that quickly propagated from the root to the outer surface. This event was documented by Condition Report B1998-00839.

After a new valve (i.e., 1RC8029B) was installed on a ¾" loop-bypass vent line, the pipe to valve socket weld failed shortly after returning to service. The failure mechanism was determined to be high cyclic vibration caused by the heavier new valve. Note: this line is exempted from ISI/RI-ISI examination by IWB 1220 due its size (i.e., <1½" NPS). This event was documented by Condition Report B1999-01973.

<u>RAI Question By.7:</u>
Table 1 of Attachment 2 identifies the reactor coolant (RC) system as one of the systems for RI-ISI implementation.

<u>RAI Question By.7(a):</u>
Footnote 2 of Table 1 clarifies that pressurizer relief piping was included. Are thermowells also included with this system?

<u>Byron Response to Question By.7(a):</u>
Thermowells are included as Code Item B9.40 in the Byron Station ISI Program Plan and are contained within the RC system.

<u>RAI Question By.7(b):</u>
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.

<u>Byron Response to Question By.7(b):</u>
Pressurizer piping is identified as the "RY" system (i.e., pressurizer surge, spray, and relief). All SG primary side safe-end welds are contained within the RC system, and the secondary-side piping is identified with the applicable Feedwater (FW) or MS systems.

<u>RAI Question By.8:</u>
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."

<u>RAI Question By.8(a):</u>
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.

<u>Byron Response to Question By.8(a):</u>
Of the 160 FW welds initially classified as Risk Category 3, 33 welds were subsequently removed from the RI-ISI population due to the lack of a degradation mechanism other than Flow Accelerated Corrosion (FAC). These welds are now subject to the High Energy Line Break (HELB) augmented program in

addition to the FAC Program. Of the remaining 127 Category 3 welds, 32 were selected for examination. This results in a selection percentage of 25.20%.

RAI Question By.8(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.

Byron Response to Question By.8(b):
Of the 274 FW welds initially classified as Risk Category 3, 32 welds were subsequently removed from the RI-ISI population due to the lack of a degradation mechanism other than FAC. These welds are now subject to the HELB augmented program in addition to the FAC Program. Of the remaining 242 Category 3 welds, 61 were selected for examination. This results in a selection percentage of 25.21%.

RAI Question By.8(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.

Byron Response to Question By.8(c):
The note for Table 6 is misleading. The data given on Table 5 is applicable to Unit 1 as indicated in the table title. The data on Table 6 is applicable to Unit 2 as indicated in the table title. The "NOTE" on Table 6 will be deleted and replaced with the same NOTES (1) and (2) currently shown on Table 5 at the next revision of the Tier 2 document.

RAI Question By.8(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.

Byron Response to Question By.8(d):
The Essential Service Water System (SX) welds were removed from the RI-ISI element selection population and continue to be addressed by the service water inspection program. The SX piping segments were all categorized as high consequence. The SX welds are in piping supplying the containment fan coolers and are all, except for four welds at each unit, inside the containment. The SX welds were eliminated from element selection due to their inclusion in an augmented inspection program and the absence of any other damage mechanisms.

The Byron Station Service Water Integrity Program that complies with Generic Letter (GL) 89-13, "Service Water System Problems Affecting Safety-Related Equipment," consists of the following five program elements.

- Inspect, chemically treat, and flush service water flow paths.
- Test program for safety-related heat exchangers cooled with service water.
- Silting, erosion, and corrosion inspections.
- Confirmation that the service water system will perform its intended function.

- Confirmation that maintenance practices, operating and emergency procedures and training are adequate for the service water system.

Inspection of the in-plant service water piping and components are addressed in the "silting, erosion, and corrosion inspections" program element. A systematic approach similar to that developed for the FAC program is used to select the five most susceptible points with low flow and five most susceptible points with high flow, for periodic ultrasonic examinations. Low flow locations are typically characterized by low flow rate areas (i.e., <3 feet per second) and dead legs while high flow locations are typically characterized by high flow velocities and flow restrictions. Engineering judgement resulting from system walkdowns is also considered in the selection process. Selection of the most susceptible locations in the service water system for periodic examination assures that structural integrity of the complete service water system is monitored without the expenses of conducting the examination on each subsystems or code class.

Ultrasonic examination is performed on the five low flow and five high flow locations once every refueling outage until acceptability of the pipe can be determined, at which time less frequent inspections may be performed. Examinations on the low flow locations will assess pipe degradation caused by Microbiologically Influenced Corrosion (MIC) where silting may be occurring while examinations on the high flow locations will identify if any high flow erosion problems are occurring. If problems are found, they will be evaluated and the necessary corrective actions will be taken.

With regard to localized corrosion mechanisms (e.g., MIC, pitting, etc.), Section 3.6.7 in EPRI TR-112657, "Revised Risk-Informed Inservice Inspection Evaluation Procedure," Revision B-A, December 1999, provides alternative element selection criteria that may be used in lieu of the sampling percentages in Section 3.6.4.2. These alternatives include either (1) the use of existing plant programs (e.g., service water inspection program) per Subsection 3.6.7.1; (2) enhancements to existing plant programs per Subsection 3.6.7.2; or (3) the development and implementation of a replacement inspection program. At Byron Station, alternative (1) was used for the Class 2 SX system piping. In doing so the existing service water piping inspection program at Byron Station was determined to be an acceptable alternative to the sampling percentages in Section 3.6.4.2 for the following reasons.

1. The effectiveness of the Byron Station service water inspection program has been reviewed by the NRC and was determined to have met the recommendations specified in GL 89-13.
2. The MIC potential for the Class 2 SX system piping evaluated as part of the RI-ISI program is considered to be low. This piping normally operates continuously under full flow conditions and the time at which the piping is subjected to low/intermediate flows is relatively small. Secondly, daily biocide (i.e., chlorination) protection measures have been implemented for the entire SX system to mitigate the effects of microbiological fouling and microbiological influenced corrosion. Also a scale inhibitor is continuously injected.
3. Ultrasonic examinations that monitor wall thickness are designed to focus at system locations where the MIC attack is expected to be most aggressive (e.g., low/stagnate flow and/or where silting may occur). Although these high MIC potential areas do not include any of the Class 2 piping, the process is to expand the inspection scope to other less susceptible SX system piping should significant MIC problems be noted. In addition, all SX system piping is subjected to VT2 examinations during routine leakage/pressure testing in accordance with American Society of Mechanical Engineers (ASME) Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components."
4. The risk associated with a pipe rupture is HIGH for all SX system piping because of the potential for MIC and the high consequences that could result from a SX system pipe break. Therefore, by

focusing ultrasonic examinations at those locations in the system that have the highest potential for MIC damage, the service water inspection program is consistent with the objectives for RI-ISI and the existing degradation management activities and VT2 examinations implemented on the less susceptible Class 2 SX system piping are considered adequate.

<u>RAI Question By.8(e):</u>
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% of these welds are not inspected as required by the code case.

<u>Byron Response to Question By.8(e):</u>
Of the 16 total MS welds initially classified as Risk Category 3, all 16 welds were subsequently removed from the RI-ISI population due to the lack of a degradation mechanism other than FAC. These welds are now subject to the HELB augmented program in addition to the FAC Program.

<u>RAI Question By.9:</u>
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.

<u>Byron Response to Question By.9:</u>
Class 1 and Class 2 socket welds will receive a VT-2 examination during each refueling outage as specified in Code Case N-578-1, Table 1 Examination Category R-A, Note 12.

<u>RAI Question By.10:</u>
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 I2R-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.

<u>Byron Response to Question By.10:</u>
Code Case N-524 will no longer be directly applicable to the inspection of Class 2 welds and therefore will be removed from the ISI plan upon approval of the relief request. (Note: Byron Station Class 1 piping does not contain longitudinal welds). Code Case N-524 is approved as an alternative to Section XI, Examination Categories C-F-1 and C-F-2. Upon approval of the risk-informed submittal, the requirements of Examination Categories C-F-1 and C-F-2 will no longer be applicable to Class 2 welds; therefore, Code Case N-524 will no longer be directly applicable. However, Byron Station understands that the alternative requirements of the Code Case are still valid under the risk-informed inspection

program. To this extent and in accordance with footnote (4) of Code Case N-578-1, Table 1, Examination Category R-A, Byron Station will examine those longitudinal welds that intersect the circumferential welds selected under the risk-informed process. For those intersecting longitudinal welds, the portion of the weld within the associated circumferential weld volume will be inspected, and the inspection requirements for the longitudinal weld will be met for both transverse and parallel flaws.

RAI Question 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.

Byron Response to Question By.11:

The PRA model used to support the RI-ISI is documented in two calculations, Byron Station PRA CDF Calculation, BYR-99-040, Rev. 0 and Byron Station PRA LERF Calculation, BYR-99-096, Rev. 0. The Byron Station Unit 1 model was used for all quantifications as the CDF and LERF results from the Unit 1 and 2 models did not differ appreciably for this application. Separate Unit 1 and Unit 2 models are available to support configuration risk management applications that require unit specific results. The CDF from the Unit 1 model is 4.98E-05 and the LERF is 5.55E-06.

RAI Question 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?

Byron Response to Question By.12:

How was this synergy reflected in the risk impact?

For segments with two or more ISI amenable damage mechanisms, the associated failure rates and rupture frequencies for these and design and construction errors are summed, with the exception that Intergranular Stress Corrosion Cracking (IGSCC) and FAC contributions are not added if the weld is part of the associated augmented inspection program for IGSCC 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 RI-ISI program were included. However, when there are two or more damage mechanisms, including IGSCC 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.

The above treatment was made because the service data upon which the EPRI methodology for damage mechanism assessment was based does not explicitly address multiple damage mechanisms. Two 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 corrosion cracking and the corrosion cracking is not covered in the augmented program for IGSCC (i.e., a hypothetical case), the failure rates for design and construction errors, thermal fatigue, and stress corrosion cracking from EPRI TR-111880, "Piping System Failure Rates and

Replacement Frequencies for use in Risk Informed Inservice Inspection Applications," would be summed and then this result would be multiplied by a factor of three for synergy. The rupture frequencies would be determined in the same way. But if the segment was found susceptible to the same three damage mechanisms and the stress corrosion cracking was covered in the augmented IGSCC program, the stress corrosion cracking contribution would not be included in the failure rate or rupture frequency, but its synergy effects would be included by the factor of three.

<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 RI-ISI 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 TR-111880. Hence if a location was susceptible to say two or more ISI amenable damage mechanism 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 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.

<u>RAI Question By.13:</u>
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.

<u>Byron Response to Question By.13:</u>
Welds identified as having FAC as the only degradation mechanism are removed from the RI-ISI 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 Section XI will not be selected for inspection under the RI-ISI program, but will continue to be addressed by the FAC program. The FAC-only welds are listed in the delta risk calculation tables, but no change in risk is calculated for these welds when Section XI examinations are eliminated at any of these welds.

<u>RAI Question By.14:</u>

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?

<u>Byron Response to Question By.14:</u>

The differences in the segments and weld populations are a result of the physical differences between the units.

For the RC systems, the weld number differences between the units are due to differences in the as-built conditions, principally in the routing of small-bore (i.e., <4") piping. See table below.

Table RAI-By.14-1: Population of RC welds (all RI-ISI Categories) for Byron Unit 1 and Unit 2

RC ITEM NUMBERS	UNIT 1 BEFORE SGR	CURRENT UNIT 1	CURRENT UNIT 2
B5.10	8	8	8
B5.70	8	8	8
B9.11	198	206	200
B9.21	43	43	43
B9.31	11	11	11
B9.32	44	44	44
B9.40	267	273*	241
TOTAL	**579**	**593**	**555**

* Includes 6 welds added to the RC system during the Loop Stop Isolation Valve project after the SGR.

For the FW system, the Unit 1 SG replacement eliminated all of the Auxiliary FW piping in containment and reduced the length of the Main FW lines. This accounts for fewer welds in the Unit 1 FW system as compared to Unit 2. See table below.

Table RAI-By.14-2: Population of FW welds (all RI-ISI Categories) for Byron Unit 1 and Unit 2

FW ITEM NUMBERS	UNIT 1 BEFORE SGR	CURRENT UNIT 1	CURRENT UNIT 2
C5.11	0	16	0
C5.51	267	148	274
C5.81	0	4	0
TOTAL	**267**	**168**	**274**

RAI Question By.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.

Byron Response to Question By.15:

The "separate Markov calculation" in the original submittal represents the unisolable LOCA outside containment for lines connected to the RC system. However, a simplified approach has been taken by assuming the CLERP/Conditional Core Damage Probability (CCDP) ratio for those systems susceptible to unisolable LOCA outside containment (i.e., RH and SI) was 1.0, (i.e., ΔLERF = ΔCDF). This is further explained in the response to RAI By.17.

RAI 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 Station RI-ISI evaluations that bear on the acceptability of the element selections that were made in implementing the EPRI RI-ISI methodology.

The NRC position reflected in this RAI question is that since the failure rates from the EPRI Topical Report, 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 RC, SI, CV, and RH systems which capture most of the segments in which elements were removed and fully encompass the segments with significant CCDP values.

After review of this RAI 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-11180. When these frequencies were applied to the RC system, the delta CDF calculations failed to meet the system level success criterion of 1E-7/year. As a result, additional inspections were added to the Byron Station RI-ISI 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 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 Station 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 17 for a discussion of the differences between the bounding and Markov calculations.

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

(1) Butt weld

Table RAI By-16-C: Revised Risk Impact Results for Byron Station 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	0E+00	0E+00
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 Station 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	-4E-08	-5E-09
MS	9E-11	1E-11
RCS	9E-08	2E-09
RHR	-8E-10	-8E-10
SI	-5E-08	-5E-08
SX	0E+00	0E+00
TOTAL	-1E-08	-5E-08

<u>RAI Question By.17:</u>
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 Br. 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 a previously approved relief request for South Texas Project. The change in risk for a particular system was calculated using the following.

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

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 RI-ISI 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 RI-ISI evaluation scope is calculated by summing the changes in risk for each individual system, as follows:

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

The ΔLERF for each system was calculated as the product of the ΔCDF, and a factor equivalent to the ratio of the CLERP to the CCDP selected for each system. In addition, the ΔLERF from unisolable LOCAs outside containment was added for those systems with piping segments subject to this phenomenon (i.e., SI and RH). 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 RI-ISI 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 RC system 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 RG 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 RG 1.174 of 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 RC system exceeded the change in CDF criterion of 1.0E-07 per system per year. The total change in CDF was 4E-08, 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 -9E-08, 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 RC system, the Markov modeling approach was applied. Using the Markov model, the details of which are discussed in response to Questions 18 and 19, 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 Station Unit 1 and By-17-B for Byron Station Unit 2.

Table By-17-A: Comparison of Risk Impact Results for Byron Station 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	0E+00	0E+00	0E+00	0E+00
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 Station 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	-5E-08	-4E-08	-6E-09	-5E-09
MS	2E-10	9E-11	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	0E+00	0E+00	0E+00	0E+00
TOTAL	4E-08	-1E-08	-9E-08	-5E-08

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

<u>RAI Question By.18:</u>

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).

<u>Byron Response to Question By.18:</u>

The requested information on equations and data sources is provided in the table below.

Table RAI-By.18: Cross References

MODEL/EQUATION	REPORT REFERENCE	PAGE, TABLE, EQUATION REFERENCES
Equations for calculating changes in CDF and LERF	EPRI TR-112657	Equation 3-9 on p. 3-86
Equation for calculating CDF and LERF	EPRI TR-110161	Equation 3.40 on p. 3-34
Markov Model used for ISI amenable damage mechanisms	EPRI TR-110161	Figure 3-9 on p. 3-24 Equations (3.26) though (3.38) on pp. 3-24 to 3-27
Definition of Inspection Effectiveness Factor for use in delta risk equation	EPRI TR-110161	$I = \frac{h_{40}\{\omega_{NEW}\}}{h_{40}\{\omega_{OLD}\}}$ This is similar to Equation (3.41) on p. 3-37 except that 40 year vs. steady state hazard rates are used. NEW corresponds with RISI and OLD with ASME Sec. XI.
Definition of the flaw inspection repair rate, ω	EPRI TR-110161	Equation (3.23) on p. 3-18
Definition of the leak detection repair rate, μ	EPRI TR-110161	Equation (3.24) on p. 3-18
Failure rates and rupture frequencies	EPRI TR-111880	Table A-9
Plant specific documentation of all other input data needed to quantify above equations	Byron Units 1 and 2 RISI Evaluation (Tier-2 Documentation)	Section 7

Attachment C provides the input parameters and contains a more detailed description of the Markov Model extracted from the Braidwood Station and Byron Station Tier-2 documents.

<u>***RAI Question By.19:***</u>
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.

<u>**Byron Response to Question By.19:**</u>
The inspection effectiveness factor is the ratio of the inspected weld rupture frequency to the non-inspected rupture frequency. The EPRI Topical Report, Section 3.7.2, discusses two methods for determining these factors, one based on an application of the Markov model and the other based on an assumption that the factor is proportional to the complement of the probability of detection (POD) of the ISI examination. The POD is the conditional probability of detection of damage in a pipe element, given the existence of a detectable flaw or crack in the pipe element that exceeds the pipe repair criteria. When the effectiveness factor is developed from the Markov model, the following variables impact its numerical value: 1) the POD which may be different whether the examination is done per ASME Section XI or per EPRI RI-ISI examination criteria, 2) the assumed failure rates and rupture frequencies which are taken to be dependent and conditional on the system, 3) pipe size and 4) applicable ISI amenable damage mechanisms. There are other inputs to the Markov model that are not varied between EPRI and ASME Section XI programs that describe the frequency and effectiveness of pipe leaks when leak before break applies.

A tabulation of all the unique inspection effectiveness factors for all pipe segments evaluated within the scope of the RI-ISI evaluation for Byron Station Units 1 and 2 is presented in Table RAI-By.19. For comparison purposes, the corresponding POD values that were used were presented along with their complements that provide the alternative method of computing the inspection effectiveness factor. A plot that compares the two approaches to computing the inspection effectiveness factors is provided in Figure RAI-By.19 for the RI-ISI exams.

As seen in these exhibits, there is relatively good agreement between these alternative approaches to estimating the inspection effectiveness factors. When the POD values are approximately .50, the Markov model predicts a higher level of inspection effectiveness, as reflected in lower inspection effectiveness factors. For higher POD values, the Markov model predicts a lower level of inspection effectiveness, as reflected in higher inspection effectiveness factors. Details documenting the inputs to computing these factors are discussed in response to RAI Question 18 above.

The inspection effectiveness factors developed using the Markov model are considered a more realistic assessment of inspection effectiveness for these reasons.

- The use of the "1-POD" model for inspection effectiveness is simply an assumption and has no real logical or scientific basis, whereas the Markov model does.
- The Markov model is based on an explicit model of the interactions between degradation phenomena and inspection processes. The results of the Markov model are a function of the POD as well as many other parameters that account for the relative frequency of cracks, leaks, and ruptures, the possibility for leak before break and leak detection and repair prior to rupture, the fraction of the weld that is accessible, the possibility for synergy between different damage mechanisms and the time intervals between inspections.

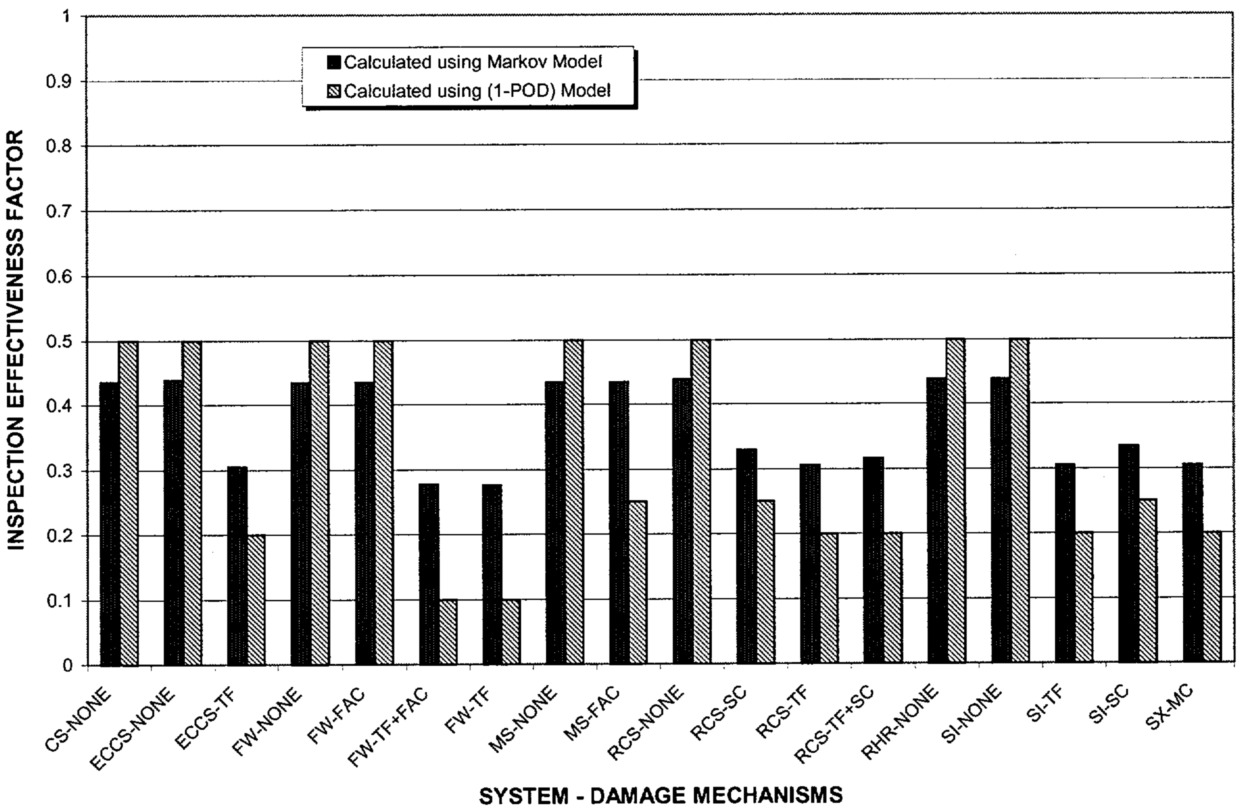
However, it is noted that in the context of developing order of magnitude estimates of risk impacts, both methods provide comparable results as seen in the presented exhibits.

Table RAI-By.19: Probability of Detection (POD) and Inspection Effectiveness Factors. Used for PWR Delta Risk Evaluations

SYSTEM	DAMAGE MECHANISM(S)	EPRI RISI EXAMS			ASME SECTION XI EXAMS		
		POD	INSPECTION EFFECTIVENESS FACTOR PER MARKOV MODEL	INSPECTION EFFECTIVENESS FACTOR PER (1-POD)	POD	INSPECTION EFFECTIVENESS FACTOR PER MARKOV MODEL	INSPECTION EFFECTIVENESS FACTOR PER (1-POD)
CS	D&C[1]	0.500	0.436	0.500	0.500	0.436	0.500
CVCS	D&C[1]	0.500	0.439	0.500	0.500	0.439	0.500
	TT	0.800	0.305	0.200	0.500	0.438	0.500
FW	D&C[1]	0.500	0.435	0.500	0.500	0.435	0.500
	FAC	0.500	0.435	0.500	0.500	0.435	0.500
	TT, TASCS, FAC	0.900	0.277	0.100	0.500	0.440	0.500
	TT, FAC	0.900	0.277	0.100	0.500	0.440	0.500
	TT	0.900	0.276	0.100	0.500	0.439	0.500
MS	D&C[1]	0.500	0.435	0.500	0.500	0.435	0.500
	FAC	0.500	0.435	0.500	0.500	0.435	0.500
RCS	D&C[1]	0.500	0.439	0.500	0.500	0.439	0.500
	TT	0.800	0.306	0.200	0.500	0.439	0.500
	TASCS	0.800	0.306	0.200	0.500	0.439	0.500
	IGSCC	0.750	0.329	0.250	0.500	0.439	0.500
	TT, TASCS	0.800	0.306	0.200	0.500	0.439	0.500
	PWSCC	0.750	0.329	0.250	0.500	0.439	0.500
	TT, PWSCC	0.800	0.316	0.200	0.500	0.450	0.500
RHR	D&C[1]	0.500	0.439	0.500	0.500	0.439	0.500
SI	D&C[1]	0.500	0.435	0.500	0.500	0.435	0.500
	IGSCC	0.750	0.334	0.250	0.500	0.450	0.500
	TASCS	0.800	0.305	0.200	0.500	0.438	0.500
	TT	0.800	0.305	0.200	0.500	0.438	0.500
	TT, TASCS	0.800	0.305	0.200	0.500	0.438	0.500
SX	MIC, PIT	0.500	0.435	0.500	0.500	0.435	0.500

1) Design and construction errors were included for all welds and are shown here only for cases with no other damage mechanism present.

Figure RAI-By.19: Comparison of Inspection Effectiveness Factors for EPRI RISI Exams at Byron and Braidwood Units 1 and 2*



* Notes regarding system damage mechanisms plotted in Figure RAI-By.19:

1) All weld locations are considered susceptible to Design and Construction Errors including welds listed with NONE for damage mechanisms
2) SC refers to stress corrosion cracking mechanisms such as IGSCC and PWSCC
3) TF refers to thermal fatigue and includes Thermal Transients (TT) and Thermal Stratification, Cycling and Striping (TASCS)
4) MC refers to microbiologically influenced corrosion (MIC) and pitting (PIT)

<u>RAI Question By.20:</u>

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.

Byron Response to Question By.20:

Under the current Section XI program, 22 SX system welds are being inspected at Unit 1 and 22 SX system welds are being inspected at Unit 2.

No other damage mechanisms, other than MIC/PIT, were identified in the Damage Mechanism Assessment (DMA) for Byron Station SX systems.

SX system welds were removed from the RI-ISI population for element selection and no SX welds were selected for RI-ISI examination. The removal of Section XI exams from SX welds was not included in the RI-ISI delta risk calculations because the piping integrity of these welds was being addressed by the augmented inspection programs. However, to close the open issue on the delta risk impacts of welds covered by augmented programs a sensitivity analysis was performed for welds removed from the RI-ISI population because their damage mechanisms were covered fully in the FAC and Service Water (MIC) programs. These sensitivity analyses, which confirm the judgement, made in the original submittal that such risk impacts are insignificant, are described below

In the Byron Station RI-ISI analysis, welds having FAC as the only damage mechanism or service water welds with MIC/PIT as the only damage mechanism were removed from the population for element selection, i.e., no RI-ISI exams were selected for any of these elements. The FAC-only and service water program elements were assumed to be addressed by their respective augmented inspection programs. In the RI-ISI risk impact calculations, removal of Section XI exams from these welds was not included in the delta risk calculations.

A bounding calculation of delta risk was performed that did not make use of the Markov model. This bounding calculation is now included in Section 7 and Appendix C of the Byron Station Tier-2 RI-ISI documentation.

In order to determine the impact of not including FAC-only and service water program elements in the delta risk calculation, new calculations were performed to include the impact of eliminating Section XI exams from FAC-only and service water program elements. Both Markov model calculations and bounding calculations were made for the systems that had FAC-only or service water program welds within the RI-ISI scope. The results of these calculations of ΔCDF for Byron Unit 1 are shown in Table 1 below. The largest ΔCDF contribution, accounting for the FAC-only and service water program welds, came from the FW system at 1.71E-09 for the realistic Markov calculation and 7.27E-09 for the bounding calculation, more than an order of magnitude below the system ΔCDF limit of 1.00E-07 and an insignificant contributor to the total ΔCDF across all systems. The contributions from the MS system and SX system were all more than two orders of magnitude below the system ΔCDF limit of 1.00E-07.

The results of these calculations of ΔCDF for Byron Station Unit 2 are shown in Table 2 below. The largest ΔCDF contribution, accounting for the FAC-only and service water program welds, came from the SX system at 4.33E-10 for the realistic Markov calculation and 7.66E-10 for the bounding calculation, more than two orders of magnitude below the system ΔCDF limit of 1.00E-07 and an insignificant contributor to the total ΔCDF across all systems. After including the FAC-only welds in the ΔCDF calculation, the delta risk associated with the MS system remained more than two orders of magnitude below the system ΔCDF limit of 1.00E-07 and the delta risk associated with the FW system remained negative (i.e., a risk improvement) for both the realistic and bounding calculations.

Table RAI-By.20-1: Byron Station Unit 1 Impact of Including FAC and Service Water Welds in CDF Delta Risk Calculations

SYSTEM	RI-ISI		INCLUDING FAC WELDS		INCLUDING MIC/PIT WELDS	
	MARKOV	BOUNDING	MARKOV	BOUNDING	MARKOV	BOUNDING
FW	1.63E-09	7.12E-09	1.71E-09	7.27E-09	N/A	N/A
MS	9.89E-11	1.75E-10	1.23E-10	2.18E-10	N/A	N/A
SX	0.00E+00	0.00E+00	N/A	N/A	4.33E-10	7.66E-10

Table RAI-By.20-2: Byron Station Unit 2 Impact of Including FAC and Service Water Welds in CDF Delta Risk Calculations

SYSTEM	RI-ISI		INCLUDING FAC WELDS		INCLUDING MIC/PIT WELDS	
	MARKOV	BOUNDING	MARKOV	BOUNDING	MARKOV	BOUNDING
FW	-3.00E-08	-3.15E-08	-2.98E-08	-3.14E-08	N/A	N/A
MS	9.89E-11	1.75E-10	1.23E-10	2.18E-10	N/A	N/A
SX	0.00E+00	0.00E+00	N/A	N/A	4.33E-10	7.66E-10

<u>RAI Question By.21:</u>

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?

<u>Byron Response to Question By.21:</u>

External events and other modes of operation (e.g., shutdown) were considered in the RI-ISI evaluation in full accordance with the procedures set forth in the EPRI RI-ISI Topical Report and are documented in Section 3.11, "Other Modes of Operation," Section 3.12, "External Events," and Section 3.19 "Shutdown," of the Byron RI-ISI Report (i.e., Tier-2 documentation). These sections, as with all sections of the Tier-2 documentation, were reviewed by Exelon cognizant engineers in the ISI and PRA organizations. These modes and events were considered in accordance with the EPRI RI-ISI Topical report.

Attachment C

Response to Request for Additional Information

Braidwood Station Units 1 and 2
RAI Question Br.12

And

Byron Station Units 1 and 2
RAI Question By.18

Attachment C
Response to Request for Additional Information
Braidwood / Byron Stations Units 1 and 2

Braidwood Response to Question Br.12
Byron Response to Question By.18

RISK IMPACT OF IMPLEMENTING RISK INFORMED INSPECTION PROGRAM

TECHNICAL APPROACH

C.1 Qualitative Evaluation of Changes to Core Damage Frequency (CDF) and Large Early Release Frequency (LERF)

There are three situations in comparing the Risk Informed Inservice Inspection (RI-ISI) program for a particular element selection with the existing ISI program that is being changed, that would lead to changes in CDF or LERF. They are the following.

- Adding elements to the inspection program that were not in the previous inspection program.
- Improving the probability of detection of an inspection by incorporating the "inspection for cause" concept.
- Eliminating an element from the inspection program.

The first two of these items will result in a decrease in pipe failure frequency, and a corresponding decrease in CDF and LERF for each pipe element that applies. The last one will result in at least a small increase in CDF and LERF for each pipe element that applies. For any element that is not impacted by the change to the ISI program, there is no change to the CDF and LERF contribution from pipe failures at such element. Hence the net change in CDF and LERF for a system is comprised of the sum of the changes in CDF and LERF over all the elements in which there is a change in the inspection program. Moreover, even though there may be a large net reduction in the number of welds inspected in a given system, the CDF and/or LERF may actually decrease if the magnitude of changes associated with ISI program enhancements in the high risk segments exceeds that of the elements eliminated from the low risk segments.

C.2 Model for Estimating Changes in CDF and LERF

The Electric Power Research Institute (EPRI) approach to RI-ISI calls for risk impact evaluations to be performed using qualitative analyses, bounding quantitative estimates, or realistic quantitative estimates as illustrated in Figure C-2. This flow chart was developed to minimize the amount of work that was needed to address the risk impact question by first trying to evaluate based on qualitative and bounding quantitative estimates. We have determined that it is better to actually perform realistic quantitative estimates for all pipe elements for the following reasons.

- Full quantification using realistic assumptions will put this application of the Probabilistic Risk Assessment (PRA) on an equal footing with other risk informed programs.
- If any realistic quantitative estimates are needed, the data that is needed for these estimates is a large fraction of the data that is needed for full realistic quantification.

- The evaluation of risk impacts is set up using a spreadsheet which results in a minimal reduction of effort by using a mixture of three methods (i.e., qualitative, bounding quantitative, and realistic quantitative) that is suggested in Figure C-2.

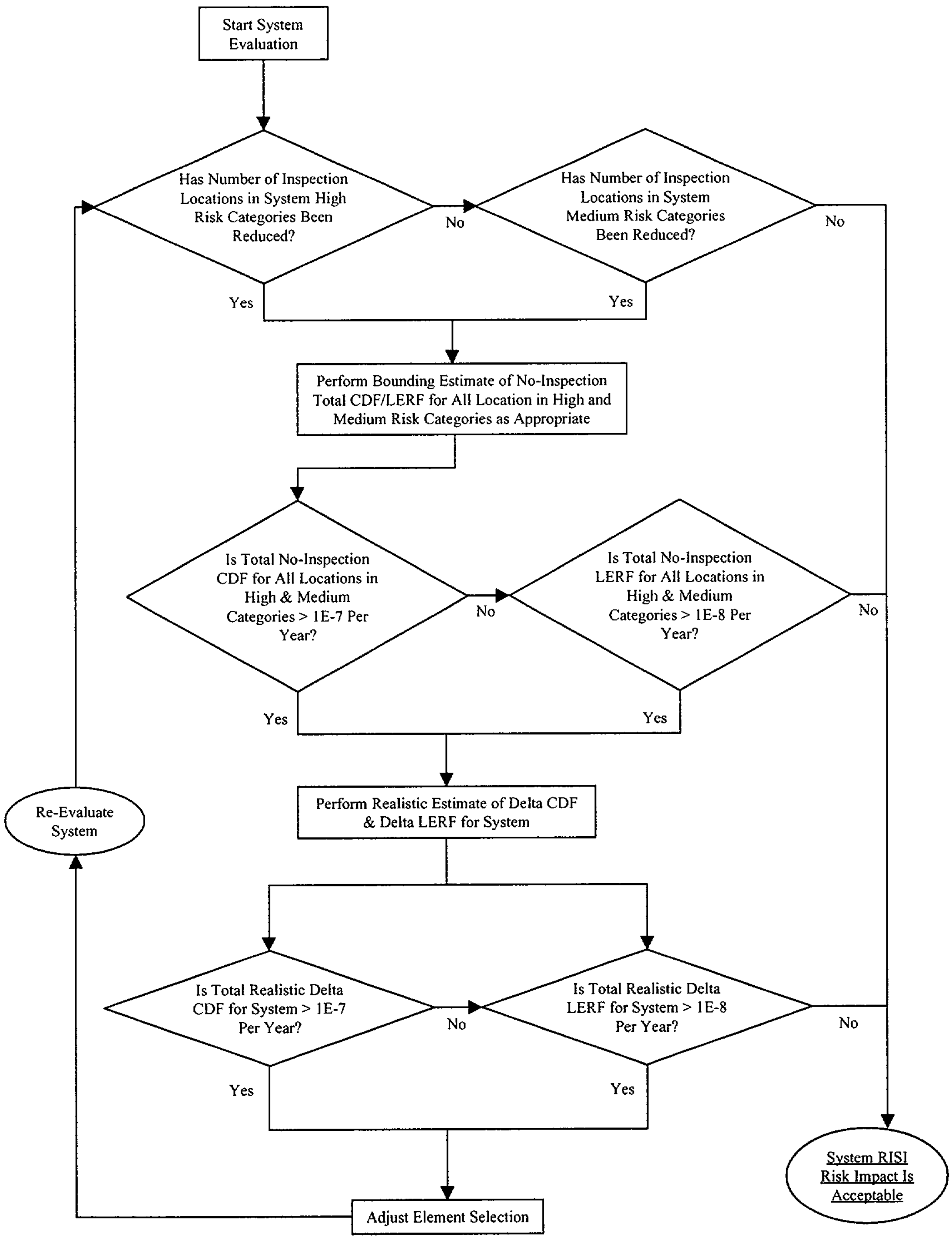


Figure C-2: Flow Chart for Evaluation of Risk Impacts [Reference C-1]

- In future updates of the RI-ISI program, and in the iterations between element selection and risk impact assessment, it is much easier to have the data in place to perform an evaluation in all segments within the scope of the evaluation.

For these reasons, it was decided to perform full realistic risk impact assessments for the entire scope of the formal RI-ISI program, which includes all non-exempt pipes in Class 1 and 2 piping systems. This approach is followed consistently for all Exelon plants as well as the qualitative evaluation steps covered in Figure C-2.

The changes in CDF and LERF associated with changes to the inspection strategy for each system are estimated using the following equations from Reference [C-1]:

$$\Delta CDF = \sum_{i=1}^{N} n_i \lambda_i P_i \langle R|F \rangle \left(I_{i,new} - I_{i,old}\right) CCDP_i \qquad \text{(C.1)}$$

The delta LERF calculations are based on the ΔCDF calculations and the ratio of Conditional Large Early Release Probability (CLERP) to Conditional Core Damage Frequency (CCDP). The CLERP/CCDP ratio is simply multiplied by the ΔCDF value calculated to determine the delta LERF value in each case.

Where:

ΔCDF	=	Change in core damage frequency due to changes in inspection strategy for the system
$\Delta LERF$	=	Change in large early release frequency due to changes in the inspection strategy for the system
i	=	Index for risk segment having the same degradation mechanisms and consequence of pipe ruptures
N	=	Number of risk segments in the system
n_i	=	Number of elements (welds) in risk segment i
λ_i	=	Failure rate for welds in risk segment i (including leak and rupture failure modes) assuming no inspections, estimated from service data
$P_i\langle R \mid F \rangle$	=	conditional probability of rupture given failure of welds in risk segment i assuming no inspections, estimated from service data
$I_{i,new}$	=	inspection effectiveness factor for proposed risk informed inspection strategy for risk segment i, calculated from Markov model
$I_{i,old}$	=	inspection factor for current ASME Section XI based inspection strategy for segment i, calculated from Markov model
$CCDP_i$	=	conditional core damage probability due to pipe ruptures in risk segment i, obtained from Consequence Evaluation (Steps 2A and 2B in Figure 7-1).

C.3 Method of Estimating Model Parameters

The input parameters in Equation (C.1) are estimated as indicated in Table C-1. Weld counts are established from the ISI database in which piping system line numbers have been subdivided into risk segments, i.e., segments with the same degradation mechanism potential and consequence potential. The pipe failure and rupture parameters are estimated using Bayesian failure rate estimation techniques that were specifically developed and approved for use in the EPRI RI-ISI applications, (i.e., References [C-3], [C-4], and [C-5]). To estimate the inspection effectiveness factors, the Markov method is used (see Reference [C-4]). The Markov method derives equations for the inspection effectiveness factors that are in turn dependent on the same failure rates and rupture frequencies and parameters that describe the inspection and leak detection processes. This method was also approved for use by the NRC in the Safety Evaluation (SE) for RI-ISI applications following the EPRI methodology, (see References [C-2], [C-5]). An overview of the Markov model for piping systems is provided in Section C.4 together with documentation of how it was applied to Class 1 and 2 piping systems at Braidwood and Byron Stations.

Table C-1: Method of Quantification of Parameters in Equations (C.1)

PARAMETER	METHOD OF QUANTIFICATION
ΔCDF	Computation of Equation (C.1)
$\Delta LERF$	Calculation based on the ΔCDF calculations and the ratio of CLERP to CCDP
i	From risk segment definition
N	From risk segment definition
n_i	From risk segment definition
λ_i	Estimated from service data
$P_i(R \mid F)$	Estimated from service data
$I_{i,new}$	Markov model solution used to develop equation in terms of parameters that describe degradation and inspection processes as explained in this section
$I_{i,old}$	Markov model solution used to develop equation in terms of parameters that describe degradation and inspection processes as explained in this section
$CCDP_i$	Evaluated using plant specific PRA models and the results of the consequence analysis

C.4 Markov Model for Piping System Reliability

C.4.1 Overview of Markov Model

There are several different approaches that have been applied to estimation of pipe failure frequencies. The most straightforward approach is to obtain statistical estimates of pipe element failure rates, which is the most common approach to this problem, (see References [C-6], [C-7], [C-8]). The primary limitation of a statistical analysis approach is that past historical data reflects some indeterminate impact of previous inspection programs and if we are going to propose changes to these programs, such changes may render the previous failure rate estimates invalid. Another approach is to make use of probabilistic fracture mechanics models to predict crack initiation and growth from existing flaws. Such models reflect our understanding of the physical processes of fracture mechanics but to date have not been fully benchmarked against service experience. To examine an alternative approach and to pursue the objective of keeping the approach practical and useful for utility piping engineers, the concept of Markov models supported by analysis of service experience was pursued.

During a third party review of the original EPRI RI-ISI methodology, (see Reference [C-9]), an idea emerged to utilize an established reliability modeling technique, known as the Markovian technique, to address the impact of inspections on pipe rupture frequencies. The objective of this approach is to explicitly model the interactions between degradation mechanisms and the inspection, detection, and repair strategies that can reduce the probability that failures occur or that failures will progress to ruptures. This Markov modeling technique starts with a representation of a piping "system" in a set of discrete and mutually exclusive states. At any instant in time, the system is permitted to change state in accordance with whatever competing processes are appropriate for that plant state. In this application of the Markov model, the states refer to various degrees of piping system degradation or repairs, i.e., the existence of flaws, leaks, or ruptures. The processes that can create a state change are the failure mechanisms operating on the pipe and the processes of inspecting or detecting flaws and leaks, and repair of damage prior to the progression of the failure mechanism to rupture.

The basic form of a Markov Model for pipe failure and inspection processes is presented in Figure C-3. This model consists of four states of a pipe segment or element (e.g., a weld or section of pipe) reflecting the progressive stages of pipe failure mechanisms: the development of flaws or detectable damage, the occurrence of leaks, and the occurrence of pipe ruptures. As seen in this model, pipe leaks and ruptures are permitted to occur directly from the flaw or leak state, or may also occur in a progression. The model accounts for state dependent failure and rupture processes and two repair processes. Once a flaw occurs, there is an opportunity for inspection and repair to account for the in-service inspection program and other programs that search for signs of degradation prior to the occurrence of pipe failures. When a pipe leak occurs, there is another opportunity for detection and repair prior to the occurrence of a rupture for failure mechanisms that have a "leak before break" characteristic.

The Markov model diagram describes the failure and inspection processes as a discrete state-continuous time problem. It is used to develop a set of differential equations, the solution of which is the time dependent probability of the system occupying each state. For the study of pipe ruptures, state "R" is the failure state of interest. Once the solution is obtained, the hazard rate of the system can be determined. For this example, the hazard rate corresponds to the time dependent frequency or failure rate for pipe ruptures. The time dependent failure rate for ruptures asymptotically converges to a constant value, which is a function only of the parameters of the model. This long-term failure rate or hazard rate is the long-term pipe rupture frequency that determines the long-term risk of pipe ruptures. These parameters are in turn related to the time constants of the underlying processes. The occurrence rates for flaws, leaks, and ruptures are estimated from service data. The occurrence rates for inspections and repairs are estimated based on the characteristics of the inspection process, non-destructive examination (NDE) reliability, time interval of leak detection, and mean time to repair flaws and leaks upon detection. Application of the Markov model can be accomplished based on this steady state hazard, or as a time dependent hazard that varies over the life of the plant.

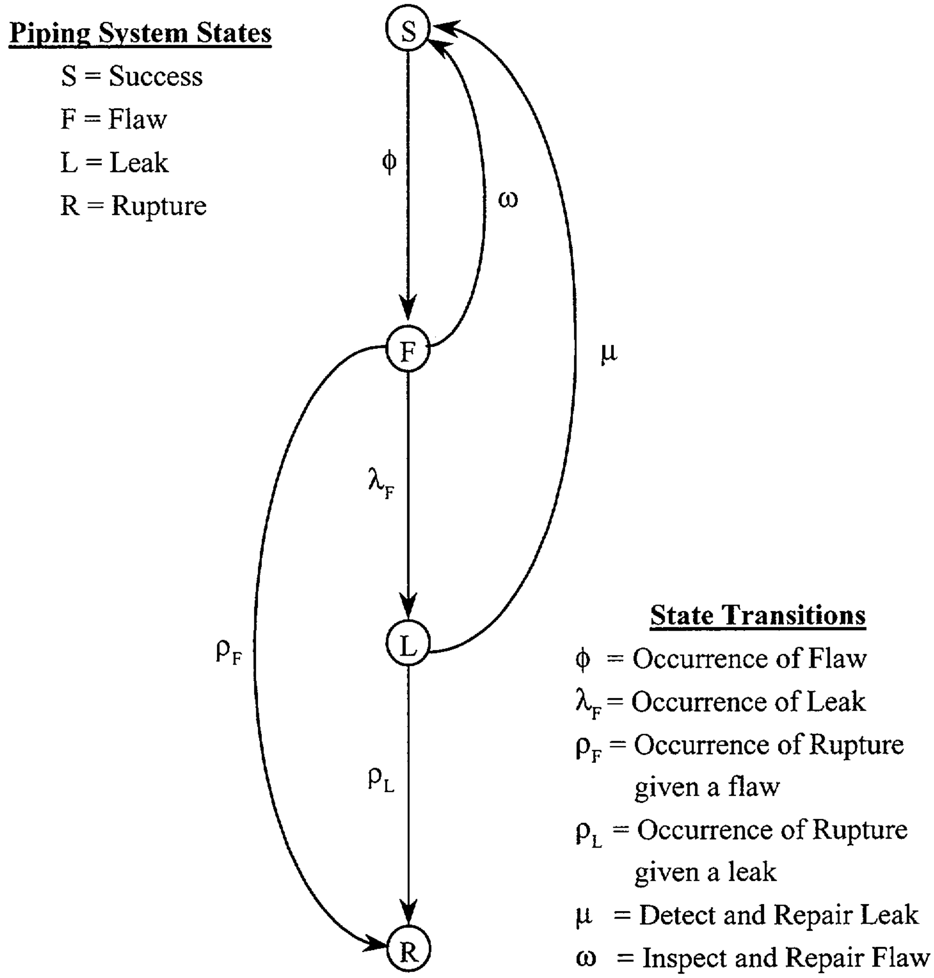


Figure C-3: Markov Model for Pipe Elements with Inservice Inspection and Leak Detection

The Markov models for pipe ruptures are used to set up and solve differential equations for the time dependent state probabilities associated with the model. These equations are based on the assumption that the probability of transition from one state to another is proportional to the transition rates indicated on the diagrams and that there is no memory of how the current state is arrived at. Under the assumption that all the transition rates are constant, the Markov model equations will consist of a set of coupled linear differential equations with constant coefficients. The solution of these differential equations is obtained to compute the time dependent probability that the pipe segment in question is in each state S, F, L, or R. Once these results are obtained, other results such as the system hazard rate that defines the time dependent frequency of pipe ruptures can be developed. This frequency is the form of the result that is needed to support a PSA model of pipe ruptures as initiating events. Details of how this method is developed and solved are provided in Reference [C-4].

Based on insights from service experience, it was decided to use several different models for estimating pipe rupture frequencies depending on the specific failure mechanism. There are several reasons for this. One is that certain mechanisms can be attributed to specific elements of the piping system that are susceptible to failure. These are associated with degradation mechanisms that tend to occur either at specific welds or specific sections of pipe that exhibit the conditions necessary for these failure mechanisms. The applicable damage mechanisms for this type include corrosion, corrosion fatigue, erosion corrosion, erosion-cavitation, stress corrosion cracking, and thermal fatigue. Of these, all except corrosion and erosion corrosion, which do not necessarily occur at welds, tend to occur at or near welds. Hence, estimating pipe rupture failure frequencies in terms of ruptures per susceptible weld or ruptures per susceptible foot of pipe are viable approaches for these failure mechanisms, all of which are damage mechanisms. Another common feature exhibited by these failure mechanisms is that they have demonstrated in the service experience data to show a strong "leak before break" characteristic, i.e., the observed frequency of leak type failure modes is much greater than the rupture type failure mode.

A summary of the different models being used in the EPRI RI-ISI program is provided in Table C-2. The model we use to estimate these degradation type failure mechanisms is referred to as Model A which expressed the pipe rupture frequency in terms of a pipe failure rate or frequency and a conditional probability of pipe rupture given failure. The conditional probability of rupture given failure provides a means of quantifying the "leak before break" characteristics of the failure mechanism. In this model, the service data is broken down to support dependence of the rupture and failure parameters on the reactor vendor, system type, and specific damage mechanism. Model A1 supports estimates in terms of ruptures and failures per susceptible foot of pipe per year for corrosion and erosion corrosion, while Model A2 supports estimates of pipe rupture frequency in terms of ruptures and failures per susceptible weld per year. Model A2 is used in Equation (C.1) as all the piping of interest in this evaluation is subject to the class of degradation mechanisms that occur in welds. Although there are some piping segments in Braidwood and Byron Stations Class 1 and 2 systems subjected to erosion corrosion or Flow Accelerated Corrosion (FAC), the RI-ISI program is not proposing any changes to augmented inspection programs for FAC. Hence, any change in risk for this evaluation will be solely due to weld type degradation mechanisms that may be removed from the ISI program.

The remaining failure mechanisms that have been identified are described as loading conditions and include water hammer, over-pressurization, frozen pipes, and vibration fatigue are not amenable to in-service inspection as a means of failure prevention. Design and construction defects occur at welds and are amenable to ISI in the sense that such errors can be found during NDE type inspections. These loading conditions occur randomly and have the potential to failure or rupture anywhere in a system. Another aspect of the severe loading type failure mechanisms is that at the plant level they exhibit a weak "leak before break characteristic." For these mechanisms, we use rupture data directly to estimate rupture frequencies, and the unit of measurement that is sensible for these are ruptures for system year for different system groups and specific loading conditions. We refer to this approach as Model B.

Table C-2: Failure Rate Models Used for Different Failure Mechanisms

FAILURE MECHANISM CLASS	FAILURE MECHANISM	FAILURE RATE BASIS	FAILURE RATE MODELS EMPLOYED**
Degradation Mechanisms	Corrosion	Failures/pipe-ft-yr.*	Model A1
	Erosion Corrosion		
	Erosion Cavitation	Failures/weld-yr.*	Model A2
	Thermal Fatigue		
	Stress Corrosion Cracking		
	Corrosion Fatigue		
	Design and Construction Defects		
Severe Loading Conditions	Water Hammer	Failures/system-yr.	Models B and C
	Over-pressurization		Model B
	Frozen Pipes		
	Vibrational Fatigue		

* Failure rates applicable only to welds and section of pipe found susceptible to specified damage mechanism

** Model A Freq{Rupture} = Freq{Failure} x Prob{Rupture|Failure} failure and rupture data used in Bayes update of Generic Priors

** Model B Freq{Rupture} developed direct from rupture data and used in Bayes update of Generic Prior

** Model C Freq{Rupture} = Freq{Water Hammer} x Prob{Rupture|Water hammer} used in Bayes update of Generic Priors

A third model was developed to support the particular loading condition of water hammer. While Model B can be used to obtain a kind of average frequency of pipe ruptures due to water hammer, the available data on this mechanism (see Reference [C-11]) supports a more specialized model. This is known as Model C in which pipe ruptures from water hammer are expressed in terms of the frequency of water hammer events, obtained from a special database, and the conditional probability of pipe rupture given a water hammer event.

Models B and C are not used in this evaluation because there is no impact of inspection program changes on these failure mechanisms. While these models are relevant to the task of estimating the total failure rates and rupture frequencies of pipes due to all failure mechanisms, they are not relevant to evaluating changes in failure rates and rupture frequencies. Returning to Equation (C.1), only model A2 is relevant to determination of the inspection effectiveness factor.

C.4.2 Use of the Markov Model to Calculate Inspection Effectiveness Factor

With reference to equation (C.1) the Markov model is used to determine the inspection effectiveness factors, $I_{i,new}$ and $I_{i,old}$, associated with the new (i.e., RI-ISI) and old (i.e., ASME Section XI) inspection programs. Each factor represents the ratio of the rupture frequency with credit for inspections to that given no credit for inspections. Noting the solution of the Markov model is a set of time dependent state probabilities and rupture frequencies, the hazard rate of the Markov model at the end of the 40-year design life is used to determine these factors. More specifically, the inspection factors are defined using:

$$I_{i,new} = \frac{h_{40}\{RISI\}}{h_{40}\{noinsp\}} \quad \text{(C.2)}$$

$$I_{i,old} = \frac{h_{40}\{SecXI\}}{h_{40}\{noinsp\}} \quad \text{(C.3)}$$

Where:

$h_{40}\{RISI\}$ = hazard rate (time dependent rupture frequency) for weld subjected to the RI-ISI inspection strategy

$h_{40}\{SecXI\}$ = hazard rate (time dependent rupture frequency) for weld subjected to the Section XI inspection strategy

$h_{40}\{noinsp\}$ = hazard rate (time dependent rupture frequency) for weld subjected to no in-service inspection

The solutions to the Markov model for time dependent hazard rates are developed in Reference [C-4]. These solutions are developed in terms of closed form analytic solutions that have been applied to applicable systems in Microsoft Excel spreadsheets. Independent reviews have been performed by EdF, the University of Maryland, discussed in Reference [C-4], and Los Alamos National Laboratory, discussed in Reference [C-5]. The hazard rates are a function of time and of the parameters of the Markov model presented in Figure C-3. The quantification of these parameters is discussed in the section below.

C.4.3 Estimation of Markov Model Parameters

As seen in Figure C-3, there are six parameters that are associated with the Markov model, an occurrence rate for detectable flaws, ϕ, a failure rate for leaks given the existence of a flaw, λ_F, two rupture frequencies including one from the initial state of a flaw ρ_F, and one from the initial state of a leak, ρ_L, a repair rate for detectable flaws, ω, and a repair rate for leaks, μ.

The latter two parameters dealing with repair are further developed by the following simple models.

$$\omega = \frac{P_{FI} P_{FD}}{(T_{FI} + T_R)} \quad \text{(C.4)}$$

Where:

P_{FI} = probability that a piping element with a flaw will be inspected per inspection interval. This parameter has a value of 0 if it is not in the inspection program and 1 if it is in the inspection program.

P_{FD} = probability that a flaw will be detected given this element is inspected. This is the reliability of the inspection program and is equivalent to the term used by NDE experts, "Probability of detection (POD). This probability is conditioned on the occurrence of one or more detectable flaws in the segment according to the assumptions of the model. Also note that

T_{FI} = mean time between inspections for flaws, (inspection interval)

T_R = mean time to repair once detected. There is an assumption that any significant flaw that is detected will be repaired. Depending on the location of the weld to be repaired, the weld repair could take on the order of several days to a week. However, since this term is always combined with T_{FI}, and T_{FI} is 10 years, in practice the results are insensitive to assumptions regarding T_R

Similarly, estimates of the repair rate for leaks can be estimated according to:

$$\mu = \frac{P_{LD}}{(T_{LD} + T_R)} \qquad \text{(C.5)}$$

(Note that T_{LI} in this equation was changed to T_{LD})

Where:

P_{LD} = probability that the leak in the element will be detected per leak inspection or detection period

T_{LD} = mean time between inspections for leaks.

T_R = as defined above but for full power applications, this time should be the minimum of the actual repair time and the time associated with any LCO if the leak rate exceeds technical specification requirements.

The values for ω and μ are shown in the following table.

DEGRDATION MECHANISM	ω, μ VALUES		SI SYST.	RC SYST.	RH SYST.	CV SYST.	CS SYST.	MS SYST.	FW SYST.	ESW SYST.
SC	Sect. XI	ω	0.050	0.050	na	na	na	na	na	na
		μ	0.591	0.591	na	na	na	na	na	na
	RI-ISI	ω	0.075	0.075	na	na	na	na	na	na
		μ	0.591	0.591	na	na	na	na	na	na
TF	Sect. XI	ω	0.050	0.050	na	0.050	na	na	0.050	na
		μ	0.591	0.591	na	0.591	na	na	0.591	na
	RI-ISI	ω	0.080	0.080	na	0.080	na	na	0.090	na
		μ	0.591	0.591	na	0.591	na	na	0.591	na
EC	Sect. XI	ω	na	na	na	na	na	na	na	na
		μ	na	na	na	na	na	na	na	na
	RI-ISI	ω	na	na	na	na	na	na	na	na
		μ	na	na	na	na	na	na	na	na
TF + SC	Sect. XI	ω	na	0.050	na	na	na	na	na	na
		μ	na	0.591	na	na	na	na	na	na
	RI-ISI	ω	na	0.080	na	na	na	na	na	na
		μ	na	0.591	na	na	na	na	na	na
No Known Degradation Mechanism	Sect. XI	ω	0.050	0.050	0.050	0.050	0.050	0.050	0.050	0.050
		μ	0.591	0.591	0.591	0.591	0.591	0.591	0.591	0.591
	RI-ISI	ω	0.050	0.050	0.050	0.050	0.050	0.050	0.050	0.050
		μ	0.591	0.591	0.591	0.591	0.591	0.591	0.591	0.591

Now we have developed the root-input parameters of the Markov model, which if quantified will enable us to quantify the inspection effectiveness factors. A summary of the root input parameters of the Markov model and the general strategy for estimation of each one is presented in Table C-3. The specific basis for estimation of each of these parameters for Braidwood and Byron Class 1 and 2 systems is provided in Section C.4.4 below.

C.4.4 Estimation of Markov Model Parameters for Braidwood and Byron Class 1 and 2 Systems

Failure Rates and Rupture Frequencies from the flaw state (λ,ρ_F)

The Markov Model was applied to each system in the scope of the formal RI-ISI evaluation for Braidwood and Byron Units 1 and 2. These systems include those portions of the following systems that contain ASME Class 1 or 2 system:

- Reactor Coolant System (RC, RY, and Thermowells)
- Safety Injection System (SI)
- Chemical and Volume Control System (CV)
- Residual Heat Removal System (RH)
- Containment Spray System (CS)
- Main Steam System (MS)
- Main Feedwater System (FW)
- Essential Cooling Water System (SX)
- Containment Purge and Bypass Piping (VQ)

The first 8 systems were evaluated for CDF and LERF impacts quantitatively using the Markov model to support the estimation of pipe rupture frequencies in the CCDP and CLERP results from the Consequence Assessment. The final system was also evaluated quantitatively for LERF impacts.

Table C-3: Strategy for Estimation of Markov Model Parameters

SYMBOL	PARAMETER DEFINITION	STRATEGY FOR ESTIMATION
ϕ	Occurrence rate of a flaw	Data from results of NDE inspections and service data with cracks; for selected damage mechanisms normally estimated in terms of a multiple of the total failure rate using the argument that there must be at least one flaw to produce a damage mechanism related leak or rupture. The value of ϕ is determined by equation (C.8).
λ_F	Occurrence rate of a leak from a flaw state	Estimated in terms of failure rates conditioned on the susceptibility for the indicated damage mechanism according to the EPRI damage mechanism evaluation criteria. It is assumed that if the element is considered susceptible to a damage mechanism according to the EPRI criteria that there is at least one detectable flaw in the element. Different failure rates are estimated for different systems and damage mechanisms. The value of λ_F is determined by equation (C.7).

Table C-3: Strategy for Estimation of Markov Model Parameters (cont'd)

SYMBOL	PARAMETER DEFINITION	STRATEGY FOR ESTIMATION
ρ_F	Occurrence rate of a rupture from a flaw state	Estimated in terms of rupture frequencies conditioned on the susceptibility for the indicated damage mechanism according to the EPRI damage mechanism evaluation criteria. Different failure rates for different systems and damage mechanisms. It is assumed that if the element is considered susceptible to a damage mechanism according to the EPRI criteria that there is at least one detectable flaw in the element. The values of ρ_F are shown in Table C-4.
ρ_L	Occurrence rate of a rupture from a leak state	This rupture rate occurs during an advanced state of degradation and is normally estimated in terms of the frequency of severe loading conditions such as a water hammer event or overpressure event. The value of ρ_L is 1.97×10^{-2} / system – year. See equation (C.6).
ω	Inspection and repair rate of a flaw state	The value of ω is modeled by equation (C.4) and estimates of P_{FI}, P_{FD}, T_{FI}, and T_R.
μ	Detection and repair of a leak state	The value of μ is modeled by equation (C.5) and estimates of P_{LD}, T_{LD}, and T_R.
P_{FI}	Probability per inspection interval that the pipe element will be inspected	Set to 1 if the element is included in the inspection program, and 0 if not.
P_{FD}	Probability per inspection that an existing flaw will be detected	Estimate based on NDE reliability performance data and difficulty and accessibility of inspection for particular element based on engineering judgement. The values of P_{FD} are shown in Table C-7.
P_{LD}	Probability per detection interval that an existing leak will be detected	Estimate based on system, presence of leak detection systems, technical specifications, and locations and accessibility of element based on engineering judgement. The default value of 0.90 is used.
T_{FI}	Flaw inspection interval, mean time between in service inspections	Normally 10 years for ASME Section XI or RI-ISI piping systems.
T_{LD}	Leak detection interval, mean time between leak detections	Estimate based on method of leak detection; ranges from immediate to frequency of routine inspections for leaks. The default value of 1.5 years is used.
T_R	Mean time to repair the piping element given detection of a critical flaw or leak	Estimate of time to tag out, isolate, prepare, repair, leak test and tag in service; if to be conditioned for at power, can be no longer than technical specification limit for operating with element tagged out of service; normally set to a value of 200 hours.

The results of the degradation mechanism evaluation have found that the piping elements in the above systems have the following possibilities for degradation:

- No degradation mechanism potential
- Thermal fatigue potential (TF, includes TT and TASCSC)
- Stress Corrosion Cracking (SC, includes IGSCC, PWSCC, TGSCC)

- Erosion-corrosion (E/C, includes FAC)
- Erosion cavitation (E-C)
- Corrosion (MIC, Pitting)
- Combinations of two or more of the above mechanisms

The failure rates and rupture frequencies used in the Markov model and in Equations (C.1) have to account for each combination of system and damage mechanism possibility, after excluding those associated with augmented inspection programs that are not being changed in the RI-ISI program. This excludes erosion-corrosion, and corrosion. Hence the failure rates and rupture frequencies that are needed must account for each of the above systems, and elements subject to no degradation, TF, SC, E-C, and combinations of these mechanisms.

The failure rates and rupture frequencies used for each system and damage mechanism combination are tabulated in Reference [C-3]. These failure rates and rupture frequencies are developed from service data, the simple models described in Table C-2, and the Bayes estimation methodology that was developed in Reference [C-3] and approved by the NRC for use in RI-ISI applications in Reference [C-2]. The failure rates and rupture frequencies for data sets broken down by reactor vendor, system group, and failure mechanisms in Reference [C-3] were used for the initial delta risk evaluation for Braidwood Station and Byron Station systems. Reference [C-3] was developed to support the NRC review of the EPRI RI-ISI methodology and supporting research to confirm that the EPRI method would result in acceptable risk impacts for the EPRI pilot studies as documented in Reference [C-4].

A key assumption that is made in application of the failure rates and rupture frequencies to the Markov model is that the conditional failure rates and rupture frequencies given susceptibility to a damage mechanism, which is the basis for the numerical estimates, equals the conditional parameter estimates given the existence of a flaw or crack that exceeds the ASME Section XI repair criteria. This is a reasonable assumption because it is necessary to be susceptible to a damage mechanism to have a flaw or crack by that damage mechanism.

Most pipe segments are found to susceptible to no active damage mechanisms. For these segments, the failure rates and rupture frequencies for design and construction errors are used as the only failure mechanism found from the service data that could be identified in a pipe inspection that was not otherwise known to be subject to a damage mechanism. Note that there are other failure mechanisms that would apply to such locations such as water hammer, vibration fatigue, and others but such mechanism are not amenable to in-service inspections. Only those mechanisms that could be identified in a pipe inspection are appropriate for inclusion in this evaluation, because these are the only mechanisms that could be affected by a change in the inspection program.

For pipe segments that are found to be susceptible to one ISI amenable damage mechanism such as thermal fatigue, stress corrosion cracking, or erosion-cavitation, the failure rates and rupture frequencies for these elements are determined by combining the contributions in Table C-4 from the applicable damage mechanism with those from design and construction errors. This is done since all inspection locations are susceptible to design and construction errors. Hence in this case the failure rates and rupture frequencies are determined by summing the contributions from one ISI amenable damage mechanism and design and construction errors.

For pipe segments that are found to be susceptible to two or more ISI amenable damage mechanisms, the following rules are used: The total failure rate for the element is determined by summing the failure rates of each applicable damage mechanism plus the contribution from design and construction errors and then the resulting sum is multiplied by a factor of three to account for the possibility of synergy between the damage mechanism. For Braidwood and Byron Stations, there were a small number of segments in the RCS that were found to be susceptible to both thermal fatigue and stress corrosion cracking damage mechanism, but with the exception of systems subject FAC as discussed below, there were no other segments in the Class 1 and 2 systems that were susceptible to two or more ISI amenable damage mechanisms. The vast majority of the evaluated segments were found to be susceptible to no ISI amenable damage mechanism, and the remaining ones were only susceptible to one ISI amenable damage mechanism.

The factor of three determined via engineering judgement accounts for the possibility that two or more damage mechanism might influence the propagation of the same flaw or crack. This is viewed as a conservative assumption because no such factor should be applied if the damage mechanisms really act independently. Multiple damage mechanisms have been identified previously in the RISI Pilot Studies and other applications of the EPRI method for RI-ISI, but none of these studies included a factor to account for synergy between damage mechanisms. If the true value of this factor was much larger than the value of three used here, one would expect to see failures attributable to multiple damage mechanisms, but no such failures are evidenced in the service data. Also, the risk impact results are generally not sensitive to this factor since a small number of welds are susceptible to multiple damage mechanisms and the risk impact from any single weld is very, very small.

The RI-ISI program does not impact current augmented programs for corrosion, FAC, or IGSCC. So if a segment is susceptible to a damage mechanism covered in an augmented program, such mechanisms are not included in the failure rates and rupture frequency development describe above. However the possibility of synergy between FAC and another ISI amenable damage mechanism is accounted for as follows. There were some segments in the Feedwater systems that were found to be susceptible to both FAC and thermal fatigue. Since the failure rates and rupture frequencies for thermal fatigue were not amenable to resolution into those with thermal fatigue implications and since there is judged to be some potential for synergy, the failure rates and rupture frequencies for thermal fatigue and design and construction errors were combined and then multiplied by a factor of three. We do not include the FAC contributions, as there are no changes to the FAC program, only to the inspection programs for other ISI amenable damage mechanisms. The factor of three increase is judgmentally assigned to account for the possibility that the failure rate for thermal fatigue could be higher than that inferred from the service data due to wall thinning that could occur in the area of the welds that are subject to thermal fatigue.

The other potential situation of multiple degradation mechanisms that was considered was the case where one of the damage mechanisms is corrosion that is covered in the augmented programs for MIC. The SX system includes segments that are susceptible to MIC but for Braidwood and Byron Stations, there were no other damage mechanisms identified.

<u>Rupture Frequency from the Leak State (ρ_L)</u>

The Markov model of Figure C-3 includes the possibility that a leaking pipe element will remain undetected such that degradation may continue until the damage increases to the point of a rupture.

The probability of this occurrence is reduced by the occurrence of leak inspections with a relatively high probability of leak detection and repair of the pipe. This would be an advanced stage of pipe aging, such the frequency of pipe rupture is expected to be much greater than the case permitted by the Markov model in which ruptures occur from the initial state of a flaw or crack in which case the extent of degradation and aging is less advanced. It is difficult to estimate this parameter, as there are no data on pipe ruptures in which it is known that the pipe element was leaking previously.

However, there is another consideration that needs to be addressed in the estimation of this parameter and that is the fact that in such an advanced state of degradation, a pipe element would be much more susceptible to pipe rupture due to the combination of this degradation and a severe loading condition. Detailed analysis of service data performed by ERIN for EPRI, (see Reference [C-10]) has shown that piping failures have resulted from the following severe loading conditions:

- Water hammer;
- Overpressurization; and
- Frozen pipes.

Of these severe loading conditions, water hammer events are by far the most likely. Overpressurization is very unlikely because of the ASME requirements to protect piping in Class 1 and 2 systems with safety and relief valves and the likelihood of challenging pipes beyond the design basis of the relief valves is very small. Frozen pipes are only credible at certain U.S. sites in the wintertime in areas that are exposed to the outside atmosphere, which are not very likely with Class 1 and 2 systems. It is recognized that plants like Braidwood and Byron Stations use freeze seals to isolate sections of pipe for maintenance activities; however, freeze seals would not be applied at a section of pipe that was leaking. When freeze seals are used, it is a short section of pipe that is actually frozen to provide the seal. Therefore, the use of freeze seals is judged to not contribute to the frequency of pipe rupture given leak. On the other hand, water hammer events have occurred in practically all Classes 1 and 2 systems including the reactor coolant system (pressurizer spray lines, for example). The rupture frequency given the initial state of a leak, ρ_L, is conservatively estimated to be equal to the frequency of water hammer events that occur in piping systems. A study performed by Stone and Webster for EPRI, (see Reference [C-11]) collected data on reported water hammer events in U.S. commercial nuclear plants through 1991. In this study a total of 283 water hammer events were reported over a period of about 1,200 reactor years of experience. Using an estimate of 12 piping system per plant which is consistent with the estimate provided in Reference [C-3], the following point estimate of ρ_L is obtained:

$$\rho_L = \frac{283}{(12)(1200)} = 1.97x10^{-2} / system - year \qquad (C.6)$$

<u>Frequency of Flaws (ϕ)</u>

In the Markov model, flaws are defined as degradation that has progressed to the point of meeting the repair criteria in Section XI of the ASME code because once the flaw state of the Markov model is occupied, the model assumes that the element will be repaired if the flaw is detected.

Estimates of the frequency of flaws are determined from the same service data that is used to develop the failure rates and rupture frequencies. The service data used in development of rupture frequencies, and flaw occurrence rates is based on Reference [C-3]. This data includes cracks, pinhole leaks, leaks, and evidence that over this data set there have been no reported pipe ruptures, which are defined as failures with leak flow rates in excess of 50 gpm.

Table C-4: Mean Failure Rates, Conditional Rupture Probabilities, and Rupture Frequencies Used in Braidwood and Byron Risk Impact Assessment

DAMAGE MECHANISM	PARAMETER*	SYSTEM							
		RCS	SIS	CVCS	RHRS	CS	SX	FWC	ST
Thermal Fatigue (TF)	λ_f	9.90E-06	1.31E-06	6.53E-05	1.31E-06	1.67E-06	6.25E-05	4.16E-05	5.12E-06
	$P\langle R\|F\rangle$	5.56E-02	5.56E-02	3.53E-02	5.56E-02	3.53E-02	3.53E-02	3.53E-02	3.53E-02
	ρ_F	5.50E-07	6.89E-08	2.32E-06	6.89E-08	5.89E-08	2.20E-06	1.47E-06	1.80E-07
Stress Corrosion Cracking(SC)	λ_f	1.76E-04	9.83E-05	1.84E-04	9.83E-05	4.20E-04	2.88E-05	4.07E-05	9.64E-07
	$P\langle R\|F\rangle$	1.89E-02	1.89E-02	1.15E-02	1.89E-02	1.15E-02	1.15E-02	1.15E-02	1.15E-02
	ρ_F	3.31E-06	1.83E-06	2.12E-06	1.83E-06	4.84E-06	3.31E-07	4.71E-07	1.09E-08
Erosion-Cavitation (E-C)	λ_f	5.20E-06	2.79E-06	8.26E-06	2.79E-06	4.17E-06	3.08E-05	1.95E-05	1.28E-06
	$P\langle R\|F\rangle$	5.56E-02	5.56E-02	3.53E-02	5.56E-02	3.53E-02	3.53E-02	3.53E-02	3.53E-02
	ρ_F	2.94E-07	1.57E-07	2.90E-07	1.57E-07	1.47E-07	1.08E-06	6.89E-07	4.49E-08
Design Construction Defects(DC)	λ_f	1.78E-05	6.46E-07	2.87E-06	6.46E-07	1.36E-07	1.78E-06	6.89E-07	8.16E-07
	$P\langle R\|F\rangle$	4.76E-02	4.76E-02	1.95E-01	4.76E-02	1.95E-01	1.95E-01	1.95E-01	1.95E-01
	ρ_F	8.45E-07	3.11E-08	5.60E-07	3.11E-08	2.60E-08	3.48E-07	1.34E-07	1.59E-07
Basis for Estimates: Reference [C-3]									

* Failure rates, λ_F, and rupture frequencies, ρ_F, given in units of events/weld-year, conditional rupture probabilities, $P\langle R|F\rangle$ are dimensionless

We note that each leak or rupture that is found in the database that resulted from a particular degradation mechanism must have resulted from a flaw that progressed to the state of the rupture. Hence there is at least one flaw for each of the observed leaks and ruptures in the database. In addition, there are additional events denoted in this table in which cracks of a sufficient size to cause the need for repair of the pipe occurred and were repaired before an opportunity to failure occurred. There may have been additional flaws undetected that were created by this same experience, due to the fact that most welds are not ever inspected, and even when welds are inspected, the NDE process may have overlooked some flaws. Based on this reasoning, we take the view that the flaw occurrence rate is best estimated as a multiple of the rate of occurrence of failures, which include both leaks and ruptures.

In general, a point estimate of the frequency of pipe failure events, λ, is given by the following expression:

$$\lambda = \frac{n_F}{NT} \tag{C.7}$$

Where:

n_F = the number of failure events including both leaks and ruptures in the service data (See Table C-6)
T = the total time over which failure events were collected
N = the number of components that provided the observed pipe failures

A point estimate of the total frequency of flaws (cracks and leaks), ϕ, is given by the following expression:

$$\phi = \frac{n_C}{N \cdot T \cdot f \cdot P_{FD}} + \frac{n_F}{NT} = \frac{n_C}{N \cdot T \cdot f \cdot P_{FD}} + \lambda \tag{C.8}$$

Where:

n_C = the number of crack events (See Table C-6)
f = the fraction of welds inspected for flaws (See equation C.10)
P_{FD} = the probability that an expected weld will find an existing flaw (See Table C-7)

The other variables in Equation (C.8) are as defined above. In this equation we account for the observed cracks in the database and the fact that only a fraction of the welds in the database are inspected for this condition and that those found are subject to a finite NDE reliability. This equation also reflects the fact that each failure in the database has an additional crack that produced the failure, whose exposure parameter is the entire population of welds at risk for failure. This is based on the insight that nearly all failures are found not from NDE inspections but from independent observations. This is an important observation because all the population of welds in the surveyed data are at risk for failure observation, but only a small fraction are at risk for the observation of cracks which can only be found from NDE inspections.

If we now take the ratio of ϕ *to* λ, we get an expression for the factor by which to multiply the pipe failure rate to obtain the flaw rate:

$$R_{C/F} = \frac{\phi}{\lambda} = \frac{n_C}{n_F \cdot f \cdot P_{FD}} + 1 \tag{C.9}$$

Where:

$R_{C/F}$ = Number of cracks or flaws per pipe failure:

Point Estimates of $R_{C/F}$ for different data sets in Reference [C-3] are presented in Table C-6. The fraction of welds inspected listed in this table is estimated as follows. The current ASME Section XI requirements are to inspect 25% of the Class 1 welds and 7.5% of the Class 2 welds and these inspection requirements call for the same welds to be inspected each inspection interval. When cracks or significant flaws are found, the ASME code requires that an expanded search be made; however, the frequency of flaws and failures is so rare that this requirement adds very few additional inspections. Using data from Braidwood Station on the number of Class 1 and Class 2 welds of 1605 and 1800, respectively, which is assumed to be representative of the relative populations of Class 1 and 2 welds in this service data, the following estimate of the parameter *f* is obtained:

$$f = \frac{1605(.25)+1800(.075)}{(1605+1800)} = .157 \tag{C.10}$$

Table C-6: Estimates of the Crack to Leak Ratio for Various Damage Mechanisms in PWR Plants

PARAMETER		DAMAGE MECHANISM				
		SC	D&C	TF	NON-SC	ALL
Number of Cracks, n_c		28	6	8	16	43
Number of Failures, n_F		71	17	18	50	106
Fraction of welds inspected, *f*		0.157	0.157	0.157	0.157	0.157
$R_{C/F}$	P_{FD}=.50	6.01	5.48	6.64	6.08	6.03
	P_{FD}=.75	4.34	3.99	4.76	4.39	4.35
	P_{FD}=.90	3.78	3.49	4.14	3.82	3.80

Hence, even though the observed number of cracks in Reference [C-3] is only 40% of the observed number of failures, the flaw occurrence rates are actually much higher than the weld failure rates. The estimates for the ratio of flaws to total pipe failures obtained in Table C-6 reflect the different degrees to which pipe welds are exposed to the class of events that have been reported. The evidence for the observed crack frequency is based on an exposed weld population that is only about 16% of the exposed weld population for pipe failures as only the inspected welds are available to produce this evidence. This fact combined with the additional implicit crack that must have existed prior to each of the pipe leak events, creates an underlying failure rate for cracks that is at least four times higher than the underlying failure rate for leaks and ruptures.

<u>Probability per inspection interval that the pipe element will be inspected (P_{FI})</u>

As noted in Table C-3, this parameter is set to 1.0 if the element is selected for inspection and 0.0 if it is not. Since the Markov model is evaluated separately for the Section XI and RI-ISI programs, this parameter is set to the appropriate value for each weld for of the inspection programs.

<u>Probability per inspection that an existing flaw will be detected (P_{FD})</u>

These values are set based on engineering judgement to reflect the probability that an inspected weld with a crack or flaw that exceeds the critical flaw size will be detected in each in-service inspection. The values used in the Braidwood Station and Byron Station risk impact assessment for different situations are listed in Table C-7. Each value is scaled by a factor F_A, which is the fraction of the weld that is accessible. This value is normally 1.0 as the accessibility of the weld is one of the factors that is taken into account in the element selection process. The values assigned in this evaluation are those in the existing ISI database at Exelon. As experience in performing the RI-ISI inspections is accumulated, these values are subject to change and should be updated in the next RI-ISI program update. It is emphasized however, that as explained more fully in Reference [C-4], the Markov results are not sensitive to small variations in this parameter in the range of 0.7 to 1.0. For welds in the existing ISI program, the factor F_A is conservatively set to 1.0, which will tend to overstate the importance of the existing inspections.

Table C-7: Estimation of the Probability of Detection of Inspected Elements with Flaws, P_{FD}

APPLICABILITY	ASSUMED VALUE OF P_{FD}	BASIS
EPRI RISI of Element in Carbon Steel pipe subject to thermal fatigue	$P_{FD} = .90*F_A$; F_A = fraction of element that is accessible to inspection	EPRI RISI procedure calls for expanded inspection zone for elements susceptible to TF, assumption used in NRC reviewed Markov applications, References [C-2] and [C-4]
EPRI RISI of element in Stainless steel pipe subject to thermal fatigue	$P_{FD} = .80*F_A$; F_A = fraction of element that is accessible to inspection	Carbon steel value reduced slightly to reflect insights from EPRI NDE qualification program, Reference [C-12]
EPRI RISI of element subject to other damage mechanism subject to inservice inspection	$P_{FD} = .75*F_A$; F_A = fraction of element that is accessible to inspection	Inspection for cause principle expected to pick up most flaws above critical size but no expanded volumes as in TF
EPRI RISI of element subject to design and construction errors only	$P_{FD} = .50*F_A$; F_A = fraction of element that is accessible to inspection	Since there is no inspection for cause principle to apply, high confidence in detection cannot be assured
Section XI ISI of element due to (unknown) damage mechanism	$P_{FD} = .50*F_A$; F_A = fraction of element that is accessible to inspection	Since there is no inspection for cause principle to apply, high confidence in detection cannot be assured

<u>Probability per detection interval that an existing leak will be detected (P_{LD}) and Leak Detection Interval (T_{LD})</u>

The following default values are used for all segments in this evaluation:

$$P_{LD} = .90 \tag{C.11}$$

$$T_{LD} = 1.5 \text{ years} \tag{C.12}$$

These values are considered to be conservative for the following reasons. Some leaks from the RCS, and CVCS will be instantaneously alarmed in the control room due to high radiation levels in the containment. Other leaks will be picked up in operator walk-arounds that occur either hourly or once per shift according to the procedures. Still other leaks will be detected rather promptly via sump alarms. Hence only some leaks need to wait for the system leak test to become visible to the plant personnel. While leaks in some locations may be difficult to detect, most leaks will be identified well within the 1.5 years assumed.

<u>Flaw inspection interval, mean time between in service inspections(T_{FI})</u>

This parameter is fixed in the ASME Code Section XI to once per 10 years. The risk informed procedure used in this evaluation proposes no change in this inspection interval. If in the future a different inspection interval is selected, this parameter can easily be changed in the Markov model calculations.

<u>Mean time to repair the piping element given detection of a critical flaw or leak(T_R)</u>

This parameter is set to 200 hours, which translates into a little over 8 days. This is a conservative value for most situations, but as discussed in Reference [C-4] the results of the Markov model are not sensitive to this parameter in the slightest. Increasing this value to 1000 hours would not change the results appreciably since this term must be compared to the mean time between pipe failures which in a given pipe location is typically thousands of years.

REFERENCES

[C-1] EPRI, "Revised Risk Informed Inservice Inspection Procedure," EPRI Report No. TR-112657 Rev. B, July 1999.
[C-2] USNRC Staff Evaluation of EPRI RISI Procedure (included in Reference [C-1]).
[C-3] T.J. Mikschl and K.N. Fleming, "Piping System Failure Rates and Rupture Frequencies for Use in Risk Informed Inservice Inspection Applications", EPRI TR-111880, September 1999. EPRI Licensed Material.
[C-4] K.N. Fleming, et al, "Piping System Reliability and Failure Rate Estimation Models for Use in Risk Informed Inservice Inspection Applications", EPRI TR-110161, Prepared by ERIN Engineering and Research Inc. for EPRI, December, 1998 EPRI Licensed Material.
[C-5] H. Martz, et al., "Final (Revised) Review of EPRI Proposed Markov Modeling/Bayesian Updating Methodology for Use in Risk-Informed Inservice Inspections of Piping in Commercial Nuclear Power Plants," TSA-1/99-164, June 1999.
[C-6] Thomas, H.M., "Pipe and Vessel Failure Probability," Reliability Engineering, Vol. 2, pp. 83-124, 1981.
[C-7] Jamali, K., "Piping Failures in U.S. Commercial Nuclear Power Plants," EPRI Interim Report TR-100380, 1992.
[C-8] Jamali, K., "Piping Failures in U.S. Commercial Nuclear Power Plants," EPRI Report TR-102266, 1993.
[C-9] K.N. Fleming, D.E. True and C.G. Sellers, "Independent Review of EPRI Risk Informed Inservice Inspection Procedure," Prepared by ERIN Engineering and Research Inc. for EPRI, July 1996.
[C-10] Gosselin, Stephen R. and Karl N. Fleming, "Evaluation of Pipe Failure Potential Via Degradation Mechanism Assessment," Proceedings of ICONE 5, 5th International Conference on Nuclear Engineering, 1997, Nice, France.
[C-11] Stone and Webster Engineering Corporation, "Water Hammer Prevention, Mitigation, and Accommodation - Volume 1: Plant Water Hammer Experience," prepared by SWEC for Electric Power Research Institute, EPRI NP-6766, July 1992.
[C-12] Personal communication with Steve Gosselin, Ron Gamble and Jack Cole, April 19, 2000.