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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

Subject: 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

References:
(1) Electric Power Research Institute (EPRI) Topical Report (TR) 112657 Rev. B-A, "Revised Risk-Informed Inservice Inspection Evaluation Procedure"

(2) W. H. Bateman (U.S. NRC) to G. L. Vine (EPRI) letter dated October 28, 1999, transmitting "Safety Evaluation Report Related to EPRI Risk-Informed Inservice Inspection Evaluation Procedure (EPRI TR-112657, Revision B, July 1999)"

In accordance with 10 CFR 50.55a(a)(3)(i), we request approval of a proposed alternative to the existing American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel 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. The alternative proposed utilizes the Reference 1 methodology for a Risk-Informed Inservice Inspection program approved by the NRC to the extent and within the limitations specified in Reference 2.

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The discussion contained in Attachment 2 demonstrates that the proposed alternative will provide an acceptable level of quality and safety. The format of Byron Station's Risk-Informed inservice inspection submittal is consistent with the Nuclear Energy Institute and industry template developed for applications of the Risk-Informed inservice inspection methodology. Additional supporting documentation is available at Byron Station for NRC review.

We intend to incorporate the risk-based approach to the selection and examinations of Class 1 and Class 2 piping welds in the second period of the second interval for Byron Station, Unit 1 and the first period of the second interval for Byron Station, Unit 2. For this reason, we are requesting approval of the proposed alternative by March, 2001.

Please direct any questions you may have regarding this submittal to Ms. P. Reister, Regulatory Assurance Manager, at (815) 234-5441, extension 2280.

Sincerely,

William Levis
Site Vice President
Byron Station

WL/JL/dpk

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

2) Risk-Informed Inservice Inspection Evaluation – Byron Station Units 1 and 2, Program Summary

ATTACHMENT I

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

RELIEF REQUEST I2R-40
Revision 0

COMPONENT IDENTIFICATION

Code Class:	1 and 2
Examination Category:	B-F, B-J, C-F-1, and C-F-2
Examination Item Numbers:	B5.10, B5.40, B5.70, B9.11, B9.21, B9.31, B9.32, B9.40, C5.11, C5.21, C5.30, C5.41, C5.51, and C5.81
Description:	Class 1 pressure retaining dissimilar metal welds and pressure retaining welds in piping. Class 2 pressure retaining welds in austenitic stainless steel and carbon steel piping.
Component Number:	All welds in ASME Section XI Code Categories B-F, B-J, C-F-1, and C-F-2
References:	1) Electric Power Research Institute (EPRI) Topical Report (TR) 112657 Rev. B-A, "Revised Risk-Informed Inservice Inspection Evaluation Procedure" 2) W. H. Bateman (U.S. NRC) to G.L. Vine (EPRI) letter dated October 28, 1999 transmitting "Safety Evaluation Report Related to EPRI Risk-Informed Inservice Inspection Evaluation Procedure (EPRI TR-112657, Revision B, July 1999)" 3) Risk-Informed Inservice Inspection Evaluation – Byron Station Units 1 and 2

CODE REQUIREMENT

Table IWB 2500-1, Examination Category B-F, items B5.10, B5.40, and B5.70, requires a volumetric and surface examination on all welds.

Table IWB 2500-1, Examination Category B-J, requires a volumetric and surface examination for items B9.11 and B9.31 and a surface exam for items B9.21, B9.32, and B9.40 for those welds selected per the following:

a. All terminal end welds in each pipe or branch run connected to vessels.

b. All terminal end welds and joints in each pipe or branch run connected to other components where the stress levels exceed either of the following limits under loads associated with specific seismic events and operational conditions.
   (1) Primary plus secondary stress intensity range of $2.4S_m$ for ferritic steel and austenitic steel.
   (2) Cumulative usage factor U of 0.4.

c. All dissimilar metal welds between combinations of:
   (1) Carbon or low alloy steels to high alloy steels
   (2) Carbon or low alloy steels to high nickel alloys
   (3) High alloy steels to high nickel alloys

CODE REQUIREMENT (continued)

d. Additional piping welds so that the total number of circumferential butt welds (or branch connection or socket welds) selected for examination equals 25% of the circumferential butt welds (or branch connection or socket welds) in the reactor coolant piping system.

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

a. 7.5% but not less than 28 welds per Category of all welds not exempted by IWC-1220 (as supplemented by Code Case N-408-2). The examinations shall be distributed as follows:
   (1) The examinations shall be distributed among the Class 2 systems prorated, to the degree practicable, based on the number of non-exempt welds in each system.
   (2) Within each system, the examinations shall be distributed among terminal ends and structural discontinuities prorated, to the degree practicable, based on the number of nonexempt terminal ends and structural discontinuities in that system.

b. Within each system, examinations shall be distributed between line sizes prorated to the degree practicable

CODE REQUIREMENTS FROM WHICH RELIEF IS REQUESTED

Relief is requested from the ASME Section XI examination methodology and the selection criteria of IWB and IWC 2500-1 for Code Items B5.10, B5.40, B5.70, B9.11, B9.21, B9.31, B9.32, B9.40, C5.11, C5.21, C5.30, C5.41, C5.51, and C5.81.

BASIS FOR RELIEF

Pursuant to 10 CFR 50.55a(a)(3)(i), relief is requested on the basis that the proposed alternative would provide an acceptable level of quality and safety.

In reference 2, the NRC concludes that the proposed Risk-Informed Inservice Inspection program as described in EPRI TR-112657, Revision B, is a sound technical approach and will provide an acceptable level of quality and safety pursuant to 10 CFR 50.55a for the proposed alternative to the piping inservice inspection requirements with regard to the number of locations, locations of inspections, and methods of inspection.

PROPOSED ALTERNATE PROVISIONS

The proposed alternative, as described in Attachment 2, "Risk-Informed Inservice Inspection Program Summary - Byron Station Units 1 and 2," provides an acceptable level of quality and safety as required by 10 CFR 50.55a(a)(3)(i).

Byron Station's application of the Risk-Informed Inservice Inspection Plan, per the EPRI Topical Report, Reference (1), requires that 25% of the elements that are categorized as "High" risk (Risk Category 1, 2, or 3) and 10% of the elements that are categorized as "Medium" risk (Risk Categories 4 and 5) be selected for inspection. For this application, the guidance for the examination volume for a given degradation mechanism is provided by the EPRI Topical Report while the guidance for the examination method is provided by Code Case N-578-1.

In addition, all piping components, regardless of risk classification, will continue to receive Code-required pressure and leak testing, as part of the current ASME Section XI program. VT-2 visual examinations are scheduled in accordance with the station's pressure and leak test program, which remains unaffected by the Risk-Informed Inservice Inspection Program.

PERIOD FOR WHICH RELIEF IS REQUESTED

Relief is requested for remainder of the second ten-year inspection interval of the Inservice Inspection Program for Byron Station Unit 1 and Unit 2.

ATTACHMENT 2

Risk Informed Inservice Inspection (RISI) Program Summary

Byron Station Unit 1 and Unit 2

EXECUTIVE SUMMARY

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EXECUTIVE SUMMARY

1. INTRODUCTION

The objective of this submittal is to request the use of a risk-informed inservice inspection (RISI) program for Class 1 and Class 2 piping that is currently inspected as part of the American Society of Mechanical Engineers (ASME) Section XI based inservice inspection (ISI) program, as an alternative to the 1989 Edition of the ASME Section XI requirements for the remainder of the second inspection interval. The risk-informed process used in this submittal is described in the Electrical Power Research Institute (EPRI) RISI Topical Report (Reference 1). To strengthen the technical basis for this RISI program beyond the minimum requirements implied by the EPRI RISI Topical Report, a number of enhancements were made to the process that are described in the paragraphs below.

We plan to incorporate the RISI inspection program during the second period of the second inspection interval for Unit 1 and the first period of the second interval for Unit 2. The second 10-year inspection interval began for Units 1 and 2 on June 30, 1996, and August 16, 1998, respectively.

As a risk-informed application, this submittal meets the intent and principles of Regulatory Guides 1.174, "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions on Plant-Specific Changes to Licensing Basis" and 1.178, "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions on Inservice Inspection of Piping," as well as those set forth in the EPRI RISI Topical Report.

<u>Probability Risk Assessment (PRA) Quality</u>

The NRC Staff reviewed the Byron IPE relative to the requirements in NRC Generic Letter 88-20. The NRC Staff Evaluation Report issued in December 1997 stated that "...the staff finds that the licensee's IPE is complete with regard to the information requested by GL 88-20 (and associated guidance, NUREG-1335) and concludes that the licensee's IPE process meets the intent of GL 88-20." The staff concluded in its summary, "The licensee explicitly addressed the staff's concerns in the modified IPE submittal." The staff did note that Common Cause Factors (CCF) were lower than generic and suggested enhancements to the containment analysis while stating it was consistent with the intent of GL 88-20. ComEd has since enhanced the Byron PRA by incorporating generic CCF data from the NRC-sponsored database in NUREG/CR-5497 and by modifying the containment analysis to follow the guidance given in NUREG/CR-6595 which is explicitly accepted for regulatory applications in Regulatory Guide 1.174.

ComEd has significantly upgraded the Byron PRA since the NRC Staff Evaluation Report on the IPE was issued in December 1997. This upgrade of the Byron PRA was done in conjunction with an upgrade of the Braidwood PRA and has produced PRAs of comparable quality. Byron and Braidwood Nuclear Plants are sister plants with nearly identical designs. Hence, the Byron and Braidwood PRA models are very similar and exist in an integrated PRA model. ComEd had essentially the same personnel working on each of the PRA upgrades and these personnel performed common enhancements to both plants PRAs. The upgrades include conversion to linked fault tree models, updating initiating events data, addition of special initiators, revision of the human reliability analysis, updating equipment failure rate and unavailability data, a comprehensive update of CCF treatment, and major improvements in equipment dependancy logic.

A Westinghouse Owners Group (WOG) PRA Peer Certification Review was conducted on the upgraded Braidwood PRA in September 1999. Based on the results of this certification review, both the Braidwood and Byron PRAs were further upgraded. The NRC reviewed the results of this upgrade using ComEd's response to an RAI relative to ComEd's License Amendment Request (LAR) for extension of the allowed outage times (AOTs) of Byron's and Braidwood's emergency diesel generators (EDGs). The NRC Safety Evaluation (SE) dated September 1, 2000 approving the LAR states "the staff agrees that the licensee responded adequately to all of the certification team's findings by making appropriate changes to the PRA model." This conclusion was independently reviewed by a WOG PRA Peer Certification Review of the Byron PRA in July 2000.

The WOG PRA certification process assesses a PRA in eleven functional elements. Each element is graded on a scale of 1 to 4. A grade 3 indicates "that risk significance determinations made by the PRA are adequate to support regulatory applications, when combined with deterministic insights." A grade of 4 indicates that the PRA "is usable as a primary basis for developing licensing positions...", however, "it is expected that few PRAs would currently have many elements eligible for this grade." The Byron PRA was graded 3 in ten of the PRA elements and 4 in the eleventh (based on draft certification report; final report pending). Two of the grades 3 and the grade 4 were contingent on further PRA enhancements, principally documentation, which have negligible impact on application of the PRA to this Risk Informed ISI Relief Request.

ComEd maintains and updates each of its PRAs to be representative of the respective as-built, as-operated plant. A PRA Maintenance and Update Procedure formalizes the PRA update process. The procedure defines the process for regular and interim updates for issues identified as potentially affecting the PRA. This process assures the present PRA reflects the current plant configuration and plant procedures.

Based on the results of the NRC Staff reviews and the WOG PRA Certification Peer Reviews cited above, ComEd believes that the level of detail and quality of the Byron PRA fully supports the Byron Risk Informed ISI Relief Request

2. PROPOSED ALTERNATIVE TO CURRENT ISI PROGRAM REQUIREMENTS

2.1 ASME Section XI

ASME Section XI Categories B-F, B-J, C-F-1, and C-F-2 currently contain the requirements for examining these Class 1 and Class 2 piping components via Non Destructive Examination (NDE) methods.

2.2 Alternate RISI Program

The alternative RISI program for piping is described in the EPRI RISI Topical Report. The RISI program will be substituted for the 1989 ASME Section XI Code Edition examination program for Class 1 Category B-F and B-J welds and Class 2 Category C-F-1 and C-F-2 welds in accordance with 10 CFR 50.55a(a)(3)(i) by alternatively providing an acceptable level of quality and safety. Other portions of the ASME Section XI Code outside of this scope will be unaffected. The EPRI Topical Report provides the requirements for defining the relationship between the risk-informed examination program and the remaining unaffected portions of ASME Section XI.

2.3 Augmented Programs

As discussed in Section 6 of the EPRI Topical Report, certain augmented inspection programs may be integrated into the RISI program. Per Table 6-2 of the EPRI Topical Report, the issues raised by NRC Bulletins 88-08 and 88-11 and Information Notice 93-20 are all addressed by the evaluation of thermal fatigue that is part of the degradation assessment for RISI. Also, the issue of pipe cracks in stagnant borated water systems (IE Bulletin 79-17) is addressed by the evaluation of stress corrosion cracking that is part of the degradation assessment for RISI. These augmented programs for non-exempt piping segments are therefore subsumed in the RISI program. The following augmented programs were not subsumed into the RISI program and remain unaffected:

- Service Water Integrity Program (G.L. 89-13)
- Flow Accelerated Corrosion (FAC) (G.L. 89-08)
- High Energy Line Breaks (USNRC Branch Technical Position MEB 3-1)

Elements in the scope of this evaluation that were also covered by these augmented programs were included in the consequence assessment, degradation assessment, and risk categorization evaluations, to determine whether the affected piping was subject to damage mechanisms other than those addressed by the augmented program. 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. In the Main Feedwater System, many of the elements covered by the FAC program were also assessed for the potential for other damage mechanisms that are evaluated as part of the EPRI RISI methodology.

2.4 Multiple Damage Mechanisms

The vast majority of pipe elements that were evaluated in the RISI evaluation were found to be susceptible to none of the damage mechanisms addressed in the EPRI RISI methodology. A number of elements were found to be susceptible to one specific damage mechanism, and a relatively small number were identified to be subject to the potential for two or more damage mechanisms. Specific examples are welds in the Main Feedwater System that are subject to both FAC and thermal fatigue, as well as welds in the Reactor Coolant and Safety Injection systems that have the potential for both stress corrosion cracking and thermal fatigue. If one of the damage mechanisms was FAC, the element was assigned to the High failure potential category to be consistent with the EPRI Topical Report. If that assignment led to the decision to select that element for inspection in accordance with the 25% sampling requirement, it was retained in the FAC program for inspection for FAC as well as inspected for the remaining damage mechanism as part of the RISI program. 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.

3. RISK-INFORMED ISI PROCESS

The process used to develop the RISI program is consistent with the methodology described in the EPRI Topical Report for ASME Code Case N-578-1 applications. The process involves the following steps:

- Definition of RISI Program Scope
- Consequence Analysis
- Degradation Mechanism Assessment

- Risk Categorization
- Inspection Location Selection and NDE Selection
- Program Relief Requests
- Risk Impact Assessment
- Implementation and Monitoring Program

3.1 Definition of RISI Program Scope

The systems to be included in the RISI program are provided in Table 1. This scope covers ASME Class 1 and 2 piping systems within the scope of the existing ASME Section XI inspection program. The as-built and as-operated isometric and piping and instrumentation diagrams and additional plant information were used to define the system boundaries. The RISI evaluation system boundaries were defined using the system boundaries established in the existing plant ISI program.

3.2 Consequence Analysis

The consequences of pressure boundary failures were evaluated and ranked based on their impact on conditional core damage probability (CCDP) and conditional large early release probability (CLERP). The impact on these measures due to both direct and indirect effects was determined using the PRA model described in Section 1. Consequence categories (High, Medium, Low, or None) were assigned according to Table 3-1 of the EPRI RISI Topical Report. One of the enhancements that was incorporated into this application of the EPRI RISI methodology was the direct use of the PRA models to support the estimation of CCDP and CLERP values for each pipe element in the scope of the RISI evaluation, in lieu of the consequence tables in the EPRI Topical Report. This step was taken to reduce some of the conservatisms inherent in the consequence tables and to support a more complete and realistic quantification of the risk impacts of the RISI program in comparison with previous applications of this methodology. Another motivation was to increase consistency with other risk informed applications that directly utilize the plant-specific PRA models.

3.3 Degradation Mechanism Assessment

Failure potential was assessed using the deterministic criteria in the EPRI Topical Report to evaluate the potential for each damage mechanism that an ISI exam could identify, and supported by industry failure history, plant-specific failure history, and other relevant information. These failure estimates were determined using the guidance provided in the EPRI Topical Report.

Table 2 summarizes the degradation mechanism assessment by system for each damage mechanism that was identified as a potential failure cause. In addition, failure rates and rupture frequencies were assessed for each piping element within the scope of the RISI evaluation using information in Reference 6 as well as updated failure rate and rupture frequency estimates that were developed as part of this risk informed evaluation and described in the Tier 2 documentation (Reference 5).

3.4 Risk Categorization

In the preceding steps, each element within the scope of the RISI program was evaluated to determine the consequences of its failure, as measured by CCDP and CLERP. Each element was also evaluated to determine its potential for pipe rupture based on the potential for degradation mechanisms that were identified. The results of the consequence assessment

were then combined with the results of the degradation assessment, using the risk matrix shown in Figure 1. This provides a risk ranking and risk category for each element.

POTENTIAL FOR PIPE RUPTURE PER DEGRADATION MECHANISM SCREENING CRITERIA	**CONSEQUENCES OF PIPE RUPTURE** IMPACTS ON CONDITIONAL CORE DAMAGE PROBABILITY AND LARGE EARLY RELEASE PROBABILITY			
	NONE	**LOW**	**MEDIUM**	**HIGH**
HIGH FLOW ACCELERATED CORROSION	**LOW** **Category 7**	**MEDIUM** **Category 5**	**HIGH** **Category 3**	**HIGH** **Category 1**
MEDIUM OTHER DEGRADATION MECHANISMS	**LOW** **Category 7**	**LOW** **Category 6**	**MEDIUM** **Category 5**	**HIGH** **Category 2**
LOW NO DEGRADATION MECHANISMS	**LOW** **Category 7**	**LOW** **Category 7**	**LOW** **Category 6**	**MEDIUM** **Category 4**

Figure 1

EPRI RISI Matrix for Risk Ranking of Pipe Segments (Reference 1)

The results of this evaluation in terms of the number of elements in each of the EPRI RISI risk categories per system are summarized in Table 3 and Table 4 for Unit 1 and Unit 2, respectively.

3.5 Inspection Location Selection and NDE Selection

In general, an ASME Code Case N-578-1 application of RISI, per the EPRI RISI Topical Report, requires that 25% of the elements that are categorized as "High" risk (Risk Category 1, 2, or 3) and 10% of the elements that are categorized as "Medium" risk (Risk Categories 4 and 5) be selected for inspection and appropriate non-destructive examination (NDE). 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 selected. During the selection process, an attempt is made to ensure that all damage mechanisms and all combinations of damage mechanisms are represented in the elements selected for inspection. An element ranking process was used to incorporate several factors into the selection of specific elements to satisfy the above sampling percentages. These factors include whether the element has been previously selected for ISI exams, whether previous exams had indications of possible damage, presence of radiation fields in the vicinity of the elements, accessibility of the element for inspection, and numerical estimates of the pipe rupture frequencies at these locations. The results of the selection are presented in Tables 5 and 6 for Units 1 and 2, respectively. Section 4 of the EPRI Topical

Report was used as guidance in determining the examination requirements for these locations. As noted above, elements found to be susceptible to two or more damage mechanisms were given enhanced treatment, which resulted in a higher priority for element selection.

Longitudinal welds are considered subsumed with examination of the associated circumferential weld when the circumferential weld is selected for RISI examination. This approach, which was approved under Code Case N-524 (approved for use in Regulatory Guide 1.147), is considered to be an acceptable alternative method of scheduling and examining longitudinal welds.

In addition, all in-scope piping components, regardless of risk classification, will continue to receive Code-required pressure and leak testing, as part of the current ASME Section XI program. VT-2 visual examinations are scheduled in accordance with the station's pressure and leak test program, which remains unaffected by the RISI program.

Examination Method and Parts Examined

Code Case N-578-1 Table 1, "Examination Category R-A, "Risk-Informed Piping Examinations" will be used in conjunction with of Table 4-1 of EPRI TR-112657 to provide supplemental guidance for the examination method applicable to socket welds. N-578-1 allows a VT-2 examination of socket welds to be performed each refuel outage in lieu of a volumetric or surface examination, regardless of the degradation mechanism. Byron Station believes the VT-2 examination method is a more meaningful examination method when the nature of the flaw propagation and the socket weld configuration are considered.

Code Case N-578-1 Table 1, "Examination Category R-A, "Risk-Informed Piping Examinations" will also be used in conjunction with Table 4-1 of EPRI TR-112657 to categorize the parts examined under the RI-ISI program. Code Case N-578-1 Table 1 provides examination requirements, examination method, acceptance standards, examination extent and frequency for piping structural elements not subject to a damage mechanism.

Additional Examinations

For High and Medium Risk category piping structural elements (Categories 1 through 5), the following criteria will be used:

(a) Examinations performed that reveal flaws or relevant conditions exceeding the referenced acceptance standards shall be extended to include additional examinations. The additional examinations shall include piping structural elements with the same postulated failure mode and the same or higher failure potential.

(1) The number of additional elements shall be the number of piping structural elements with the same postulated failure mode originally scheduled for that fuel cycle.

(2) The scope of the additional examinations may be limited to those high safety-significant piping structural elements within systems, whose materials and service conditions are determined by an evaluation to have the same postulated failure mode as the piping structural element that contained the original flaw or relevant condition.

(b) If the additional required examinations reveal flaws or relevant conditions exceeding the referenced acceptance standards, the examination shall be further extended to include additional examinations.

(1) These examinations shall include all remaining piping elements whose postulated failure modes are the same as the piping structural elements originally examined.

(2) An evaluation shall be performed to establish when those examinations are to be conducted. The evaluation must consider failure mode and potential.

(c) For the inspection period following the period in which the original examination discovering the flaw or relevant condition was completed, the examinations shall be performed as originally scheduled.

3.6 Program Relief Requests

An attempt has been made to provide a minimum of >90% coverage (per Code Case N-460) when performing the risk-informed examinations. However, some limitations will not be known until the examination is performed, because some locations may be examined for the first time by the specified techniques.

In instances where a location may be found at the time of the examination that does not meet the >90% coverage requirement, the process outlined in the EPRI Topical Report will be followed.

Byron Station has reviewed all approved relief requests and recognizes that there exist differences in the examination category designation and associated item numbers between the standard Section XI program and the RI-ISI program. It is judged that the differences in the category and associated item numbers between the two programs are strictly editorial and do not nullify the intent of the Section XI relief requests as they apply to the RI-ISI program. Therefore, these approved relief requests are considered applicable to RI-ISI and remain effective through the end of the second interval. Byron Station will reevaluate and resubmit, if necessary, Section XI relief requests using the RI-ISI category for the remainder of the Second Interval ISI Program Plan.

In addition, Byron's 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 relief request I2R-40.

3.7 Risk Impact Assessment

The RISI program has been conducted in accordance with Regulatory Guide 1.174 and the EPRI methodology requirements, which were intended to result in a risk decrease, a risk neutral condition, or at most, a very small increase in risk as measured by changes in CDF and LERF.

The risk impact assessment performed in this RISI application included a comprehensive qualitative evaluation as well as a comprehensive quantitative evaluation of the changes in CDF and LERF due to changes in the ISI program for each piping segment in the scope of the RISI evaluation. This is another enhancement that was made that goes well beyond the limited quantitative analyses that are needed to implement the method described in the EPRI Topical Report.

Individual elements were evaluated for consequence and degradation mechanism and then assigned to a risk category and risk ranking as part of the risk characterization step. For the purposes of the risk impact evaluation, elements were combined into risk segments. As a result of this process, each element of a given risk segment has the same qualitative potential for pipe failure according to the potential applicable damage mechanisms and the same consequences as called for in the EPRI RISI Topical Report. The risk segments were then grouped by system and the changes in risk for each risk segment were evaluated qualitatively by noting increases and decreases in the number of exams and for the potential for increases in the NDE probability of detection where the "inspection for cause" principle was applied. Then, each segment was quantified in terms of changes in failure frequency, rupture frequency, CDF, and LERF.

Per Section 3.7.2 of EPRI TR-112657, the Markov piping reliability analysis method (Reference 1) was used to estimate the change in risk due to adding and removing locations from the inspection program. The actual CCDP and CLERP values calculated for each element in the consequence assessment was used in the risk impact calculation. Realistic quantitative estimates of failure frequencies, rupture frequencies, and risk impacts were performed for all segments and elements within the scope of the RISI evaluation, in lieu of the qualitative analysis and bounding risk estimates that are permitted under most circumstances in the EPRI RISI Topical Report.

The changes to the ISI program include changing the number and location of inspections within the risk segment and in many cases improving the effectiveness of the inspection to account for the results of the RISI degradation mechanism assessment. For example, for locations subject to thermal fatigue, examinations are to be conducted on an expanded volume and are to be focused to enhance the probability of detection (POD) during the inspection process. For other damage mechanisms, this "inspection for cause" principle is also expected to apply.

Limits are imposed by the EPRI methodology (TR-112657) to ensure that the change in risk of implementing the RISI program meets the requirements of Regulatory Guides 1.174 and 1.178. The criteria established require that the cumulative increase in CDF and LERF be less than 1E-7 and 1E-8 per year per system, respectively. Meeting these limits is consistent with meeting Regulatory Guide 1.174 risk significant thresholds of 1E-6 and 1E-7 per year for changes in CDF and LERF for a full plant scope RISI application.

The technical basis for the Markov model input parameters that were used in this evaluation are documented in the Tier 2 documentation (Reference 5). These parameters include a set of failure rates and rupture frequencies for piping systems in Westinghouse PWR plants subject to several degradation mechanisms that were identified for these systems as part of the degradation mechanism assessment. The failure rates and rupture frequencies that were used in this evaluation are tabulated in Table 7 and include those developed as part of this evaluation, and documented in the Tier 2 documentation, for the RCS, SI, RHR, and CVCS systems, and those developed in EPRI TR-111880 (Reference 7) for the remaining systems. The updated failure rates and rupture frequencies developed in this evaluation account for

service data in Westinghouse PWRs covering over 2,000 reactor years of experience, and reflect estimates of weld population exposure for Westinghouse Class 1 and 2 systems that were not available when EPRI TR-111880 was developed. The Bayes failure rate estimation methodology that was reviewed and approved by the NRC Staff for use in RISI applications was used to develop the updated failure rates and rupture frequencies listed in Table 7.

A separate Markov calculation for the change in LERF was performed for lines connected to the reactor coolant system (RC) that continue outside containment. The risk impact evaluation included the portion of these lines just outside containment up to the first isolation valve. This calculation was performed so that pipe elements whose failure could create a potential bypass concern were factored into the LERF evaluation. These results were combined with LERF results that were generated from the Byron Station PRA LERF models that account for containment isolation, bypass, and severe accident concerns that are independent of the pipe rupture effects to obtain an overall quantification of LERF impacts. Unlike previous applications of the EPRI methodology, realistic estimates of CDF and LERF contributions and changes in CDF and LERF due to all changes in the RISI program were quantified for all pipe elements, in addition to a qualitative evaluation that is part of the EPRI procedure.

The results of the risk impact assessment for each system at Byron Station Unit 1 are summarized in Table 8 and key aspects are plotted in Figures 2 and 3 for comparison against the risk significant criteria established in the EPRI RISI Topical Report. A similar set of results is presented in Table 9 and Figures 4 and 5 for Unit 2. As seen in these figures and tables, most of the systems evaluated across the two reactor units exhibited small decreases in CDF and LERF. The exceptions to this at Unit 1 were no changes to CDF and LERF in the Essential Service Water System, and small increases in CDF and LERF in the Reactor Coolant, Main Steam, and Feedwater Systems. The exceptions at Unit 2 were no change in the Essential Service Water System, and a small increase in the Reactor Coolant and Main Steam Systems. In each applicable case, the estimated increases in CDF and LERF are much smaller than the risk acceptance criteria by a large margin.

Even though the reactor units are very similar and have the same baseline PRA results, there were small differences in the risk impact assessment due to small differences in the weld populations, and different initial ISI program element selections that resulted in differences in the RISI element selections. The conclusions regarding the acceptability of the numerical risk impacts are the same for both units. The total change in CDF and LERF due to the combined changes in the RISI program for the entire scope of Class 1 and 2 systems are small in relation to Regulatory Guide 1.174 risk significance criteria.

As a sensitivity case, an evaluation was performed assuming that all NDE exams were removed from the ISI program, indicating that the EPRI RISI risk significance thresholds still would not be exceeded.

As indicated above, the risk impact evaluation has demonstrated that no significant risk impacts will occur from implementation of the RISI program for the entire scope of Class 1 and 2 piping that was included in this evaluation. This satisfies the risk significance criteria of Regulatory Guide 1.174 and the EPRI RISI Topical Report.

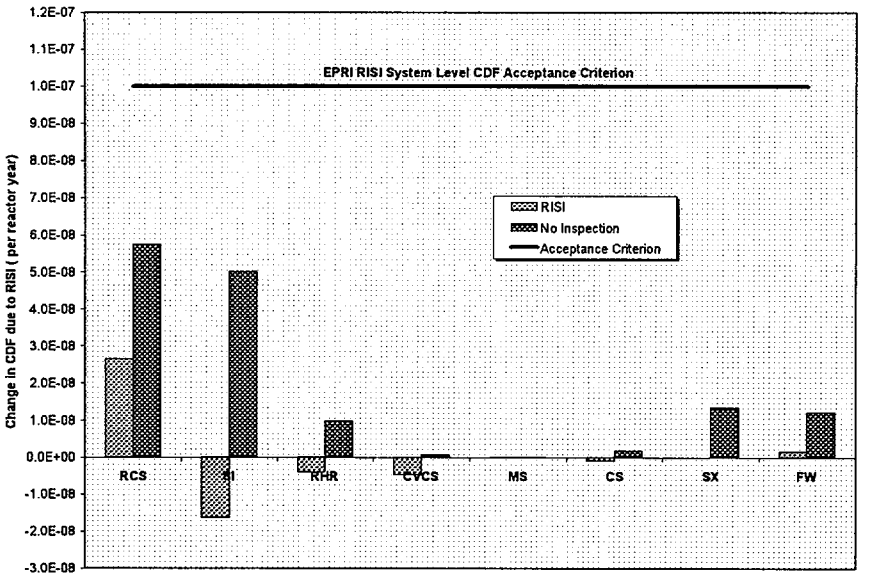

Figure 2

Change in Pipe Rupture CDF for Byron Station Unit 1 Systems

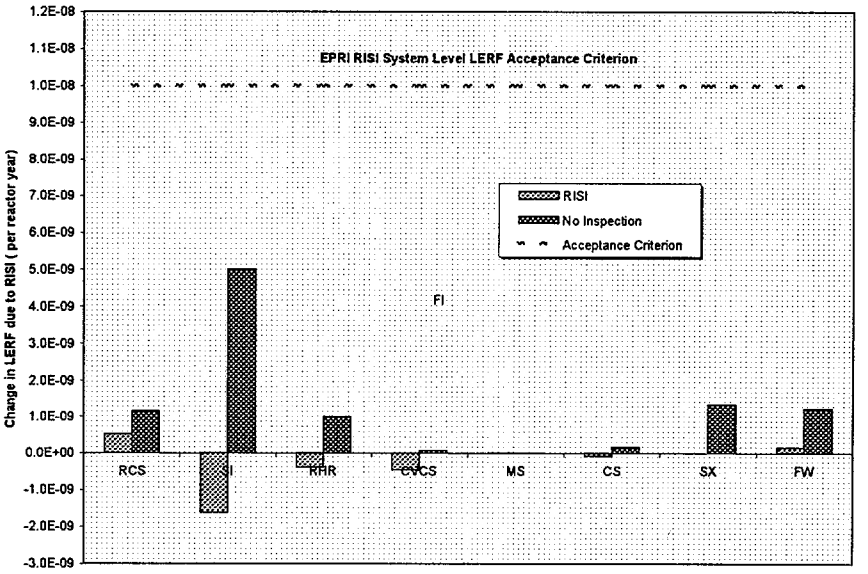

Figure 3

Change in Pipe Rupture LERF for Byron Station Unit 1 Systems

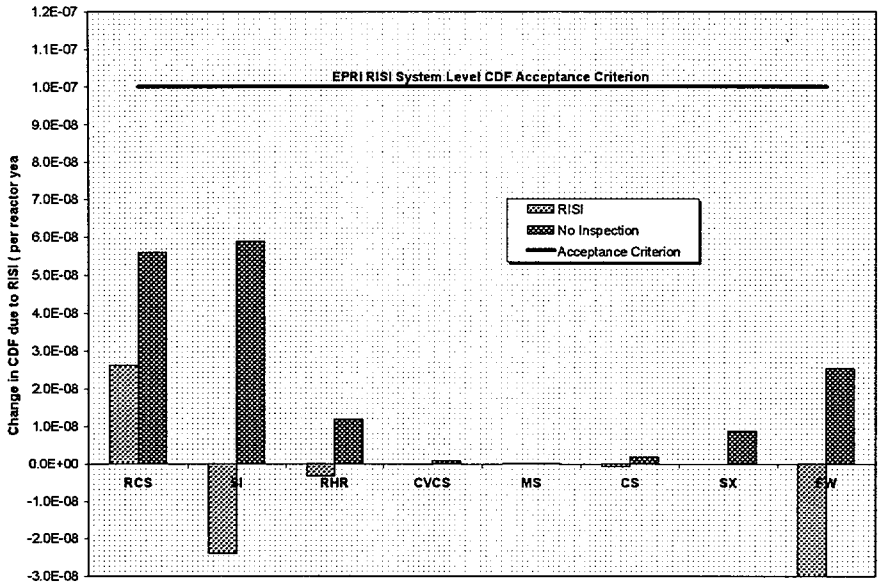

Figure 4

Change in Pipe Rupture CDF for Byron Station Unit 2 Systems

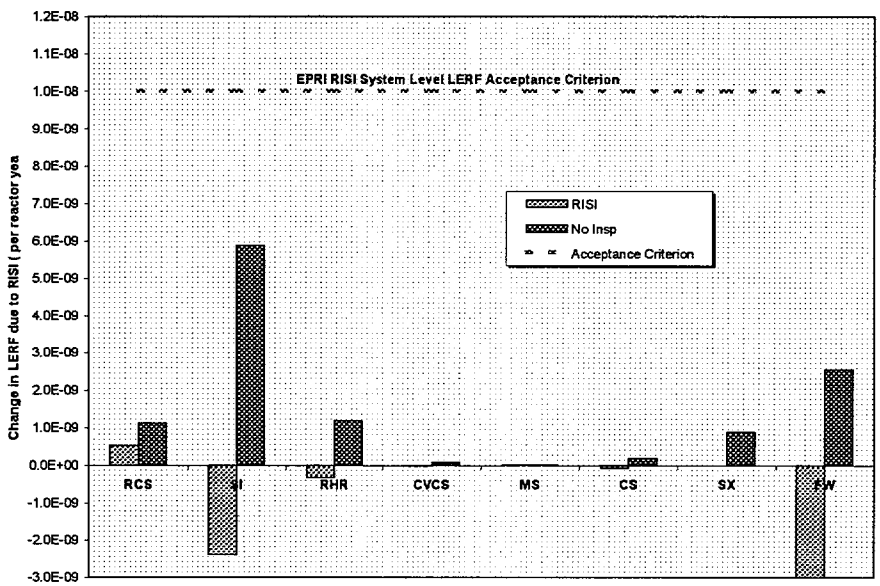

Figure 5

Change in Pipe Rupture LERF for Byron Station Unit 2 Systems

Defense-In-Depth

The intent of the inspections mandated by ASME Section XI for piping welds is to identify conditions such as flaws or indications that may be precursors to leaks or ruptures in a system's pressure boundary. Currently, the process for picking inspection locations is based upon structural discontinuity and stress analysis results. As depicted in ASME White Paper 92-01-01 Rev. 1, "Evaluation of Inservice Inspection Requirements for Class 1, Category B-J Pressure Retaining Welds," this method has been ineffective in identifying leaks or failures. EPRI TR-112657 and ASME Code Case N-578-1 provide a more robust selection process founded on actual service experience with nuclear plant piping failure data.

This process has two key independent ingredients: (1) a determination of each location's susceptibility to degradation and (2) an independent assessment of the consequence of the piping failure. These two ingredients assure defense-in-depth is maintained. First, by evaluating a location's susceptibility to degradation, the likelihood of finding flaws or indications that may be precursors to leak or ruptures is increased. Secondly, the consequence assessment effort has a single failure criterion. As such, no matter how unlikely a failure scenario is, it is ranked High in the consequence assessment, and no lower than Medium in the risk assessment (i.e., Risk Category 4), if, as a result of the failure, there is no mitigative equipment available to respond to the event. In addition, the consequence assessment takes into account equipment reliability, with less credit given to less reliable equipment.

All locations within the reactor coolant pressure boundary will continue to receive a system pressure test and visual VT-2 examination as currently required by the Code regardless of its risk classification.

4. IMPLEMENTATION AND MONITORING PROGRAM

Upon approval of the RISI program, procedures that comply with the guidelines described in EPRI RISI Topical Report will be prepared to implement and monitor the program. The new program will be integrated into the second inservice inspection interval for Byron Station Units 1 and 2. No changes to the Updated Final Safety Analysis Report are necessary for program implementation.

The applicable aspects of the ASME Code not affected by this change are to be retained, such as inspection methods, acceptance guidelines, pressure testing, corrective measures, documentation requirements, and quality control requirements. Existing ASME Section XI program implementing procedures are to be retained and modified to address the RISI process, as appropriate.

The RISI program is a living program requiring feedback of new relevant information to ensure the appropriate identification of high safety significant piping locations. Such relevant information would include major updates to the Byron Station Units 1 and 2 PRA models which could impact both the risk characterization and risk impact assessments, any new trends in service experience with piping systems at Byron Station and across the industry, and new information on element accessibility that will be obtained as the risk informed inspections are implemented. As a minimum, risk ranking of piping segments and element selections will be reviewed and adjusted on an ASME ISI interval basis. In addition, changes may occur more frequently as directed by NRC Bulletin or Generic Letter requirements, or by industry and plant-specific feedback.

5. PROPOSED ISI PROGRAM PLAN CHANGE

A comparison between the RISI program and 1989 ASME Section XI Code Edition program requirements for in-scope piping is provided in Table 5 and Table 6 for Unit 1 and Unit 2, respectively. The number of exams at Unit 1 is reduced from 550 Section XI program exams to 253 RISI program exams, a net reduction of 297 exams (54% reduction in number of exams). Unit 2 is reduced from 533 exams to 276 exams, a net reduction of 257 exams (48% reduction in number of exams). The numbers of exams added to and removed from the ISI program in the "High" and "Medium" risk categories of the EPRI RISI risk ranking process serve to explain the qualitative nature of the risk impact assessment for each system as described in the previous tables and figures.

The Reactor Coolant System at Units 1 and 2 and the Main Feedwater System at Unit 1 exhibited small increases in CDF and LERF due to net reduction in exams located in High and Medium risk segments. The Main Steam System at Units 1 and 2 exhibited very small increases in CDF and LERF due to a reduction in exams in "Low" risk segments. The Essential Services Water System at both units had no changes in the RISI program and have no change in CDF or LERF. The remaining systems at both units exhibited small decreases in CDF and LERF due to a net increase in the number of exams in High or Medium risk segments combined with enhanced "Probability of Detection" due to the inspection for cause principal. As noted previously, these changes in CDF and LERF are well within the EPRI RISI risk significance thresholds.

6. REFERENCES

1. EPRI, "Revised Risk-Informed Inservice Inspection Evaluation Procedure," TR-112657, Rev. B-A, December 1999.
2. ComEd Calculation BYR99-040, CDF Results for Byron Station Units 1 and 2.
3. ComEd Calculation BYR99-096, LERF Results for Byron Station Units 1 and 2.
4. Letter from R.A. Krich to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information related to *Request for Amendment to Technical Specifications - Extension of Allowable Completion Times and Surveillance Requirement Change for Emergency Diesel Generators*," July 7, 2000.
5. ComEd Risk Informed Inservice Inspection Evaluation, Byron Station Nuclear Power Plant Units 1 and 2 – Final Report, August 2000.
6. 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 1988. *EPRI Licensed Material*
7. 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*

Table 1
System Selection and Segment Definition for Unit 1 / Unit 2

System Description[1]	Number of Segments
Containment Spray System (CS)	12 / 12
Chemical and Volume Control System (CV)	40 / 42
Reactor Coolant System (RC, RY, PZR)[2]	134 / 125
Residual Heat Removal System (RH)	18 / 18
Safety Injection System (SI)	88 / 88
Main Feedwater System (FW)	16 / 16
Main Steam System (MS, SG)	24 / 24
Essential Service Water System (SX)	44 / 44
Containment Purge System (VQ)	5 / 5
Total	381 / 374

1. The RISI evaluation boundaries for the Chemical and Volume Control System (CV), Reactor Coolant System (RC), Residual Heat Removal System (RH), and the Safety Injection System (SI) were defined consistent with the system boundaries established in the existing inservice inspection (ISI) program.
2. Includes pressurizer relief piping.

NOTE: This table shows the number of pipe segments from each system that are Class 1 or Class 2 category B-F, B-J, C-F-1, C-F-2. The number of segments is shown for each unit.

Table 2
Failure Potential Assessment Summary for Unit 1 and Unit 2

System	Thermal Fatigue		Stress Corrosion Cracking				Localized Corrosion			Flow Sensitive	
	TASCS	TT	IGSCC	TGSCC	ECSCC	PWSCC	MIC	PIT	CC	E-C	FAC
CS											
CV		X									
RC	X	X	X								
RY	X	X				X					
RH											
SI	X	X	X								
FW	X	X									X
MS											X
SX							X	X			
VQ											

TASCS – thermal stratification, cycling and stripping, TT – thermal transients, IGSCC – intergranular stress corrosion cracking, TGSCC – transgranular stress corrosion cracking, ECSCC – external chloride stress corrosion cracking, PWSCC – primary water stress corrosion cracking, MIC – microbiologically influenced corrosion, PIT – pitting, CC – crevice corrosion, E-C – erosion-cavitation, FAC – flow accelerated corrosion

NOTE: This table shows the assessed failure mechanisms for each system. The RISI Program addresses the cumulative impact of all mechanism that were identified in each system.

Table 3
Number of Elements (Welds) by Risk Category for Unit 1

	High Risk			Medium Risk		Low Risk	TOTAL
System	**Category 1**	**Category 2**	**Category 3**	**Category 4**	**Category 5**	**Category 6 or 7**	**All Categories**
CS				172			172
CV				192	153	124	469
RC		166		404		27	597
RY		43		80			123
RH				200		91	291
SI				278	146	519	943
FW			160				160
MS			8			173	181
SX		282					282
VQ						24	24
TOTAL	0	491	168	1326	299	958	3242

NOTE: This table shows the results of the Risk Categorization for Unit 1. The risk categories are defined in Figure 3-4 of EPRI TR-112657 (Reference 1).

Table 4
Number of Elements (Welds) by Risk Category for Unit 2

	High Risk			Medium Risk		Low Risk	TOTAL
System	**Category 1**	**Category 2**	**Category 3**	**Category 4**	**Category 5**	**Category 6 or 7**	**All Categories**
CS				164			164
CV				198	150	122	470
RC		165		377		13	555
RY		37		81			118
RH				215		99	314
SI				262	160	510	932
FW			274				274
MS			8			163	171
SX		293					293
VQ						24	24
TOTAL	0	495	282	1297	310	931	3315

NOTE: This table shows the results of the Risk Categorization for Unit 2. The risk categories are defined in Figure 3-4 of EPRI TR-112657 (Reference 1). The difference in results as compared to Unit 1 are mainly due to the steam generator replacement on Unit 1, with some minor differences due to slight differences in the number of welds in other systems.

Table 5
Number of Inspections by Risk Category for Unit 1

System	High Risk						Medium Risk				Low Risk		TOTAL	
	Category 1		Category 2		Category 3		Category 4		Category 5		Category 6 or 7		All Categories	
	Sec. XI	RISI	Sec. XI	RISI	Sec. XI	RISI	Sec. XI	RISI	Sec. XI	RISI	Sec. XI	RISI	Sec. XI	RISI
CS							12	18				0	12	18
CV							14	20	61	16	57	0	132	36
RC			53	42			94	41			2	0	149	83
RY			24	11			21	8				0	45	19
RH							14	20			19	0	33	20
SI							24	28	24	17	69	0	117	45
FW					48	32					1	0	49	32
MS											12	0	12	0
VQ											1	0	1	0
TOTAL	0	0	77	53	48	32	179	135	85	33	161	0	550	253

NOTES: (1) This table provides a comparison of the RISI element selection to the original ASME Section XI program. The total number of inspections is significantly lower for the RISI program. Some RISI inspection locations are new when compared to the Section XI program, i.e., they were previously not addressed.

(2) SX inspections will be in accordance with the Service Water Inspection Program.

Table 6
Number of Inspections by Risk Category for Unit 2

System	High Risk						Medium Risk				Low Risk		TOTAL	
	Category 1		Category 2		Category 3		Category 4		Category 5		Category 6 or 7		All Categories	
	Sec. XI	RISI	Sec. XI	RISI	Sec. XI	RISI	Sec. XI	RISI	Sec. XI	RISI	Sec. XI	RISI	Sec. XI	RISI
CS							12	17				0	12	17
CV							16	20	51	15	31	0	98	35
RC			47	41			94	39				0	141	80
RY			22	10			14	9				0	36	19
RH							17	22			19	0	36	22
SI							21	26	43	16	95	0	159	42
FW					38	61						0	38	61
MS											12	0	12	0
VQ											1	0	1	0
TOTAL	0	0	69	51	38	61	174	133	94	31	158	0	533	276

NOTE: This table provides the same information as Table 5 for Unit 2.

Table 7
Mean Failure Rates, Conditional Rupture Probabilities, and Rupture Frequencies
Used in Byron Station Risk Impact Assessment

Damage Mechanism	Parameter*	System							
		RCS	SIS	CVCS	RHRS	CS	SX	FWC	ST
Thermal Fatigue (TF)	λ_f	5.31E-07	1.45E-05	5.40E-06	1.07E-05	1.67E-06	6.25E-05	4.16E-05	5.12E-06
	$P\langle R\|F\rangle$	4.26E-02	4.19E-02	4.26E-02	4.19E-02	3.53E-02	3.53E-02	3.53E-02	3.53E-02
	ρ_F	2.26E-08	6.08E-07	2.30E-07	4.48E-07	5.89E-08	2.20E-06	1.47E-06	1.80E-07
Stress Corrosion Cracking (SC)	λ_f	1.39E-04	4.58E-04	1.15E-04	8.11E-05	4.20E-04	2.88E-05	4.07E-05	9.64E-07
	$P\langle R\|F\rangle$	4.04E-03	4.10E-03	4.03E-03	4.23E-03	1.15E-02	1.15E-02	1.15E-02	1.15E-02
	ρ_F	5.62E-07	1.88E-06	4.64E-07	3.43E-07	4.84E-06	3.31E-07	4.71E-07	1.09E-08
Erosion-Cavitation (E-C)	λ_f	6.24E-06	4.88E-06	8.57E-05	2.12E-04	4.17E-06	3.08E-05	1.95E-05	1.28E-06
	$P\langle R\|F\rangle$	4.44E-02	4.18E-02	4.20E-02	4.17E-02	3.53E-02	3.53E-02	3.53E-02	3.53E-02
	ρ_F	2.77E-07	2.04E-07	3.60E-06	8.85E-06	1.47E-07	1.08E-06	6.89E-07	4.49E-08
Design Construction Defects (DC)	λ_f	3.73E-06	4.46E-07	2.40E-07	3.84E-06	1.36E-07	1.78E-06	6.89E-07	8.16E-07
	$P\langle R\|F\rangle$	3.22E-02	3.23E-02	3.26E-02	3.20E-02	1.95E-01	1.95E-01	1.95E-01	1.95E-01
	ρ_F	1.20E-07	1.44E-08	7.82E-09	1.23E-07	2.60E-08	3.48E-07	1.34E-07	1.59E-07
Basis for Estimates		See Section 7.3 Of Tier 2 Documentation (Reference 4)				Reference 6			

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

Table 8
Impact of RISI and No Inspections on CDF and LERF Due to Pipe Ruptures for Byron Station Unit 1 Systems

System	System CDF Events/Reactor-Year			Δ CDF Events/Reactor-Year			Δ LERF Events/Reactor-Year		
	Section XI	RISI	No Inspection	RISI	No Inspection	Acceptance Criterion	RISI	No Inspection	Acceptance Criterion
RCS	2.79E-07	3.05E-07	3.36E-07	2.64E-08	5.73E-08	1.00E-07	5.29E-10	1.15E-09	1.00E-08
SI	1.29E-06	1.27E-06	1.34E-06	-1.63E-08	5.01E-08	1.00E-07	-1.63E-09	5.01E-09	1.00E-08
RHR	2.37E-07	2.33E-07	2.47E-07	-4.01E-09	9.79E-09	1.00E-07	-3.98E-10	9.82E-10	1.00E-08
CVCS	7.65E-08	7.19E-08	7.71E-08	-4.56E-09	6.40E-10	1.00E-07	-4.56E-10	6.40E-11	1.00E-08
MS	2.81E-09	2.91E-09	2.91E-09	9.89E-11	9.89E-11	1.00E-07	9.89E-12	9.89E-12	1.00E-08
CS	4.30E-08	4.21E-08	4.47E-08	-8.80E-10	1.76E-09	1.00E-07	-8.80E-11	1.76E-10	1.00E-08
SX	1.03E-08	1.03E-08	2.36E-08	0	1.33E-08	1.00E-07	0	1.33E-09	1.00E-08
FW	4.77E-08	4.93E-08	5.98E-08	1.63E-09	1.21E-08	1.00E-07	1.63E-10	1.21E-09	1.00E-08
Total	1.99E-06	1.99E-06	2.13E-06	2.46E-09	1.45E-07	1.00E-06	2.46E-10	1.45E-08	1.00E-07

Table 9
Impact of RISI and No Inspections on CDF and LERF Due to Pipe Ruptures for Byron Station Unit 2 Systems

System	System CDF Events/Reactor-Year			ΔCDF Events/Reactor-Year			ΔLERF Events/Reactor-Year		
	Section XI	RISI	No Inspection	RISI	No Inspection	Acceptance Criterion	RISI	No Inspection	Acceptance Criterion
RCS	2.36E-07	2.62E-07	2.92E-07	2.62E-08	5.60E-08	1.00E-07	5.24E-10	1.12E-09	1.00E-08
SI	1.20E-06	1.17E-06	1.25E-06	-2.38E-08	5.88E-08	1.00E-07	-2.38E-09	5.88E-09	1.00E-08
RHR	2.54E-07	2.50E-07	2.66E-07	-3.32E-09	1.19E-08	1.00E-07	-3.32E-10	1.19E-09	1.00E-08
CVCS	7.64E-08	7.62E-08	7.71E-08	-1.40E-10	7.53E-10	1.00E-07	-1.40E-11	7.53E-11	1.00E-08
MS	2.66E-09	2.76E-09	2.76E-09	9.89E-11	9.89E-11	1.00E-07	9.89E-12	9.89E-12	1.00E-08
CS	4.09E-08	4.01E-08	4.26E-08	-7.33E-10	1.76E-09	1.00E-07	-7.33E-11	1.76E-10	1.00E-08
SX	1.52E-08	1.52E-08	2.40E-08	0	8.73E-09	1.00E-07	0	8.73E-10	1.00E-08
FW	2.99E-07	2.69E-07	3.24E-07	-3.00E-08	2.54E-08	1.00E-07	-3.00E-09	2.54E-09	1.00E-08
Total	4.24E-06	4.17E-06	4.56E-06	-6.35E-08	3.27E-07	1.00E-06	-8.44E-09	2.82E-08	1.00E-07