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RS-01-219

10 CFR 50.55a(a)(3)(i)

October 15, 2001

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

Clinton Power Station, Unit 1
Facility Operating License No. NPF-62
Docket No. 50-461

Subject: Alternative to the ASME Boiler and Pressure Vessel Code Section XI Requirements for Class 1 and 2 Piping Welds Risk-Informed Inservice Inspection Program

References:
(1) Electric Power Research Institute (EPRI) Topical Report (TR) 112657 Revision B-A, "Revised Risk-Informed Inservice Inspection Evaluation Procedure," dated December 1999

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

In accordance with 10 CFR 50.55a, "Codes and standards," paragraph (a)(3)(i), AmerGen Energy Company, LLC (i.e., AmerGen) is submitting 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 by Clinton Power Station (CPS) uses Reference 1 methodology for a Risk-Informed Inservice Inspection (RISI) program approved by the U. S. Nuclear Regulatory Commission (NRC) to the extent and within the limitations specified in Reference 2.

The attached Relief Request 4208 and the RISI Program Plan Summary for CPS Unit 1 demonstrate that the proposed alternative would provide an acceptable level of quality and safety, as required by 10 CFR 50.55a (a)(3)(i). The format of the CPS RISI submittal is consistent with the Nuclear Energy Institute (NEI) and industry template developed for applications of the RISI methodology.

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The RISI program will be incorporated for the entire second Inservice Inspection interval for CPS, which began on January 1, 2000, and the projected end date is December 31, 2009. Implementation of this RISI program will reduce the number of ASME Section XI piping weld inspections by approximately 80 percent with little change in the risk to the public, while reducing occupational radiation exposure.

Approval of this proposed alternative is requested by March 15, 2002, to support the refueling outage scheduled for the spring of 2002.

Should you have any questions concerning this letter, please contact Mr. J. L. Peterson at (217) 937-2810.

Respectfully,

K. A. Ainger

K. A. Ainger
Director – Licensing
Midwest Regional Operating Group

Attachments
1) Relief Request 4208
2) Risk-Informed Inservice Inspection Program Plan Summary – Clinton Power Station, Unit 1

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Clinton Power Station

Attachment 1

Relief Request 4208

Clinton Power Station

ASME Section XI Relief Request

RELIEF REQUEST 4208 (Revision 0)

SYSTEM/ COMPONENT(S) FOR WHICH RELIEF IS REQUESTED	All American Society of Mechanical Engineers (ASME) Code Class 1 and 2 piping welds under Examination Category B-F, B-J, C-F-1, and C-F-2. Examination Item Numbers are B5.10, B5.130, B9.11, B9.12, B9.31, C5.11, C5.12, C5.51, and C5.52.

CODE REQUIREMENT	ASME Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 1989 Edition, Table IWB 2500-1, Examination Category B-F requires a volumetric and surface examination on all piping welds for Items B5.10 and B5.130.

Table IWB 2500-1, Examination Category B-J requires a volumetric and surface examination on all piping welds for Items B9.11, B9.12, and B9.31.

Table IWC 2500-1, Examination Categories C-F-1 and C-F-2 require volumetric and surface examinations for Items C5.11, C5.12, C5.51, and C5.52.

IWB-2430, "Additional Examinations," requires that any indications revealed that exceed the acceptance standards of Table IWB-3410-1 shall be extended to include additional examinations during the same outage. The additional examinations shall include the remaining welds, areas, or parts in the same inspection period and subsequent period. If the additional examinations revealed any indications exceeding the acceptance standards of Table IWB-3410-1, the examination shall be further extended to include additional examinations. The additional examinations shall include all remaining piping welds, areas, or parts of similar design, size and function.

IWC-2430, "Additional Examinations," requires that any indications revealed that exceed the allowable standards of IWC-3000 shall be extended to include an additional number of components (or areas) within the same category, approximately equal to the number of components (or areas) examined initially during the inspection. If the additional

examinations detect further indications exceeding the allowable standards of IWC-3000, the remaining number of similar components (or areas) within the examination category shall be examined.

BASIS FOR RELIEF

This relief is requested pursuant to 10CFR50.55a, "Codes and standards", paragraph (a)(3)(i). The proposed alternative of utilizing the examination methodology and selection criteria of Electric Power Research Institute (EPRI) TR-112657 Rev. B-A, "Revised Risk Informed Inservice Inspection Evaluation Procedure," along with evaluation and sample expansion requirement enhancements identified in ASME Code Case N-578-1, "Risk Informed Requirements for Class 1, 2, and 3 Piping, Method B," will provide an acceptable level of quality and safety.

In a letter from W.H. Bateman (U.S. NRC) to G.L. Vine (EPRI), dated October 28, 1999, "Safety Evaluation Report Related to EPRI Risk Informed Inservice Inspection Evaluation Procedure," the NRC states the following.

> "The staff concludes that the proposed RI-ISI 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 10CFR50.55a for the proposed alternative to the piping ISI requirements with regard to the number of locations, locations of inspections, and methods of inspection."

In lieu of the evaluation and sample expansion requirements of EPRI TR-112657, Revision B, Section 3.6.6.2, "RI-ISI Selected Examinations," Clinton Power Station (CPS) will utilize the requirements of Subarticle-2430, "Additional Examinations," which is contained in Code Case N-578-1. The alternative criteria for additional examinations contained in Code Case N-578-1 provides more guidance for examination method and categorization for parts to be examined.

ALTERNATE EXAMINATIONS

CPS proposes to utilize the proposed alternative described in Attachment 2 to this submittal, "Risk Informed Inservice Inspection Program Plan Summary, Clinton Power Station, Unit 1".

JUSTIFICATION FOR THE GRANTING OF RELIEF

The proposed alternative described in Attachment 2 to this submittal, "Risk Informed Inservice Inspection Program Plan Summary, Clinton Power Station, Unit 1,"

provide an acceptable level of quality and safety as required by 10CFR50.55a(a)(3)(i).

The CPS risk-informed inservice inspection (RISI) program requires that 25% of the elements that are categorized as "High Risk" (Risk Categories 1, 2, or 3) and 10% of the elements that are categorized as "Medium Risk" (Risk Categories 4 and 5) be selected for volumetric examination. For this application, the guidance for the examination volume for a given degradation mechanism is provided by EPRI TR-112657 and supplemented by Code Case N-578-1 for examination method and categorization for parts to be examined.

In addition, all Section XI 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 CPS pressure and leak test program, which remains unaltered by the RISI program.

IMPLEMENTATION SCHEDULE

CPS plans to incorporate the RISI program for the entire second 10-year interval. The second 10-year interval began on January 1, 2000 and the projected end date is December 31, 2009. The first inspection period began on January 1, 2000, and the projected end date is December 31, 2002.

Attachment 2

Risk Informed Inservice Inspection Program Plan Summary

Clinton Power Station
Unit 1

RISK INFORMED INSERVICE INSPECTION PROGRAM PLAN SUMMARY

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RISK INFORMED INSERVICE INSPECTION PROGRAM PLAN SUMMARY

1. INTRODUCTION

The objective of this submittal is to request approval of 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 ASME Section XI based ISI program. The RISI program is proposed 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 EPRI RISI Topical Report (Reference 1) and the accompanying NRC staff Safety Evaluation Report (SER) on the EPRI method. 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.

AmerGen plans to incorporate the RISI inspection program during the first Inspection Period of the second Inspection Interval for Clinton Power Station (CPS). The Second Inservice Inspection Interval started on January 1, 2000 for CPS, and the projected end date is December 31, 2009. CPS has completed the first outage of the First Inspection Period at this time. The RISI program will start with the second outage of the First Inspection Period. CPS will take credit for those welds examined in the first outage of the first period if these welds are also selected for the RISI program. If necessary, additional welds from the RISI program will be examined in order to meet the minimum code requirements for the First Inspection Period. The examination distribution will be consistent with ASME Section XI requirements (e.g., the minimum examinations completed at the end of the three Inspection Periods under Program B should be 16 percent, 50 percent, and 100 percent, respectively, and the maximum examinations credited at the end of the respective Inspection Periods should be 34 percent, 67 percent, and 100 percent). This method of RISI incorporation would result in the completion of 100% of the RISI components selected for examination within a ten-year time frame as would occur if the RISI program was started at the beginning of the Inspection Interval. The current Period and Interval dates will not be altered by this method.

As a risk-informed application, this submittal meets the principles of Regulatory Guides 1.174, "An Approach For Using Probabilistic Risk-Informed Decisions On Plant-Specific changes to the Licensing Basis", and 1.178, "An Approach For Plant-Specific Risk-Informed Decisionmaking Inservice Inspection of Piping", as well as those set forth in the EPRI RISI Topical Report and the NRC staff SER on the EPRI RISI method. Several recurring questions have been seen in the RAIs for the other Exelon plant RISI submittals. The information required to answer those questions is contained in Reference 4 documentation. A summary of the information necessary to address the issues raised in these questions for CPS is included as Attachment A.

PRA Quality

The CPS PRA used for the risk determinations for this regulatory application is a recent upgrade to the "Clinton Power Station Individual Plant Examination" (IPE), submitted to the NRC dated September 1992 (Reference 7). The total core damage frequency (CDF) calculated by the CPS PRA model is 1.4E-05/yr. and the total large early release frequency (LERF) is 1.4E-07/yr.

The original CPS PRA was performed by Illinois Power to support the IPE submittal. The current CPS PRA is a third generation upgrade to that study. The CPS PRA addresses internal events at full power and it includes internal flooding. Internal fire risk characterizations are taken from the CPS Individual Plant Examination For External Events (IPEEE), but its results are considered to be conservative in many of their assumptions. Therefore, fire risk is not directly comparable to other quantified internal events risk results.

The LERF was estimated using the containment analysis (level 2) PRA model. Cutsets from the core damage (level 1) model are binned by accident class and are used as the input to the full level 2 PRA analysis. Those containment event tree sequences that correspond to large radiological releases in an early time frame relative to the time it would take for protective actions for the public are designated as LERF sequences. LERF results calculated in this way properly take into account the effects of the equipment failures leading to core damage. The LERF model is used to calculate the conditional large early release probabilities (CLERPs) that are used in the consequence analysis.

Both the CPS PRA model and its supporting bases documentation were reviewed by a BWROG Peer Review/Certification Team during August 2000. The review was conducted using NEI 00-02, "NEI PSA Certification Peer Review Process," using a team of industry PRA experts. This independent review was performed to evaluate the quality of the PRA and completeness of the PRA documentation. Based on the results of past NRC Staff reviews and the BWROG Certification Peer Review, AmerGen is confident that the level of detail and quality of the CPS PRA fully supports this risk-informed regulatory application.

AmerGen maintains and updates each of its PRAs to be representative of the respective as-built, as-operated plant. Project Instructions formalize the PRA update process. The Instructions define the process for updating the PRA to ensure that it adequately represents the CPS as-built, as-operated plant. This process assures the present PRA reflects the current plant configuration and plant procedures.

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 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 EPRI RISI Topical Report (Reference 1). The RISI program will be substituted for the 1989 ASME Section XI Code Edition examination program for Class 1 Category B-J and B-F 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 imposed inservice inspection program outside of this RISI scope will be unaffected. Reference 1 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 Reference 1, certain augmented inspection programs may be integrated into the RISI program. At this time, no augmented programs are subsumed in the RISI program, with the exception of the IGSCC Category A welds. The following augmented programs were not subsumed into the RISI program and remain unaffected.

- IGSCC in BWR Austenitic Stainless Steel Piping (Generic Letter 88-01 and NUREG-0313) other than IGSCC Category A welds which have been subsumed into the RISI program.
- Service Water Integrity Program (Generic Letter 89-13)
- Flow Accelerated Corrosion (FAC) (Generic Letter 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 the damage mechanisms at those elements and 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. If another damage mechanism was identified, the element was retained within the scope of consideration for element selection as part of the RISI 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. The entire scope of the RISI evaluation including those elements covered by augmented programs and not included in the RISI selection population were included in the risk impact assessment phase of the evaluation described below.

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 at the Reactor Pressure Vessel Nozzles (AAI) that have the potential for both IGSCC 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 Reference 1. 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 Reference 1 for ASME Code Case N-578-1 (Reference 6) 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 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 (i.e., High, Medium or Low) were assigned according to Table 3-1 of Reference 1. 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 Reference 1. 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 at AmerGen that directly utilize the plant-specific PRA models.

3.3 Degradation Mechanism Assessment

Failure potential was assessed using the deterministic criteria in Reference 1 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 Reference 1.

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 5 and described in the Reference 4 documentation.

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.

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 for Clinton.

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 Elements (Reference 1)

3.5 Inspection Location Selection and NDE Selection

In general, an ASME Code Case N-578-1 application of RISI, per Reference 1, requires that 25% of the elements that are categorized as "High" risk (i.e., Risk Category 1, 2, or 3) and 10% of the elements that are categorized as "Medium" risk (i.e., 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 Table 4. Section 4 of Reference 1 and ASME Code Case N-578-1 (Reference 6) were used as guidance in determining the examination methods and requirements for these locations. From the Class 1 butt welded elements that were considered within the scope of the RISI evaluation, a total of 12.0% were selected for volumetric examination as part of the risk informed inspection program. The total Class 1 welds selected for RISI evaluation was 12.0% because there are no Class 1 socket welds. As noted above, elements found to be susceptible to two or more damage mechanisms were given enhanced treatment by retaining them within the scope of the augmented programs and in the risk informed program for the applicable damage mechanisms.

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.

Additional Examinations

Examinations performed that reveal flaws or relevant conditions exceeding the applicable 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 (i.e., Risk Group Categories 1 through 5) within systems, whose material 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.

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.

If there are not enough high safety significant elements (i.e., in the same and higher "Risk Group Categories") with the same postulated failure mode, lower safety significant elements (i.e., in lower "Risk Group Categories" other than Risk Group Categories 6 and 7) with the same postulated failure mode will be selected such that the number of additional elements is at least equal to the number of elements with the same postulated failure mode originally scheduled for that fuel cycle.

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

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 Reference 1 will be followed.

3.7 Risk Impact Assessment

The RISI program has been developed in accordance with Regulatory Guides 1.174 and 1.178, and Reference 1, which require an evaluation to show that implementation of a risk informed inspection program would result in acceptably small changes, if any, in CDF and LERF.

The risk impact assessment performed in this RISI application included a 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 and element 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 methods described in Reference 1.

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. In the risk impact assessment, each element was quantified in terms of changes in failure frequency, rupture frequency, CDF, and LERF due to proposed changes in the risk informed inspection program. Then, the elements results were grouped by system to determine the change in risk (CDF and LERF) and overall total risk change.

Per Section 3.7.2 of Reference 1, the Markov piping reliability analysis method 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 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 Reference 1.

The changes to the ASME Section XI ISI program include changing the number and location of inspections within the system, 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 favorably impact the POD.

Limits are imposed by Reference 1 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 $1x10^{-7}$ and $1x10^{-8}$ per year per system, respectively. Meeting these limits is consistent with meeting Regulatory Guide 1.174 risk significant thresholds of $1x10^{-6}$ per year and $1x10^{-7}$ per year for changes in CDF and LERF, respectively, 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 Reference 4. These parameters include a set of failure rates and rupture frequencies for piping systems in General Electric BWR 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 those developed in Table A-11 in EPRI TR-111880 (Reference 5).

Separate Markov calculations were performed for the change in CDF and the change in LERF. This calculation was performed so that pipe elements whose failure could create a potential containment failure or bypass concern were factored into the LERF evaluation. 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 CPS are summarized in Table 5 and key aspects are plotted in Figures 2 and 3 for comparison against the risk significant criteria established in Reference 1. As seen in these figures and table, the Reactor Pressure Vessel (AAP), Feedwater (PFW), and Low Pressure Core Spray (PLP) system groups exhibited small decreases in CDF due to the changes from the RISI program. The AAP system group exhibited a small decrease in LERF. The remaining systems evaluated exhibited very small increases in CDF and LERF. In each case in which a risk increase was identified, the estimated increases in CDF and LERF are much smaller than the risk acceptance criteria. Each system was found to have a change in LERF that is less than or equal to 2% of the EPRI RISI risk significance threshold of $1x10^{-8}$/system-year, and a change in CDF that is less than 3% of the associated threshold of $1x10^{-7}$/system-year.

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 very small in relation to Regulatory Guide 1.174 risk significance criteria. The margin for these risk metrics is more than two orders of magnitude.

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 Reference 1.

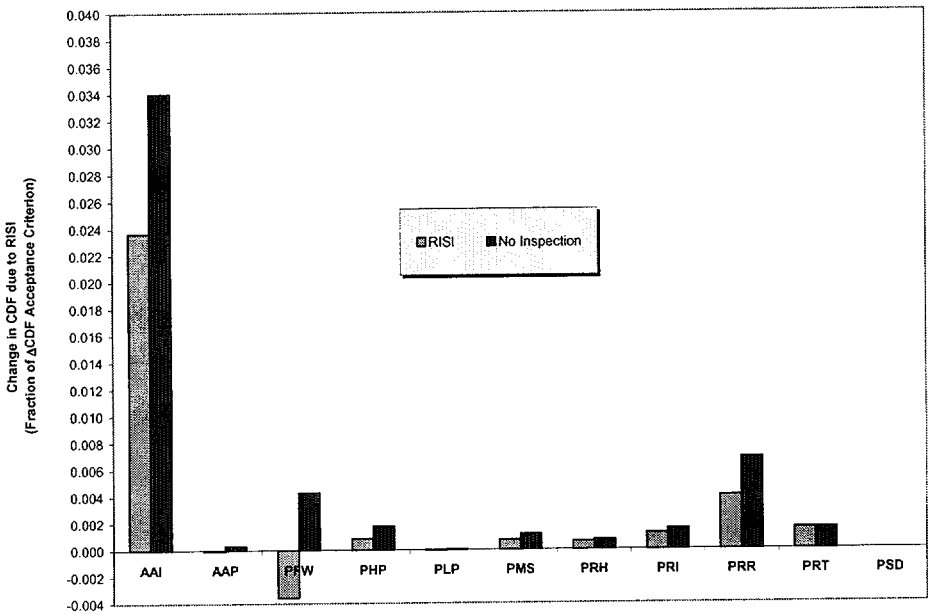

Figure 2
Change in Pipe Rupture CDF for Clinton Systems
as a Fraction of EPRI Risk Significance Criterion

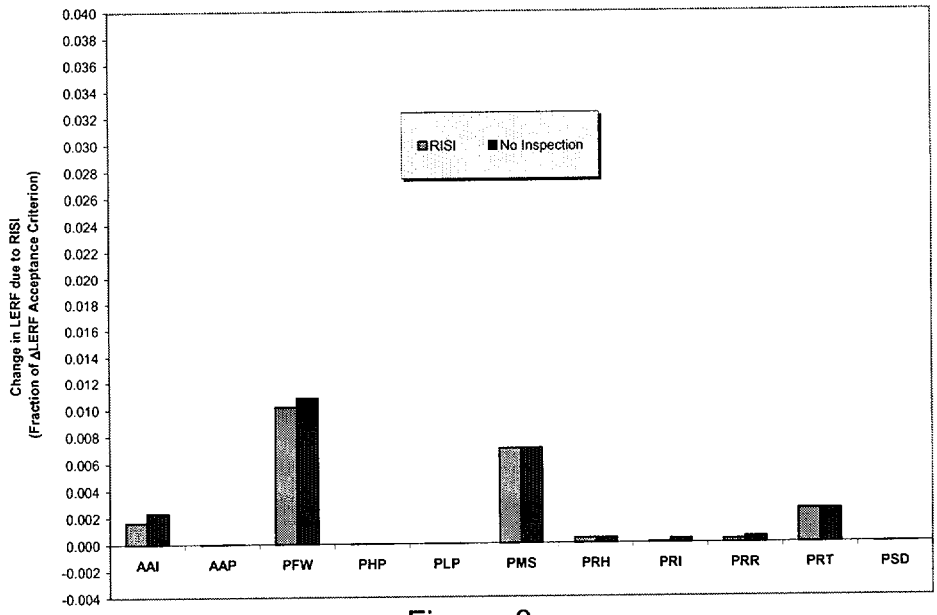

Figure 3
Change in Pipe Rupture LERF for Clinton Systems
as a Fraction of EPRI Risk Significance Criterion

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 Reference 1 will be prepared to implement and monitor the program. The new program will be integrated into the first period of the second inservice inspection interval for CPS. No changes to the Updated Safety Analysis Report (USAR) are necessary for program implementation.

The applicable aspects of the ASME Code not affected by this change are to be retained, such as acceptance criteria, 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 CPS PRA models which could impact both the risk characterization and risk impact assessments, any new trends in service experience with piping systems at CPS 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 service experience 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 4. The number of exams is reduced from 322 Section XI program exams to 74 RISI program exams, a net reduction of 248 exams. An additional 45 Section XI exams were also eliminated from the FAC and IGSCC augmented program welds for a total reduction of 293 exams compared to the 368 Section XI total (80% reduction). As shown in Table 5, the total increase in CDF and LERF due to the net changes in number and location of inspections in all systems that were evaluated in this risk informed evaluation was found to be less than $3x10^{-9}$ per year for both risk measures. These risk impacts are acceptably small in relation to the risk significance thresholds of Reference 1 and those in Regulatory Guide 1.174. Examinations performed for augmented inspections programs not subsumed in the RISI program are not impacted and remain in place.

6. REFERENCES

1. EPRI, "Revised Risk-Informed Inservice Inspection Evaluation Procedure," TR-112657, Rev. B-A, December 1999 (includes NRC staff Safety Evaluation Report on Procedure).
2. Clinton Nuclear Station Updated PRA Model (CDF), Rev. 3a.
3. Clinton Nuclear Station Updated PRA Model (LERF), Rev. 3a.
4. Risk Informed Inservice Inspection Evaluation, Clinton Nuclear Power Plant – Final Report, September 2001.
5. 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, 1999, September 1999. EPRI Licensed Material.

6. ASME Code Case N-578-1, "Risk-Informed Requirements for Class 1, 2, and 3 Piping, Method B, Section XI, Division 1."

7. Illinois Power, "Clinton Power Station Individual Plant Examination Final Report," September 1992.

Table 1

RISI System List

System Description
Main Steam (PMS)
Reactor Core Isolation Cooling (PRI)
Feedwater (PFW)
Reactor Recirculation System (PRR)
Low Pressure Core Spray (PLP)
High Pressure Core Spray (PHP)
Nuclear Boiler (PNB)
Reactor Water Cleanup System (PRT)
Residual Heat Removal (PRH)
Scram Discharge Volume (PSD)
Reactor Pressure Vessel (AAI, AAP)

NOTE:

This table shows the systems containing Class 1 or Class 2 category B-J, B-F, C-F-1, or C-F-2 welds.

Table 2

Failure Potential Assessment Summary for CPS

System	Thermal Fatigue		Stress Corrosion Cracking				Localized Corrosion			Flow Sensitive	
	TASCS	TT	IGSCC	TGSCC	ECSCC	PWSCC	MIC	PIT	CC	E-C	FAC
AAI[1]	X	X	X								X
AAP	X	X							X		
PFW	X	X									X
PHP	X										
PLP	X										
PMS											
PRH	X	X	X							X	X
PRI	X	X									
PRR			X								
PRT			X								X
PSD											

1. Includes nuclear boiler (NB).

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 mechanisms that were identified in each system.

Table 3

Number of Elements (Welds) by Risk Category for CPS[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
AAI[1]	0	28	9	8	0	0	45
AAP	0	0	0	0	1	3	4
PFW	53	0	14	0	0	0	67
PHP	0	3	0	33	0	89	125
PLP	0	0	0	1	3	92	96
PMS	0	0	0	78	0	4	82
PRH	0	4	68	4	28	593	697
PRI	0	1	0	7	56	146	210
PRR	0	2	0	131	0	0	133
PRT	51	8	3	0	0	0	62
PSD	0	0	0	0	0	44	44
TOTAL	104	46	94	262	88	971	1565

1. Includes nuclear boiler (PNB).
2. Includes 126 High Risk Category welds in augmented programs (FAC only and IGSCC only). These 126 welds are not included in the selection percentages.

Table 4

Number of Inspections by Risk Category for CPS

	High Risk						Medium Risk				Low Risk		All Risk Categories	
	Category 1		Category 2		Category 3		Category 4		Category 5		Category 6 or 7			
System	Sec. XI[2]	RISI	Sec. XI[2]	RISI	Sec. XI[2]	RISI	Sec. XI[2]	RISI	Sec. XI[2]	RISI	Sec. XI[2]	RISI	Sec. XI[2]	RISI
AAI[1]			28	4	3		8	1					39	5
AAP									1	1			1	1
PFW	9	14			6	4							16	18
PHP				1			18	4			4		22	5
PLP							1	1		1	19		20	2
PMS							13	8			4		17	8
PRH			4		4	9	2	1	14	3	91		115	13
PRI			1	1			1	1	56	6	11		69	8
PRR			2				42	14					44	14
PRT	10		8		3								21	
PSD											4		4	
TOTAL	19	14	43	6	16	13	85	30	71	11	133		368	74

1. Includes nuclear boiler (PNB).
2. 1989 ASME Section Code Edition.

Table 5

Impact of RISI and No Inspections on CDF and LERF Due to Pipe Ruptures for CPS 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
AAI[1]	2.70E-09	5.06E-09	6.10E-09	2.36E-09	3.40E-09	≤1.00E-07	1.63E-11	2.35E-11	≤1.00E-08
AAP	7.12E-11	6.26E-11	1.01E-10	-8.59E-12	2.95E-11	≤1.00E-07	-1.20E-13	4.12E-13	≤1.00E-08
PFW	3.69E-09	3.34E-09	4.12E-09	-3.55E-10	4.31E-10	≤1.00E-07	1.02E-10	1.08E-10	≤1.00E-08
PHP	8.66E-10	9.51E-10	1.04E-09	8.45E-11	1.78E-10	≤1.00E-07	8.08E-14	1.09E-13	≤1.00E-08
PLP	8.21E-11	7.62E-11	8.75E-11	-5.88E-12	5.39E-12	≤1.00E-07	8.08E-14	9.48E-14	≤1.00E-08
PMS	6.76E-10	7.49E-10	7.93E-10	7.30E-11	1.16E-10	≤1.00E-07	7.04E-11	7.07E-11	≤1.00E-08
PRH	4.35E-10	4.97E-10	5.08E-10	6.24E-11	7.32E-11	≤1.00E-07	3.84E-12	3.96E-12	≤1.00E-08
PRI	3.24E-10	4.48E-10	4.77E-10	1.24E-10	1.53E-10	≤1.00E-07	8.01E-13	3.27E-12	≤1.00E-08
PRR	3.73E-09	4.13E-09	4.41E-09	4.03E-10	6.85E-10	≤1.00E-07	2.79E-12	4.74E-12	≤1.00E-08
PRT	6.89E-10	8.49E-10	8.49E-10	1.60E-10	1.60E-10	≤1.00E-07	2.54E-11	2.54E-11	≤1.00E-08
PSD	1.00E-14	1.00E-14	1.00E-14	1.00E-14	1.00E-14	≤1.00E-07	1.94E-13	1.94E-13	≤1.00E-08
Total	1.33E-08	1.62E-08	1.85E-08	2.90E-09	5.23E-09	≤1.00E-06	2.22E-10	2.41E-10	≤1.00E-07

1. Includes nuclear boiler (PNB).

ATTACHMENT A

Issues Common To Other Exelon Plants

A.0 INTRODUCTION

Some common questions have been included in the RAIs for the RISI submittals from other Exelon Midwest ROG plants (Dresden, Quad Cities, Byron and Braidwood) that use the same methodology that was used for the Clinton Power Station (CPS). The information necessary to answer these questions is found in the CPS Tier 2 documentation for the RISI program. Portions of that information have been extracted and are presented in this attachment to proactively address some of these issues. Each of the following sections provides this information for one of these issues.

A.1 SYNERGY BETWEEN MULTIPLE DAMAGE MECHANISMS

The following excerpt from the Reference A7-4 documentation describes how failure rates and rupture frequencies were impacted by synergy for the conservative assumptions in the delta risk evaluation.

- *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 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 3 to consider the possible effects of synergy, i.e., to consider the potential that through wall cracks would occur more quickly when two or more mechanisms were present at the same location. Design and construction errors are not considered a separate damage mechanism for the purpose of determining whether or not the synergy factor will be applied.*

The above treatment was made because the service data upon which the 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 (TT, TASCS or both) and corrosion cracking and the corrosion cracking is not covered in the augmented program for IGSCC (hypothetical case), the failure rates for design and construction errors, thermal fatigue, and stress corrosion cracking from EPRI TR-111880 [Reference A7-3] would be summed and then this result would be multiplied by a factor of 3 for synergy. The rupture frequencies would be determined in the same way. But if the segment was found susceptible to the same 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 increased by the factor of 3 applied to the sum of the failure rate or rupture frequency for the design and construction errors and thermal fatigue damage mechanism.

The potential for synergy was considered using engineering judgement in the delta risk evaluation as explained above, the assignment of failure potential categories in the application of the EPRI RISI [Reference A7-1] risk matrix was not changed as a result of this consideration of synergy. Hence, 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. Our judgement was that a factor of 3 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 requirements set forth in RG 1.174, RG 1.178, and the EPRI RISI Topical Report [Reference A7-1].

A.2 EQUATIONS AND INPUT USED FOR CHANGE IN RISK CALCULATIONS

The Table below provides references for the equations used in the delta risk calculation.

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-11
Plant specific documentation of all other input data needed to quantify above equations	Clinton Unit 1 RISI Evaluation (Tier 2 Documentation)	See description of input parameters following this table

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.

$$\omega = \frac{P_{FI} P_{FD}}{(T_{FI} + T_R)} \tag{A.1}$$

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 element 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:

$$\mu = \frac{P_{LD}}{(T_{LI} + T_R)} \tag{A.2}$$

Where:

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

T_{LI} = mean time between inspections for leaks. For pipes containing radioactive fluid such as the RCS, the time interval between leaks can be essentially instantaneous if the leak is picked up by radiation alarms, to as long as the time period between leak tests performed on the system. All ASME Class 1, 2, and 3 piping must be tested for leaks at least once per refueling outage.

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 inspection effectiveness factors can be quantified once the root input parameters of the Markov model have been quantified. A summary of the root input parameters of the Markov model and the general strategy for estimation of each one is presented in Table A-1. The specific basis for estimation of each of these parameters for CPS Class 1 and 2 systems is provided in Table A-1.

Table A-1

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. See Equation A.3 and accompanying text.
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. See Table A-2 for values used.
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. See Table A-2 for values used.
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. Set to 1.97E-02 based on data for water hammer events [Reference A7-5].
ω	Inspection and repair rate of a flaw state	Model of Equation (A.1) and estimates of P_{FI}, P_{FD}, T_{FI}, T_R as estimated below.
μ	Detection and repair of a leak state	Model of Equation (A.2) and estimates of P_{LD}, T_{LI}, T_R as estimated below.
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. See Table A-3 for values used.
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. P_{LD} is assumed to be 0.90.
T_{FI}	Flaw inspection interval, mean time between in service inspections	Set to 10 years for ASME Section XI or RISI piping systems.
T_{LI}	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, set to refueling outage interval (1.5 years).
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; set to a value of 200 hours.

Table A-2

Mean Failure Rates, Conditional Rupture Probabilities, and Rupture Frequencies

Used in CPS Risk Impact Assessment

Damage Mechanism	Parameter*	EPRI TR-111880 System Group** [Reference A7-3]							
		RCS	SIR	CS	RAS	AUXC	FWC	ST	FPS
Thermal Fatigue (TF)	λ_f	9.01E-05	7.96E-07	2.10E-06	4.33E-05	1.26E-06	4.82E-06	8.55E-06	1.29E-06
	$P\langle R\|F\rangle$	5.56E-02	5.56E-02	3.53E-02	3.53E-02	3.53E-02	3.53E-02	3.53E-02	3.53E-02
	ρ_F	5.11E-06	4.41E-08	7.32E-08	1.53E-06	4.25E-08	1.69E-07	3.01E-07	4.75E-08
Stress Corrosion Cracking (SC)	λ_f	4.36E-04	4.10E-04	1.33E-04	6.12E-04	4.53E-05	2.32E-04	8.13E-05	2.36E-06
	$P\langle R\|F\rangle$	1.89E-02	1.89E-02	1.15E-02	1.15E-02	1.15E-02	1.15E-02	1.15E-02	1.15E-02
	ρ_F	8.23E-06	7.76E-06	1.52E-06	7.06E-06	5.25E-07	2.63E-06	9.32E-07	2.78E-08
Erosion-Cavitation (E-C)	λ_f	3.98E-04	1.37E-05	6.06E-06	2.38E-06	4.81E-05	2.84E-05	2.76E-06	2.46E-05
	$P\langle R\|F\rangle$	5.56E-02	5.56E-02	3.53E-02	3.53E-02	3.53E-02	3.53E-02	3.53E-02	3.53E-02
	ρ_F	2.23E-05	7.62E-07	2.08E-07	8.46E-08	1.70E-06	1.00E-06	9.30E-08	8.77E-07
Design Construction Defects (DC)	λ_f	4.69E-05	1.77E-06	1.88E-07	4.42E-06	3.91E-06	7.08E-06	3.04E-06	1.08E-07
	$P\langle R\|F\rangle$	4.76E-02	4.76E-02	1.95E-01	1.95E-01	1.95E-01	1.95E-01	1.95E-01	1.95E-01
	ρ_F	2.23E-06	8.44E-08	3.76E-08	8.60E-07	7.64E-07	1.38E-06	5.95E-07	2.07E-08

* 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

** Definition of System Groups:

RCS	Reactor Coolant System	Used for CPS PRR, PNB, AAI, and AAP System Groups
SIR	Safety Injection and Recirculation	Used for CPS PHP, PRH, PLP, and PRI System Groups
RAS	Reactor Auxiliary System	Used for CPS PRT and PSD System Groups
AUXC	Auxiliary Cooling Systems	Not Used for CPS
FWC	Feedwater and Condensate	Used for CPS PFW System
ST	Steam Systems	Used for CPS PMS System Group
FPS	Fire Protection Systems	Not Used for CPS
CS	Core Spray System	Used for CPS PCS System

Frequency of Flaws (ϕ)

The frequency of flaws, ϕ is calculated from the pipe failure frequency, λ and the ratio of cracks to leaks, $R_{C/F}$ using the following expression:

$$R_{C/F} = \frac{\phi}{\lambda} \qquad \text{(A.3)}$$

For CPS, the crack-to-leak ratio that was used is 9.19 for IGSCC piping that is part of the IGSCC augmented program, and 4.28 for other damage mechanisms in all piping within the scope of the RISI program.

Table A-3

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	EPRI RISI procedure calls for expanded inspection zone for elements susceptible to TF, assumption used in NRC reviewed Markov application [Reference A7-2] and [Reference A7-4]
EPRI RISI of element in Stainless steel pipe subject to thermal fatigue	P_{FD} =.80	Carbon steel value reduced slightly to reflect insights from EPRI NDE qualification program [Reference A7-6]
EPRI RISI of element subject to other damage mechanism subject to inservice inspection	P_{FD} = .75	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	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	Since there is no inspection for cause principle to apply, high confidence in detection cannot be assured

A.3 INSPECTION EFFECTIVENESS FACTOR

The inspection effectiveness factor is the ratio of the inspected weld rupture frequency to the non-inspected rupture frequency. Section 3.7.2 of Reference A7-1 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 of the ISI exam, or POD. 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: the POD which may be different whether the exam is done per ASME Section XI or per EPRI RISI examination criteria, the assumed failure rates and rupture frequencies which are taken to be dependent and conditional on the system, pipe size, and 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 RISI evaluation for CPS Unit 1 is presented in Table A-4. 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.

The inspection effectiveness factors developed using the Markov model are viewed as a more realistic assessment of inspection effectiveness for several reasons, including:

- 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 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, the time intervals between inspections etc.

Having stated this, 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 A-4.

Table A-4

Probability of Detection (POD) and Inspection Effectiveness Factors Used for CPS 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)
PSD	D&C[1]	0.500	0.435	0.500	0.500	0.435	0.500
PRH	D&C[1]	0.500	0.438	0.500	0.500	0.438	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
	EC	0.750	0.322	0.250	0.500	0.438	0.500
	TASCS, FAC	0.800	0.305	0.200	0.500	0.438	0.500
	EC, FAC	0.750	0.322	0.250	0.500	0.438	0.500
	IGSCC	0.750	0.322	0.250	0.500	0.438	0.500
	FAC	0.500	0.435	0.500	0.500	0.435	0.500
PFW	TASCS, FAC	0.900	0.273	0.100	0.500	0.436	0.500
	TASCS, TT, FAC	0.900	0.273	0.100	0.500	0.436	0.500
PHP	D&C[1]	0.500	0.438	0.500	0.500	0.438	0.500
	TASCS	0.800	0.305	0.200	0.500	0.438	0.500
PLP	D&C[1]	0.500	0.438	0.500	0.500	0.438	0.500
	TASCS	0.800	0.305	0.200	0.500	0.438	0.500
PMS	D&C[1]	0.500	0.435	0.500	0.500	0.435	0.500
RPV[2]	D&C[1]	0.500	0.439	0.500	0.500	0.439	0.500
	IGSCC	0.750	0.322	0.250	0.500	0.439	0.500
	FAC	0.500	0.435	0.500	0.500	0.435	0.500
	TASCS, IGSCC	0.800	0.306	0.200	0.500	0.439	0.500
	TASCS, TT, IGSCC	0.800	0.306	0.200	0.500	0.439	0.500
	TASCS, TT, CC	0.800	0.306	0.200	0.500	0.439	0.500
PRT	D&C[1]	0.500	0.435	0.500	0.500	0.435	0.500
	IGSCC	0.750	0.319	0.250	0.500	0.435	0.500
	FAC	0.500	0.435	0.500	0.500	0.435	0.500
PRI	D&C[1]	0.500	0.435	0.500	0.500	0.435	0.500
	TT	0.800	0.305	0.200	0.500	0.438	0.500
	TASCS, TT	0.800	0.305	0.200	0.500	0.438	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.
2. RPV includes AAI, AAP, PNB and PRR systems.

According to the ASME Code Section XI, all Class 1 piping systems must be inspected for leaks by performing a system leak test and observing for leaks at least once per

refueling cycle. For Class 2 piping, the requirement is to perform these leak tests once per ISI inspection period. In between these leak tests there are other opportunities to identify leaks via routine plant walkdowns and other test and maintenance activities on the piping systems that occur much more frequently than the ASME Section XI imposed leak tests. The following default values used for all segments in this evaluation for the probability of detecting a leak (P_{LD}) and the time interval between opportunities for detecting leaks (T_{LD}) are:

$$P_{LD} = .90$$

$$T_{LD} = 1.5 \text{ years}$$

The same values are used for both Class 1 and Class 2 segments and were not varied between the Section XI and RISI evaluation cases. Since the Markov model results are not sensitive to variations in this parameter and because the parameter does not differentiate between ASME Section XI and RISI programs, it was not necessary to develop segment dependent inputs for this parameter.

A.4 BOUNDING CALCULATION OF DELTA RISK

A simplified and conservative risk impact calculation, not using the Markov model calculation of pipe break frequency, was performed for the other Exelon Midwest ROG plants (Byron, Braidwood, Dresden, Quad Cities, and LaSalle) as a sensitivity study and was also performed for CPS. This calculation was performed using the same approach as was implemented for the previously-approved relief request for South Texas Project, which was performed by ERIN. The change in risk for a particular system was calculated using the following:

$$\Delta CDF_j = \sum_i [FR_{i,j} * (SXI_{i,j} - RISI_{i,j}) * CCDP_{i,j}] \qquad \text{(A.4)}$$

where

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

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

$$\Delta CDF_{TOTAL} = \sum_j \Delta CDF_j \qquad \text{(A.5)}$$

Similar calculations were performed using the CLERP to determine the change in LERF for each system and the total change in LERF due to implementing the RISI program. Results of these calculations are presented in Table A-5 for CPS. Also shown in Table A-5 are the results of the Markov model calculation of the change in risk, for comparison purposes.

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.

These conservative results are regarded as a sensitivity study as they only reflect upper bounds on the expected risk impacts. Note that the delta risk results for both the Markov model calculation and the bounding calculation are far below the risk acceptance criteria of EPRI TR-112657 [Reference A7-1]. The results obtained using the Markov model are considered more reasonable and realistic for the following reasons.

- There were many cases in which the effectiveness of the inspection will be increased as a result of the application of the "inspection for cause" principle in which the knowledge of the applicable damage mechanisms and the application of mechanism specific inspection methods provide a reasonable basis to expect enhanced inspection effectiveness. A good example is the case of locations susceptible to thermal fatigue in which the EPRI RISI exams call for an expanded examination volume into the Heat Affected Zone (HAZ) of the weld in comparison with ASME Section XI examination requirements. This expanded volume recommendation is based on insights from service experience that indicate the location of cracks in the areas of welds caused by thermal fatigue. These inspection for cause effects are ignored in the bounding evaluations.

  The conservative calculation assumes that all the change in risk in a given risk segment comes from the net change in the number of exams; which implies that there can be no change from redistributing a fixed number of welds. This does not reflect the true philosophy of risk management as expressed in Regulatory Guides 1.178 and 1.174, or the EPRI Topical Report regarding the balancing of resources away from areas with marginal risk impact toward areas of more significant risk impact.

Table A-5

Comparison of Risk Impact Results for CPS

CPS Risk Impact Report *				
System	CDF		LERF	
	Bounding Delta CDF	Realistic Delta CDF using Markov Model	Bounding Delta LERF	Realistic Delta LERF using Markov Model
AAI	1.05E-08	2.36E-09	7.24E-11	1.63E-11
AAP	0.00E+00	-8.59E-12	0.00E+00	-1.20E-13
PFW	-3.18E-10	-3.55E-10	1.83E-10	1.02E-10
PHP	1.73E-10	8.45E-11	1.47E-13	8.08E-14
PLP	-7.84E-12	-5.88E-12	1.47E-13	8.08E-14
PMS	4.57E-10	7.30E-11	4.53E-10	7.04E-11
PRH	6.12E-10	6.24E-11	1.03E-11	3.84E-12
PRI	2.31E-10	1.24E-10	1.47E-12	8.01E-13
PRR	9.82E-10	4.03E-10	6.79E-12	2.79E-12
PRT	1.19E-09	1.60E-10	5.12E-11	2.54E-11
PSD	0.00E+00	0.00E+00	3.44E-13	1.94E-13
Total	1.38E-08	2.90E-09	7.78E-10	2.22E-10

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

- The risk impact of changing the inspection strategy of a given weld is one of the factors that was considered in the element selection. If that input to the selection is skewed by conservative assumptions that do not uniformly impact across the elements in the program, the goal of an optimized program is not as well supported in comparison with the case where realistic assumptions are used for all the welds in the examination.
- The inspection effectiveness factors obtained using the Markov model provide a more realistic perspective on the benefits of ISI exams. This permits better treatment in balancing the combined influences of removing exams, redistributing exam locations, and enhancing the effectiveness of exams through the inspection for cause principle.
- This approach of performing a realistic risk impact assessment provides a better basis to normalize risks and risk impacts across different risk informed initiatives

such as Risk-Informed Inservice Inspection (RISI), Risk-Informed Inservice Testing (RIST), and risk informed technical specifications, in contrast to limiting the analysis for RISI to a conservative bounding assessment. If one of these applications uses conservative bounding estimates and the remaining ones use realistic treatment, the balancing of resources expected from risk informed regulation is not as well supported as when all applications aspire for a comparable level of realism.

A.5 TREATMENT OF AUGMENTED PROGRAM ELEMENTS

CPS has a total of 42 Class 1 IGSCC Category B through G welds. From the 42 Class 1 welds, 28 welds were removed from the RISI element selection population since no other damage mechanism was identified. The remaining 14 Class 1 IGSCC Category B through G welds are included in the RISI element selection population. Of the 14 Class 1 welds remaining in the RISI element selection population, 4 welds are selected under the RISI program, therefore they are credited in both the RISI and IGSCC programs. When inspections are credited under the RISI and IGSCC programs, all inspection requirements for both programs are met. The Class 1 welds removed from the RISI selection population continue to be addressed by the IGSCC program.

FAC elements, which have no other degradation mechanism, are modeled and inspected in accordance with the FAC program. Inspection locations within a FAC element are selected in accordance with the FAC program. The extent of examination for selected inspection points is in accordance with Section 4.7, "Flow Accelerated Corrosion" of EPRI TR 112657. Welds identified as having FAC as the only degradation mechanism are removed from the RISI population for element selection. FAC-only welds currently inspected under Section XI will not be selected for inspection under the RISI program, but will continue to be addressed by the FAC program.

CPS has a total of 98 welds identified as having FAC as the only degradation mechanism. The 98 FAC-only welds were removed from the element selection population and no RISI exams were selected for any of these welds. CPS has 100 welds identified as having FAC and at least one other damage mechanism. These welds remained in the element selection population. Of the 100 welds remaining in the population, 14 Risk Category 1 welds and 13 Risk Category 3 welds were selected for examination under the RISI program.

The FAC-only and IGSCC welds that are not included in the selection population for the RISI program are all included in the delta risk calculations. Those Section XI examinations eliminated at any of these welds would result in a slight increase in risk for those specific welds and contribute to the overall delta risk that was quantified for the system.

A.6 DISTRIBUTION OF RISI EXAMS BY PIPE CLASS AND SYSTEM

The table below summarizes the RISI inspections by risk category, system and piping class. Augmented inspections for FAC and IGSCC are not credited in the RISI program.

RISI Examinations			
Risk Category	**System**	**Class 1**	**Class 2**
1	PFW	14	
2	AAI	4	
	PHP	1	
	PRI	1	
3	PFW	4	
	PRH		9
4	AAI	1	
	PHP	4	
	PLP	1	
	PMS	8	
	PRH	1	
	PRI	1	
	PRR	14	
5	AAP	1	
	PLP	1	
	PRH	2	1
	PRI	6	
Total		64	10

A.7 REFERENCES

[A7-1] EPRI, "Revised Risk Informed Inservice Inspection Procedure," EPRI Report No. TR-112657 Rev. B, July 1999.

[A7-2] USNRC Staff Evaluation of EPRI RISI Procedure (included in Reference [A7-1]).

[A7-3] 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.

[A7-4] 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.

[A7-5] 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.

[A7-6] USNRC Regulatory Guide 1.178, "An Approach for Plant Specific Risk-informed Decision Making: Inservice Inspection for Piping," July 1998.

[A7-7] US NRC Regulatory Guide 1.174, "An Approach For Using Probabilistic Risk Assessment In Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," July 1998.