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

Subject: Braidwood Station Interval 2 Inservice Inspection Program: Relief Request I2R-39, Alternative to the ASME Boiler and Pressure Vessel Code Section XI Requirements for Class 1 and Class 2 Piping Welds

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)"

As required by 10 CFR 50.55a(3), Braidwood is submitting, for U. S. Nuclear Regulatory Commission (NRC) approval, a proposed alternative to the 1989 edition of the 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 Braidwood Station 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).

The use of a Risk-Informed Inservice Inspection (ISI) Program for the selection and examination of Class 1 and Class 2 piping welds is being proposed under 10 CFR 50.55a(a)(3)(i). The discussion contained in the attachment to Relief Request I2R-39, "Risk-Informed Inservice Inspection Evaluation - Braidwood Station Units 1 and 2," demonstrates that the proposed alternative would provide an acceptable level of quality and safety. The format of Braidwood Station's Risk Informed ISI submittal is consistent with the Nuclear Energy Institute and industry template developed for applications of the Risk Informed ISI methodology. Additional supporting documentation is available at Braidwood Station for NRC review.

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Braidwood Station intends to incorporate the risk based approach to the selection and examination of Class 1 and Class 2 piping welds in the First Period of the Second Inservice Inspection Interval for Units 1 and 2. Braidwood is currently in the First Period of the Second Inservice Inspection Interval which is effective from July 29, 1998 through July 28, 2008 for Braidwood Unit 1 and effective from October 17, 1998 through October 16, 2008 for Braidwood Unit 2. The Risk Informed ISI examinations and the balance of the ASME Section XI examinations will be integrated into a single program having the same interval and period start and end dates. The start and end dates of the Braidwood Station ASME Code Periods are as follows:

Unit 1, Period 1 (July 29, 1998 through July 28, 2002*)
Unit 1, Period 2 (July 29, 2001 through July 28, 2005)
Unit 1, Period 3 (July 29, 2005 through July 28, 2008)
Unit 2, Period 1 (October 17, 1998 through October 16, 2002*)
Unit 2, Period 2 (October 17, 2001 through October 16, 2005)
Unit 2, Period 3 (October 17, 2005 through October 16, 2008)

* Using the IWB-2412(b) allowance, examinations performed during Braidwood Station outages A1R09 (April 2002) and A2R09 (September 2001) will be credited for Period 1 examinations.

Braidwood will schedule and credit both risk informed and the balance of the ASME Code exams consistent with ASME Section XI requirements, (i.e., a minimum of 16%, 50%, and 100% of required exams for each of the 3 periods and a maximum of 34%, 67% and 100% per period). Selected Risk Category 2, 3, 4, and 5 welds that have been examined in Period 1 prior to approval of the RISI program will be credited in the RISI program.

In order to effectively incorporate the risk informed program into the remainder of the First Inspection Period, Braidwood Station is requesting resolution of the proposed alternative by March, 2001.

Please direct any questions you may have regarding this submittal to Mr. T. W. Simpkin, Regulatory Assurance Manager, at (815) 458-2801, x2980.

Sincerely,

Timothy J. Tulon
Site Vice President
Braidwood Station

Attachments: 1) Relief Request I2R-39, 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 Program Plan - Braidwood Station Units 1 and 2

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

Attachment 1

Relief Request I2R-39

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

RELIEF REQUEST I2R-39
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 (1989 Edition) 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 - Braidwood 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.

RELIEF REQUEST I2R-39
REVISION 0

CODE REQUIREMENT (continued)

(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

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:

1. 7.5% but not less than 28 welds (per Category) of all welds not exempted by IWC-1220 as modified 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.

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

RELIEF REQUEST I2R-39
REVISION 0

CODE REQUIREMENTS FROM WHICH RELIEF IS REQUESTED

Relief is requested from the ASME Section XI, 1989 edition, 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. As stated in the Reference (2) evaluation:

> "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 10 CFR 50.55a for the proposed alternative to the piping ISI requirements with regard to the number of locations, locations of inspections, and methods of inspection."

PROPOSED ALTERNATE PROVISIONS

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

Braidwood Station's application of the Risk Informed ISI, 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 ISI program. Also, visual examination of piping supports and integral attachments will continue to be performed on Class 1, 2 and 3 systems in accordance with the ASME Section XI program.

RELIEF REQUEST I2R-39
REVISION 0

PERIOD FOR WHICH RELIEF IS REQUESTED

Braidwood Station intends to incorporate the risk based approach to the selection and examination of Class 1 and Class 2 piping welds in the First Period of the Second Inservice Inspection Interval for Units 1 and 2 therefore Relief is requested for Periods 1, 2 and 3 of the second ten-year inspection interval of the Inservice Inspection Program. Selected Risk Category 2, 3, 4, and 5 welds that have been examined in Period 1 prior to approval of the RISI program will be credited in the RISI program.

Attachment 2

Risk-Informed Inservice Inspection Evaluation - Braidwood Station Units 1 and 2

Risk Informed Inservice Inspection
Program Plan

Braidwood Station

Units 1 and 2

Table of Contents

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 ASME Section XI based 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 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.

ComEd plans to incorporate the RISI inspection program during the first period of the second inspection interval for Braidwood Station Units 1 and 2. The second 10-year inspection interval began for Units 1 and 2 on July 29, 1998, and October 17, 1998, respectively.

As a risk-informed application, this submittal meets the intent and principles of Regulatory Guides 1.174 and 1.178 as well as those set forth in the EPRI RISI Topical Report.

<u>PRA Quality</u>

The NRC Staff reviewed the Braidwood PRA relative to the requirements in NRC Generic Letter 88-20. The NRC Staff Evaluation Report issued in October 1997 stated that "...the staff finds that the licensee's RE [risk evaluation] is complete with regard to the information requested by GL 88-20 (and associated guidance, NUREG-1335) and concludes that the licensee's RE 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 although stating it was consistent with the intent of GL 88-20. ComEd has since enhanced the Braidwood 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 Braidwood PRA since the NRC Staff Evaluation Report was issued in October 1997. This upgrade of the Braidwood PRA was done in conjunction with an upgrade of the Byron PRA and has produced PRAs of comparable quality. Braidwood and Byron Nuclear Plants are sister plants with nearly identical designs. Hence, the Braidwood and Byron 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 of Braidwood's and Byron's emergency diesel generators. The NRC Safety Evaluation Report (SER) 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 Braidwood PRA was graded 3 in all eleven elements with eight grades being contingent on further specified enhancements. Following implementation of significant enhancements to both the Braidwood and Byron PRAs as noted above, 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 even 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 Braidwood PRA fully supports the Braidwood 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 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-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 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, "Thermal Stresses in Piping Connected to Reactor Coolant Systems," and 88-11, "Pressurizer Surge Line Thermal Stratification," and Information Notice 93-20, "Thermal Fatigue Cracking of Feedwater Piping to Steam Generators" are all addressed by the evaluation of thermal fatigue that is part of the degradation assessment for RISI. These augmented programs are therefore subsumed in the RISI program. The following augmented programs were not subsumed into the RISI program and remain unaffected:

- Stagnant Borated Water Systems (IE Bulletin 79-17)
- 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. However, for the socket welds selected in the RISI, the examination method (VT-2) and examination frequency (each refueling outage) defined in Code Case N-578-1, Table 1 will be used. 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 or Low) 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 at ComEd 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 4).

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 and ASME Code Case N-578-1 (Reference 7) were used as guidance in determining the examination requirements for these locations with Code Case N-578-1 used to clarify the examination method and examination frequency for selected socket welds. From the Class 1 butt welded elements that were considered within the scope of the RISI evaluation at Unit 1, a total of 8.9% were selected for volumetric examination as part of the risk informed inspection program. Because the total of volumetric examinations was less than 10%, a review of the Unit 1 element population per Topical Report Section 3.6.4.2 was performed and the total selected for volumetric examination was determined to be reasonable. For Unit 2, the percentage of Class 1 butt welded elements that were selected for volumetric examination was 10.1%. 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 program 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. Also, visual examination of piping supports and integral attachments will continue to be performed on Class 1, 2 and 3 systems in accordance with the ASME Code Section XI program.

Additional Examinations

Since the RISI program may require examinations on a number of elements constructed to lesser preservice inspection requirements, the program in all cases will determine through an engineering evaluation the root cause of any unacceptable flaw or relevant condition found during examination. The evaluation will include the applicable service conditions and degradation mechanisms to establish that the element(s) will still perform their intended safety function during subsequent operation. Elements not meeting this requirement will be repaired or replaced.

The evaluation will include a determination of whether other elements in the segment or segments are subject to the same root cause conditions. Additional examinations will be performed on these elements up to a number equivalent to the number of elements required to be inspected on the segment or segments initially. If unacceptable flaws or relevant conditions are again found similar to the initial problem, the remaining elements identified as susceptible will be examined. No additional examinations will be performed if there are no additional elements identified as being susceptible to the same root cause conditions.

3.6 Program Relief Requests

At this time, the selected risk-informed examination locations provide >90% coverage. 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.

With the approval of RISI, the following Braidwood Station Station Relief Requests will no longer be required:

- Relief Request I2R-01, "Alternative Examination for Branch Pipe Connections Designed with Reinforcement Saddles"
- Relief Request I2R-03, "Limited Examination of Reactor Coolant System Piping Weld Due to Whip Restraint Obstructions"
- Relief Request I2R-04, "Limited Volumetric Examination of Main Steam and Feedwater System Welds Due to Ultrasonic Scanning Obstructions"

These three relief requests were approved in a Safety Evaluation Report contained in the letter from A. J. Mendiola (U.S. NRC) to O.D. Kingsley (ComEd), "Evaluation of the Second 10-Year Interval Inservice Inspection Program Plan Requests for Relief for Braidwood Station, Units 1 and 2," dated January 6, 2000.

3.7 Risk Impact Assessment

The RISI program has been conducted in accordance with Regulatory Guide 1.174 and the EPRI methodology requirements consistent with Regulatory Guide 1.178, 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 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 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 4). 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 for the remaining systems. The updated failure rates and rupture frequencies developed in this evaluation account for service data in Westinghouse PWRs covering over 1,000 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 (Reference 6) 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 Braidwood 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 Braidwood 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. As seen in these figures and table, most of the systems exhibited small increases in CDF and LERF but these increases are much smaller than the risk acceptance criteria by a large margin.

A similar set of results is presented in Table 9 and Figures 4 and 5 for Unit 2. Not shown in these figures and tables are the results for the containment purge system. Since there were no elements selected in either the current or RISI program for the system, there were no risk impacts to consider. 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.

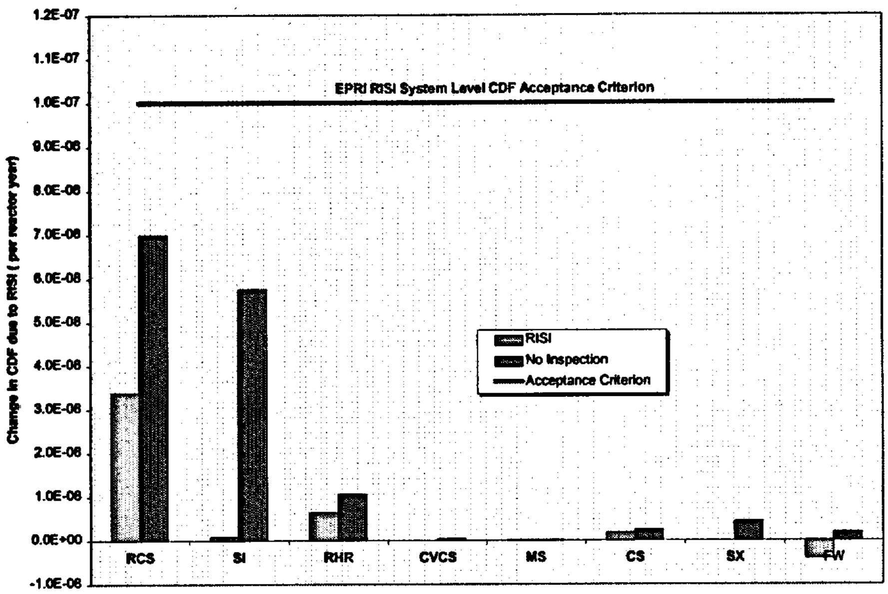

Figure 2
Change in Pipe Rupture CDF for Braidwood Station Unit 1 Systems

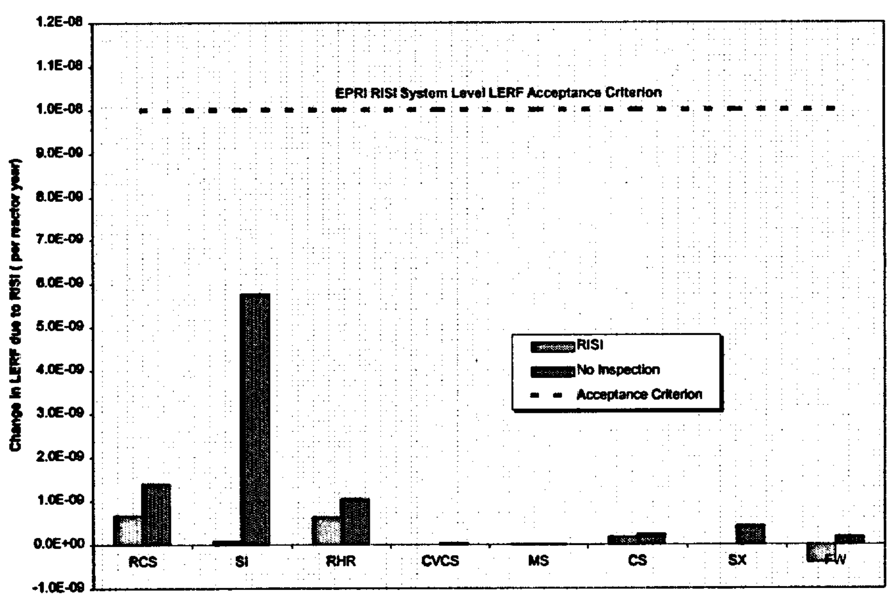

Figure 3
Change in Pipe Rupture LERF for Braidwood Station Unit 1 Systems

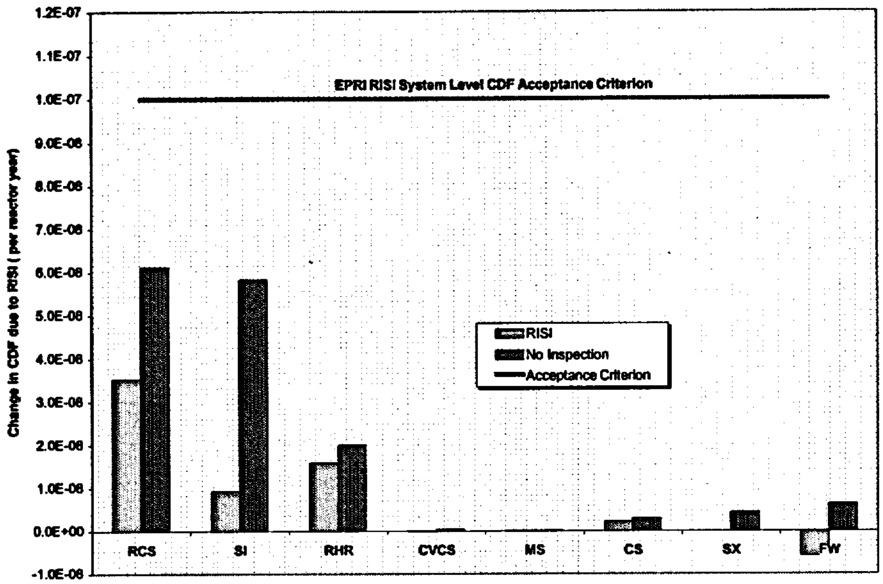

Figure 4
Change in Pipe Rupture CDF for Braidwood Station Unit 2 Systems

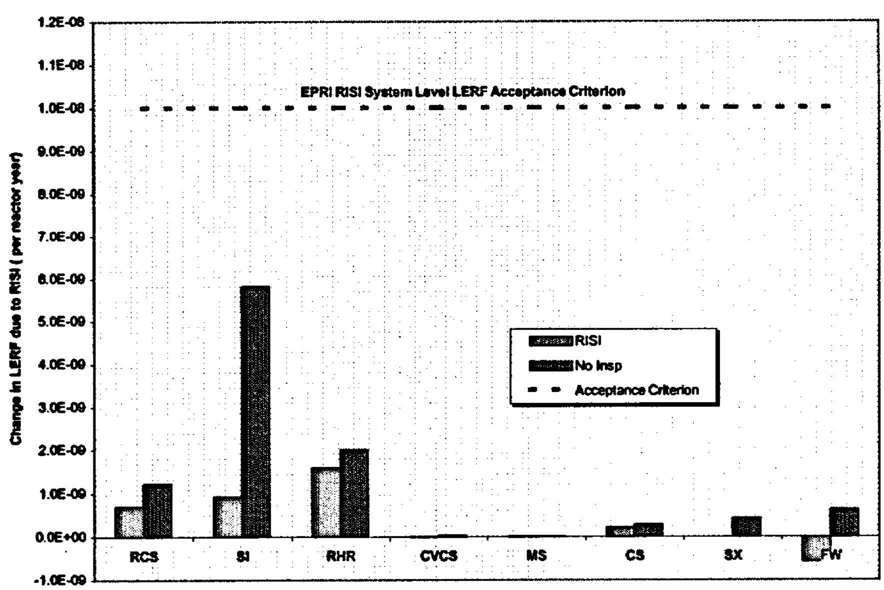

Figure 5
Change in Pipe Rupture LERF for Braidwood Station Unit 2 Systems

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 first inspection period of the second inservice inspection interval for Braidwood Station Units 1 and 2. Selected Risk Category 2, 3, 4, and 5 welds that have been examined in Period 1 prior to approval of the RISI program will be credited in the RISI program. 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 Braidwood 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 Braidwood 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 537 Section XI program exams to 199 RISI program exams, a net reduction of 338 exams (63% reduction in number of exams). Unit 2 is reduced from 530 exams to 228 exams, a net reduction of 302 exams (57% 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 Main Feedwater systems at Units 1 and 2 and the CVCS at Unit 2 exhibited a net increase in RISI exams in the High and Medium risk categories and were found to have a small reduction in CDF and LERF due to RISI changes. The Essential Service Water (SX) system had no changes in the RISI program and, hence, no change in CDF and LERF. The remaining systems exhibited small decreases in the number of exams in High and Medium risk locations with small increases in CDF and LERF due to the RISI changes. As noted previously, these increases 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 BRW-99-0136-N, CDF Results for Braidwood Units 1 and 2.
3. ComEd Calculation BRW-99-0324-N, LERF Results for Braidwood Units 1 and 2.
4. ComEd Risk Informed Inservice Inspection Evaluation, Braidwood Nuclear Power Plant Units 1 and 2 – Final Report, May 2000 (Draft).
5. 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*

6. 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*

7. ASME Code Case N-578-1, "Risk-Informed Requirements for Class 1, 2, and 3 Piping, Method B, Section XI, Division 1 approved by the ASME Main Committee, February 2000.

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

System Description[1]	Number of Segments
Containment Spray System (CS)	6 / 8
Chemical and Volume Control System (CV)	35 / 34
Reactor Coolant System (RC, RY)[2]	125 / 124
Residual Heat Removal System (RH)	19 / 18
Safety Injection System (SI)	84 / 89
Main Feedwater System (FW)	27 / 23
Main Steam System (MS)	48 / 48
Essential Service Water System (SX)	16 / 12
Containment Purge System (VQ)	5 / 5
Total	365 / 361

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 thermowells and pressurizer relief piping.

NOTE: This table shows the number of pipe segments from each system that are Class 1 or Class 2 category B-J, B-F, 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									
RY	X	X				X					
RH											
SI	X	X	X								
FW	X	X									X
MS											
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				40			40
CV				77	138	114	329
RC		149		460		13	622
RY		41		83			124
RH				53		85	138
SI				128	174	558	860
FW			128		4		132
MS						176	176
TOTAL	0	190	128	841	316	946	2421

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				46			46
CV				84	137	123	344
RC		153		483		13	649
RY		41		73			114
RH				49		83	132
SI				135	153	507	795
FW			236				236
MS						175	175
TOTAL	0	194	236	870	290	902	2491

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 at Unit 1 that has not occurred yet at Unit 2, 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							16	4				0	16	4
CV							8	8	17	14	15	0	40	22
RC			85	38			104	46				0	189	84
RY			32	11			18	8				0	50	19
RH							15	6			24	0	39	6
SI							17	13	38	18	123	0	178	31
FW					11	32			1	1		0	12	33
MS											13	0	13	0
TOTAL	0	0	117	49	11	32	178	85	56	33	175	0	537	199

NOTE: 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.

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							16	5				0	16	5
CV							8	9	12	14	36	0	56	23
RC			56	39			93	48			5	0	154	87
RY			31	11			20	8				0	51	19
RH							17	5			24	0	41	5
SI							19	14	43	16	122	0	184	30
FW					13	59						0	13	59
MS											15	0	15	0
TOTAL	0	0	87	50	13	59	173	89	55	30	202	0	530	228

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 Braidwood 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⟨R|F⟩	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⟨R|F⟩	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⟨R|F⟩	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⟨R|F⟩	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⟨R|F⟩, are dimensionless

Table 8
Impact of RISI and No Inspections on CDF and LERF due to Pipe Ruptures for Braidwood Unit 1 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	2.39E-07	2.73E-07	3.09E-07	3.38E-08	6.99E-08	<1.0E-07	6.76E-10	1.40E-09	<1.0E-08
SI	1.00E-06	1.00E-06	1.06E-06	8.45E-10	5.76E-08	<1.0E-07	8.46E-11	5.76E-09	<1.0E-08
RHR	5.45E-08	6.09E-08	6.50E-08	6.37E-09	1.05E-08	<1.0E-07	6.40E-10	1.05E-09	<1.0E-08
CVCS	6.07E-09	6.08E-09	6.45E-09	1.36E-12	3.71E-10	<1.0E-07	1.36E-13	3.71E-11	<1.0E-08
MS	1.89E-09	1.97E-09	1.97E-09	7.92E-11	7.92E-11	<1.0E-07	7.92E-12	7.92E-12	<1.0E-08
CS	8.05E-09	9.81E-09	1.04E-08	1.76E-09	2.35E-09	<1.0E-07	1.76E-10	2.35E-10	<1.0E-08
SX	1.41E-08	1.41E-08	1.84E-08	0	4.33E-09	<1.0E-07	0	4.33E-10	<1.0E-08
FW	2.58E-08	2.18E-08	2.76E-08	-4.00E-09	1.85E-09	<1.0E-07	-4.00E-10	1.85E-10	<1.0E-08
Total	1.35E-06	1.39E-06	1.50E-06	3.89E-08	1.47E-07	<1.0E-07	3.89E-09	1.47E-08	<1.0E-07

Table 9
Impact of RISI and No Inspections on CDF and LERF Due to Pipe Ruptures for Braidwood 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	2.44E-07	2.79E-07	3.05E-07	3.53E-08	6.11E-08	<1.0E-07	7.06E-10	1.22E-09	<1.0E-08
SI	9.60E-07	9.69E-07	1.02E-06	9.26E-09	5.82E-08	<1.0E-07	9.26E-10	5.82E-09	<1.0E-08
RHR	5.40E-08	6.99E-08	7.40E-08	1.59E-08	2.00E-08	<1.0E-07	1.59E-09	2.01E-09	<1.0E-08
CVCS	6.79E-09	6.74E-09	7.15E-09	-4.83E-11	3.63E-10	<1.0E-07	-4.83E-12	3.63E-11	<1.0E-08
MS	1.86E-09	1.96E-09	1.96E-09	9.25E-11	9.25E-11	<1.0E-07	9.25E-12	9.25E-12	<1.0E-08
CS	9.03E-09	1.12E-08	1.20E-08	2.20E-09	2.93E-09	<1.0E-07	2.20E-10	2.93E-10	<1.0E-08
SX	1.29E-08	1.29E-08	1.72E-08	0	4.35E-09	<1.0E-07	0	4.35E-10	<1.0E-08
FW	5.21E-08	4.64E-08	5.85E-08	-5.76E-09	6.33E-09	<1.0E-07	-5.76E-10	6.33E-10	<1.0E-08
Total	1.34E-06	1.40E-06	1.49E-06	5.69E-08	1.53E-07	<1.0E-06	5.69E-09	1.53E-08	<1.0E-07