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May 18, 2001

10CFR50.55a(a)(3)(i)

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

LaSalle County Station, Units 1 and 2
Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

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"

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

In accordance with 10 CFR 50.55a(a)(3)(i), Exelon Generation Company (EGC), LLC is submitting, for U. S. Nuclear Regulatory Commission (NRC) review and approval, 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 LaSalle County Station uses the Reference 1 methodology for a Risk-Informed Inservice Inspection (RI-ISI) program approved by the NRC to the extent and within the limitations specified in Reference 2.

Relief Request CR-35 and the RI-ISI Program Plan Summary are attached and demonstrate that the proposed alternative would provide an acceptable level of quality and safety, as required by 10CFR 50.55a (a)(3)(i). The format of LaSalle County Station's RI- ISI submittal is consistent with the Nuclear Energy Institute (NEI) and industry template developed for applications of the RI-ISI methodology.

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The RI-ISI program will be incorporated for the entire second and third periods of the second Inservice Inspection Interval for both Unit 1 and Unit 2. For Unit 1, the second Inservice Inspection Interval began on November 23, 1994 and the projected end date is October 11, 2006. For Unit 2, the second Inservice Inspection Interval began on October 17, 1994 and the projected end date is July 4, 2007.

Implementation of this RI-ISI program will reduce the number of ASME Section XI piping weld inspections by over 60 percent with little change in the risk to the public, while reducing occupational radiation exposure.

100 percent of the RI-ISI piping weld inspection locations will be inspected over the current second Inservice Inspection Interval by either the current ASME Section XI ISI program or by the proposed RI-ISI program.

Approval of the proposed alternative is requested by October 1, 2001, to support the refueling outage scheduled in the fall of 2001.

Should you have any questions concerning this letter, please contact Mr. William Riffer, Regulatory Assurance Manager, at (815) 357-6761, extension 2383.

Respectfully,

Charles G. Pardee
Site Vice President
LaSalle County Station

Attachments 1) Relief Request CR-35

2) Risk-Informed Inservice Inspection Program Plan Summary– LaSalle County Station, Units 1 and 2

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

Attachment 1

Relief Request CR-35

Relief Request CR-35:
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, B5.130, B5.150, B9.11, B9.12, B9.21, B9.31, B9.32, B9.40, C5.11, C5.12 C5.41, C5.51, C5.52, C5.70, and C5.81
Description:	Alternate Selection and Examination Criteria for Category B-F, B-J, C-F-1, and C-F-2 Pressure Retaining Piping Welds
Component Number:	All welds in American Society of Mechanical Engineers (ASME) Section XI Code Categories B-F, B-J, C-F-1, and C-F-2

References:

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

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

3) Risk-Informed Inservice Inspection Evaluation – LaSalle County Station Units 1 and 2, Volumes 1, 2, and 3 (Dated February 2001)

4) American Society of Mechanical Engineers (ASME) Code Case N-578-1, "Risk-Informed Requirements for Class 1, 2, or 3 Piping, Method B"

Relief Request CR-35:
Revision 0

CODE REQUIREMENT

Table IWB 2500-1, Examination Category B-F, requires a volumetric and/or surface examination on all welds for Items B5.10, B5.130, and B5.150.

Table IWB 2500-1, Examination Category B-J, requires a volumetric and/or surface examination on welds for Items B9.11, B9.12, B9.21, B9.31, B9.32, and B9.40.

The Category B-J weld population selected for inspection includes the following.

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

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

c. All dissimilar metal welds between combinations of the following.
   (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.

d. Additional piping welds so that the total number of circumferential butt welds (or branch connection or socket welds) selected for examination equals 25% of the circumferential butt welds (or branch connection or socket welds) in the reactor coolant piping system. This total does not include welds excluded by IWB-1220.

Table IWC-2500-1, Examination Categories C-F-1 and C-F-2 require volumetric and/or surface examinations for items C5.11, C5.12, C5.41, C5.51, C5.52, C5.70, and C5.81.

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Revision 0

CODE REQUIREMENT (Continued)

The Category C-F-1 and C-F-2 weld population selected for inspection includes the following.

> Welds selected shall include 7.5%, but not less than 28 welds of all austenitic or high alloy steel welds (Category C-F-1) or of all carbon and low alloy steel welds (Category C-F-2) not exempted by IWC-1220. Some welds not exempted by IWC-1220 are not required to be nondestructively examined per Examination Categories C-F-1 and C-F-2. These welds, however, shall be included in the total weld count to which the 7.5% sampling rate is applied. The examinations shall be distributed as follows.
>
> a. The examinations shall be distributed among the Class 2 systems prorated, to the degree practicable, on the number of nonexempt austenitic or high alloys welds (Category C-F-1) or carbon and low alloy welds (Category C-F-2) in each system (i.e., if a system contains 30% of the nonexempt welds, then 30% of the nondestructive examinations required by the Examination Category (C-F-1 or C-F-2) shall be performed on that system),
>
> b. within a system, the examinations shall be distributed among terminal ends and structural discontinuities prorated, to the degree practicable, on the number of nonexempt terminal ends and structural discontinuities in the system, and
>
> c. within each system, examinations shall be distributed between line sizes prorated to the degree practicable.

BASIS FOR RELIEF

Pursuant to 10 CFR 50.55a(a)(3)(i), relief is requested on the basis that the proposed alternative of utilizing the examination methodology and selection criteria of Reference 1 along with evaluation and sample expansion requirement enhancements identified in ASME Code Case N-578-1, "Risk-Informed Requirements for Class 1, 2, or 3 Piping, Method B", Reference 4, will provide an acceptable level of quality and safety.

Relief Request CR-35:
Revision 0

BASIS FOR RELIEF (Continued)

In Reference 2, 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 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."

In lieu of the evaluation and sample expansion requirements of Section 3.6.6.2, "RI-ISI Selected Examinations," contained in Reference 1, LaSalle County Station (LaSalle) will utilize the requirements of Subsubarticle-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.

PROPOSED ALTERNATE PROVISIONS

The proposed alternative described in Attachment 2 to this submittal, "Risk Informed Inservice Inspection Program Plan Summary, LaSalle County Station Units 1 and 2," provide an acceptable level of quality and safety as required by 10 CFR 50.55a(a)(3)(i).

Our use of the Reference 1 RI-ISI program at LaSalle, 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 EPRI TR-112657 while the guidance for the examination method 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 LaSalle pressure and leak test program, which remains unaffected by the RI-ISI program.

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APPLICABLE TIME PERIOD

LaSalle plans to incorporate the RI-ISI program for the entire second and third periods of the second Inservice Inspection Interval for both Unit 1 and Unit 2. For Unit 1, the second Inservice Inspection Interval began on November 23, 1994 and the projected end date is October 11, 2006. For Unit 2, the second Inservice Inspection Interval began on October 17, 1994 and the projected end date is July 4, 2007. For Unit 1 the current second inspection period began on October 12, 1999 and the projected end date is October 11, 2003. For Unit 2 the current second inspection period began on July 5, 2000 and the projected end date is July 4, 2004.

Attachment 2

Risk Informed Inservice Inspection Program Plan Summary

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

The plan is to incorporate the RI-ISI inspection program during the second period of the second inspection interval for LaSalle County Station (LaSalle), Units 1 and 2. The second Inservice Inspection Interval started on November 23, 1994 for LaSalle Unit 1, and the projected end date is October 11, 2006 including all extensions currently being taken. The second Inservice Inspection Interval started on October 17, 1994 for LaSalle Unit 2, and the projected end date is July 4, 2007 including all extensions currently being taken.

As a risk-informed application, this submittal is consistent with the guidance of Regulatory Guides (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis," and 1.178, "An Approach for Using Probabalistic Risk Assessment In Risk Informed Decisions on Inservice Inspection of Piping," as well as those set forth in the EPRI RI-ISI Topical Report.

PRA Quality

The LaSalle Probabalistic Risk Assessment (PRA) used for the risk determinations for this submittal are recent upgrades to the "Modified Individual Plant Examination (IPE)," submitted to the NRC by letters dated April 28, 1994 and December 12, 1994. The modified IPE had been accepted by the NRC by Staff Evaluation Report (SER) letter dated March 14, 1996. The NRC letter noted that the modified IPE submittal met the intent of Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities – 10 CFR 50.54(f)," dated November 23, 1988.

The original LaSalle PRA was performed by Sandia Labs, on behalf of the NRC, as part of the Risk Methods Integration and Evaluation Program (RMIEP) Study (NUREG/CR-4832). The current LaSalle PRA is a third generation upgrade to that study. The LaSalle PRA addresses internal events at full power, and it includes internal flooding, as well as certain other external events. Internal fire risk is taken from the original LaSalle RMIEP Study, 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.

For the Level 2 analysis (i.e., the containment analysis), large early release frequency (LERF) was estimated using the methodology in NUREG/CR-6595, "An Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events," dated January 1999. This approach to LERF evaluation, while somewhat simplified, supports a realistic quantification of systemic contributions to containment isolation failures and bypass sequences that are actually derived from the Level 1 event sequence model.

Both the LaSalle PRA model and its supporting bases documentation were reviewed by a Boiling Water Reactor Owners Group (BWROG) Peer Review/Certification Team in the spring of 2000. The review was conducted using Nuclear Energy Institute (NEI) NEI 00-02, "NEI PSA Certification Peer Review Process," and a team of industry PRA experts. This independent review was performed to evaluate the quality of the PRA and completeness of the PRA documentation. The Certification Team found that the LaSalle PRA was a sound model, and its element grades demonstrate that it is adequate for use in regulatory submittals.

The NEI PSA certification process assesses a PRA in eleven functional elements. Each element is graded on a scale of 1 to 4. A grade of 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 LaSalle PRA was graded 3 in all eleven of the PRA elements.

Exelon Generation Company (EGC), LLC 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. The last update of the LaSalle PRA was completed in the spring of 2000.

Based on the results of past NRC reviews and the BWROG Certification Peer Review, we believe that the level of detail and quality of the LaSalle PRA fully supports this risk-informed regulatory application.

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 RI-ISI Program

The alternative RI-ISI program for piping is described in EPRI RI-ISI Topical Report (Reference 1). The RI-ISI 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 RI-ISI 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 RI-ISI program. In Table 6-2 of the EPRI Topical Report, the issues raised by NRC Bulletin 88-08 are addressed by the evaluation of thermal fatigue that is part of the degradation assessment for RI-ISI. These augmented programs are therefore subsumed in the RI-ISI program. The following augmented programs were not subsumed into the RI-ISI program and remain unaffected:

- IGSCC in Boiling Water Reactor (BWR) Austenitic Stainless Steel Piping (Generic Letter 88-01 and NUREG-0313), except for Category A weldments
- 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 whether the affected piping was subject to damage mechanisms other than those addressed by the augmented program. Elements that were determined to be only susceptible to Flow-Accelerated Corrosion (FAC) or Intergranular Stress Corrosion Cracking (IGSCC) (Categories B thru G) were not included in the element selection. These elements will remain part of the FAC or IGSCC programs which already perform inspections to detect these degradation mechanisms. Those elements susceptible to FAC or IGSCC along with another degradation mechanism (e.g., thermal fatigue) were retained as a part of the RI-ISI scope and included in the element selection process for the purpose of performing exams to detect the additional degradation mechanism. 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 RI-ISI methodology.

2.4 Multiple Damage Mechanisms

The vast majority of pipe elements that were evaluated in the RI-ISI evaluation were found to be susceptible to none of the damage mechanisms addressed in the EPRI RI-ISI 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 High Pressure and Low Pressure Core Spray Systems 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 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 RI-ISI 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 RI-ISI program is consistent with the methodology described in the EPRI Topical Report for ASME Code Case N-578-1, "Risk-Informed Requirements for Class 1, 2, or 3 Piping, Method B Section XI, Division 1" (Reference 4) applications. The process involves the following steps:

- Definition of RI-ISI 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 RI-ISI Program Scope

The systems to be included in the RI-ISI 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 isometric and piping and instrumentation diagrams and additional plant information were used to define the system boundaries. The RI-ISI 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 RI-ISI Topical Report. One of the enhancements that was incorporated into this application of the EPRI RI-ISI 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 RI-ISI evaluation, in lieu of the consequence tables in the EPRI Topical Report. This step was taken to reduce some of the conservatism inherent in the consequence tables and to support a more complete and realistic quantification of the risk impacts of the RI-ISI program in comparison with previous applications of this methodology. Another motivation was to increase consistency with other risk informed applications at EGC 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 RI-ISI evaluation using information in Reference 3 and described in the Tier 2 documentation (Reference 2).

3.4 Risk Categorization

In the preceding steps, each element within the scope of the RI-ISI 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 RI-ISI risk categories per system are summarized in Table 3 and Table 4 for LaSalle Unit 1 and Unit 2, respectively.

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 RI-ISI Matrix for Risk Ranking of Pipe Segments (Reference 1)

3.5 Inspection Location Selection and NDE Selection

In general, an ASME Code Case N-578-1 application of RI-ISI, per the EPRI RI-ISI 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 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.

Welds having only IGSCC or only FAC as a degradation mechanism were removed from the population for element selection prior to applying the 25% and 10% sampling percentages. The

following table provides a break down of welds with only IGSCC or only FAC as a degradation mechanism.

	IGSCC Only		FAC only
Unit	High Risk	Medium Risk	High Risk
1	136	11	278
2	125	9	288

The results of the selection are presented in Tables 5 and 6 for LaSalle Units 1 and 2, respectively. Section 4 of the EPRI Topical Report and ASME Code Case N-578-1 were used as guidance in determining the examination methods and requirements for these locations.

For Unit 1, eleven (11) of the RI-ISI selected welds are IGSCC Category B through G welds and therefore inspections will be credited under both the RI-ISI and IGSCC programs. When inspections are credited under the RI-ISI and IGSCC programs, all inspection requirements for both programs are met. Similarly for Unit 2, twelve (12) welds will be credited under both the RI-ISI and IGSCC programs.

From the Class 1 butt welds elements within the scope of the RI-ISI program (welds having only IGSCC or only FAC as a degradation mechanism are not included in this calculation), a total of 12.8% were selected for volumetric examination on Unit 1 and 12.0% on Unit 2. When considering all Class 1 welds within the RI-ISI program the percentages (Class 1 butt welds examined, divided by the number of Class 1 butt and socket welds) become 12.0% for Unit 1 and 11.3% for Unit 2. Although some socket welds have been selected for examination, they are not included in the above percentages.

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.

Longitudinal welds are considered subsumed with examination of the associated circumferential weld when the circumferential weld is selected for RI-ISI examination. In accordance with footnote (4) of Code Case N-578-1, Table 1, Examination Category R-A, LaSalle will examine those longitudinal welds that intersect the circumferential welds selected under the risk-informed process. For those longitudinal welds intersecting circumferential welds, the portion of the weld within the associated circumferential weld volume will be inspected, and the inspection requirements for longitudinal weld will be met for both transverse and parallel flaws. This approach is consistent with the alternative method of scheduling and examining longitudinal welds provided in Code Case N-524, which has been approved for use in Regulatory Guide 1.147.

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 RI-ISI program.

Examination Method and Parts Examined

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

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

Additional Examinations

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

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

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

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

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

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

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

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

3.6 Program Relief Requests

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.

LaSalle will not need any additional relief requests or need to revise any relief request to implement the RI-ISI program.

3.7 Risk Impact Assessment

The RI-ISI program has been conducted in accordance with Regulatory Guides 1.174 and 1.178, and the EPRI Topical Report, 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 RI-ISI 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 RI-ISI evaluation. This is an enhancement that was made that goes well beyond the limited quantitative analyses that are needed to implement the methods 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 RI-ISI 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 due to proposed changes in the risk informed inspection program.

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 RI-ISI evaluation, in lieu of the qualitative analysis and bounding risk estimates that are permitted under most circumstances in the EPRI RI-ISI 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 RI-ISI 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 the EPRI TR-112657 methodology to ensure that the change in risk of implementing the RI-ISI program meets the requirements of RGs 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 RG 1.174 risk significant thresholds of $1x10^{-6}$ per year and $1x10^{-7}$ per year for changes in CDF and LERF for a full plant scope RI-ISI application.

The technical basis for the Markov model input parameters that were used in this evaluation are documented in the Tier 2 documentation (Reference 2). 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 3).

Separate Markov calculations were performed for the change in CDF and the change in LERF. This calculation was performed so pipe elements whose failure could create a potential containment failure or bypass concern were factored into the LERF evaluation. Due to the relatively high LERF to CDF ratios for these BWR reactor units, the change in LERF tended to be more limiting than the change in CDF evaluations when comparing the results to the EPRI RI-ISI risk significance thresholds. 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 RI-ISI 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 LaSalle Unit 1 are summarized in Table 7 and key aspects are plotted in Figures 2 and 3 for comparison against the risk significant criteria established in the EPRI RI-ISI Topical Report. A similar set of results is presented in

Table 8 and Figures 4 and 5 for Unit 2. As seen in these figures and tables, the Reactor Water Cleanup System (RWCU), Feedwater System (FW), and Emergency Core Cooling System (ECCS) groups for Unit 1 and the ECCS group for Unit 2 exhibited small decreases in CDF due to the changes from the RI-ISI program. The RWCU, FW, and ECCS groups for Unit 1 and the High pressure Core Spray (HPCS) and ECCS system groups for Unit 2 exhibited small decreases in LERF. The remaining systems evaluated across the two reactor units 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 by a large margin. Each system was found to have a change in LERF that is less than or equal to 2% of the EPRI RI-ISI risk significance threshold of $1x10^{-8}$/system-year, and a change in CDF that is less than 1% of the associated threshold of $1x10^{-7}$/system-year.

The total change in CDF and LERF due to the combined changes in the RI-ISI program for the entire scope of Class 1 and 2 systems are very small in relation to RG 1.174 risk significance criteria. The margin for these risk metrics is more than an order 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 RI-ISI 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 RI-ISI program for the entire scope of Class 1 and 2 piping that was included in this evaluation. This satisfies the risk significance criteria of RG 1.174 and the EPRI RI-ISI Topical Report.

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 pressure boundary. Currently, the process for selecting inspection locations is based upon structural discontinuity and stress analysis results. As depicted in ASME White Paper 92-01-01, "Evaluation of Inservice Inspection Requirements for Class 1, Category B-J Pressure Retaining Welds," Rev. 1, 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.

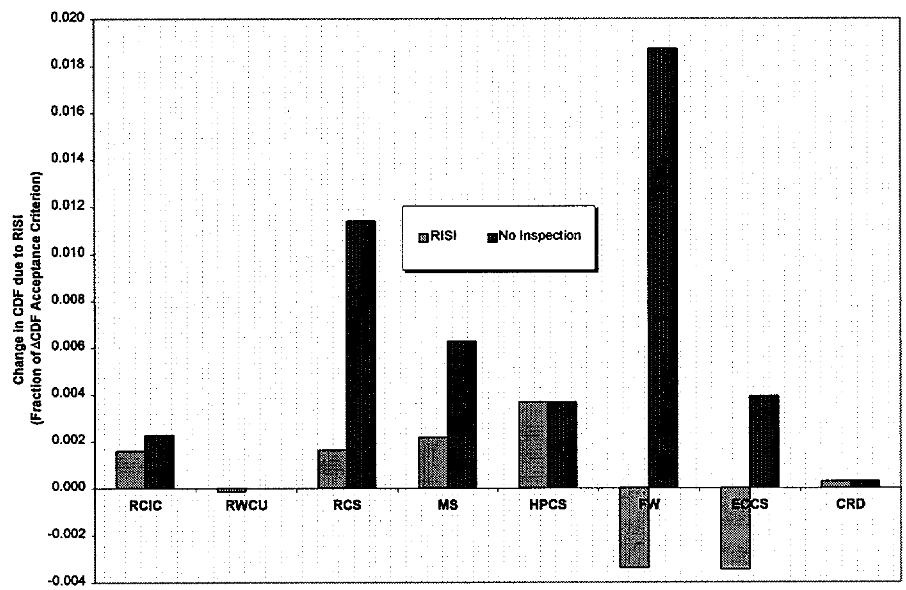

Figure 2
Change in Pipe Rupture CDF for LaSalle Unit 1 Systems

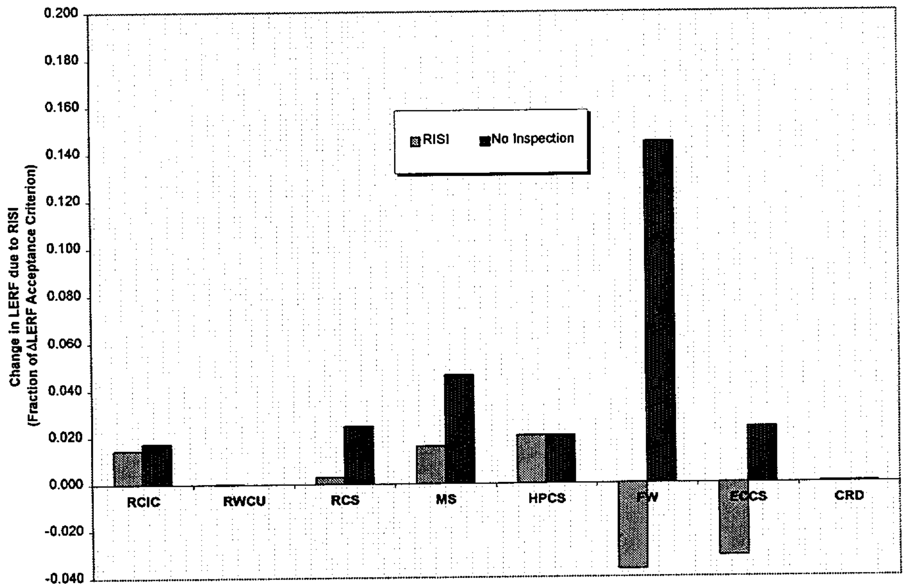

Figure 3
Change in Pipe Rupture LERF for LaSalle Unit 1 Systems

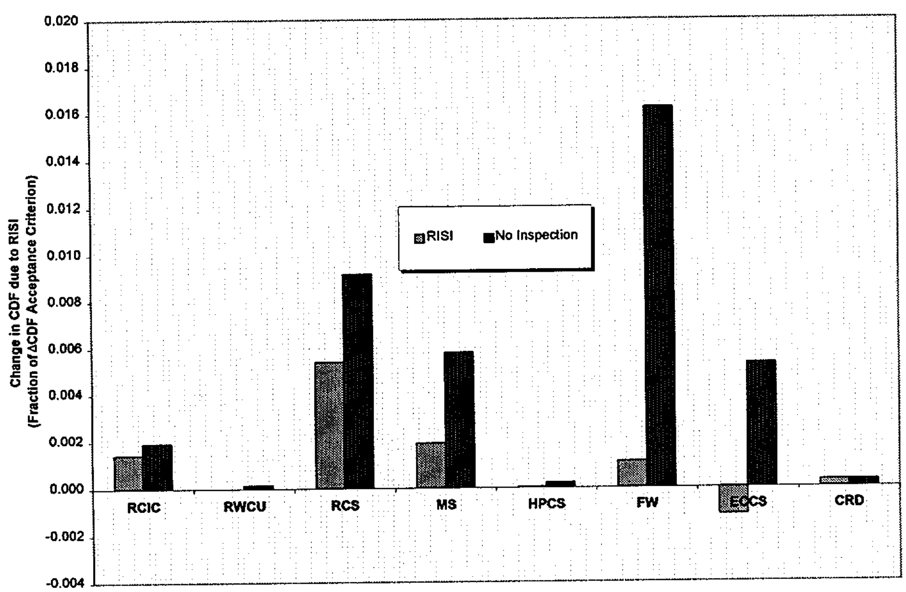

Figure 4
Change in Pipe Rupture CDF for LaSalle Unit 2 Systems

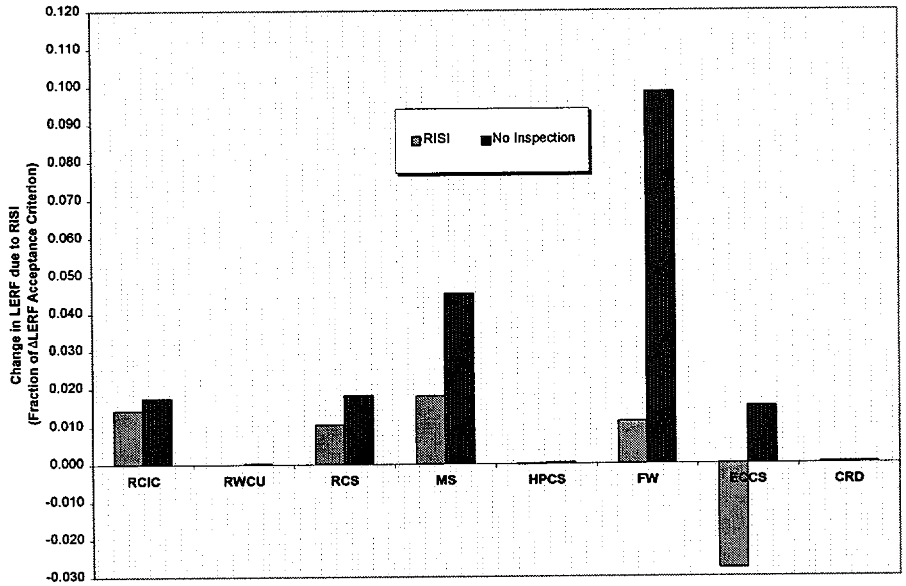

Figure 5
Change in Pipe Rupture LERF for LaSalle Unit 2 Systems

3.8 Sensitivity Studies

3.8.1. Sensitivity of Delta Risk Calculation Using Alternate Method

A simplified and conservative risk impact calculation, not using the Markov model calculation of pipe break frequency, was performed for LaSalle Units 1 and 2. This calculation was performed using the same approach as was implemented for the previously approved relief request for South Texas Project. The acceptance of this relief request is documented in a letter from the US NRC to STP Nuclear Operation Company, "South Texas Project, Units 1 and 2 – Request for relief from ASME Code Requirements for the Second 10-Year Interval Inservice Inspection Program Based on Risk-Informed Alternative Approach," dated September 11, 2000. The change in risk for a particular system was calculated using the following:

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

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:

$$\Delta CDF_{TOTAL} = \sum_j \Delta CDF_j$$

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. The risk impact calculations were also performed excluding the Low risk category welds from the calculation.

Results of these calculations are presented in Tables 9 and 10. Also shown in the tables, for comparison purposes, are the results of the Markov model calculation of the change in risk.

The results of the LaSalle Unit 1 risk impact calculation are shown in Table 9. Even using the simplified risk impact approach and including all of the welds in the RISI scope, none of the systems approached the CDF criterion of $1x10^{-7}$ per system. The largest change in CDF came from the HPCS system, at $6.52x10^{-10}$. The total change in CDF was $2.10x10^{-9}$, well below the criterion of $1x10^{-6}$ for all systems combined. Similarly, the change in LERF values were all well below the criterion of $1x10^{-8}$ per system. Again, the largest change came from the HPCS system,

at 3.60×10^{-10}. The total change in LERF was 9.32×10^{-10}, well below the criterion of 1×10^{-7} for all systems combined.

The results of the LaSalle Unit 2 risk impact calculation are shown in Table 10. Even using the simplified risk impact approach and including all of the welds in the RISI scope, none of the systems came close to the CDF criterion of 1×10^{-7} per system. The largest change in CDF came from the RCS system, at 9.65×10^{-10}. The total change in CDF was 2.62×10^{-9}, well below the criterion of 1×10^{-6} for all systems combined. Similarly, the change in LERF values were all well below the criterion of 1×10^{-8} per system. The largest change came from the feedwater system, at 4.28×10^{-10}. The total change in LERF was 1.05×10^{-9}, well below the criterion of 1×10^{-7} for all systems combined.

These conservative results are regarded as a sensitivity study as they only reflect upper bounds on the expected risk impacts. The results obtained using the Markov model are considered more reasonable and realistic.

3.8.2 Change in Risk Sensitivity for FAC-Only and IGSCC-Only Degradation

Welds having FAC or IGSCC as the only degradation mechanism were removed from the RI-ISI population as discussed in Section 2.3. The FAC-only and IGSCC-only elements were considered addressed by their respective augmented inspection programs. The RI-ISI risk impact calculations took credit for the IGSCC and FAC program inspections in that no change in risk was considered for these welds when a Section XI examination was removed.

To assess the impact of including these welds in the change in risk calculations, a bounding evaluation was performed which conservatively took no credit for the IGSCC and FAC program inspections. Both Markov model calculations and bounding calculations (see Section 3.8.1) were made for the systems that had the FAC-only or IGSCC-only examinations eliminated from the RI-ISI population. The results of these calculations for LaSalle Unit 1 are shown in Table 11. The largest system risk change contribution, taking no credit for the FAC and IGSCC inspection programs, came from the RCS system. The change in risk was 3.44×10^{-9} for the realistic Markov calculation and 6.42×10^{-9} for the bounding calculation. This result is more than an order of magnitude below the system risk change limit of 1×10^{-7} and an insignificant contributor to the total change in risk for all systems.

The results of these calculations for LaSalle Unit 2 are shown in Table 12. The largest system risk change contribution, taking no credit for the FAC and IGSCC inspection programs, came from the RCS system. The change was 2.09×10^{-9} for the realistic Markov calculation and 3.73×10^{-9} for the bounding calculation. These results were still almost two orders of magnitude below the system risk change limit of 1×10^{-7} and an insignificant contributor to the total change in risk across all systems.

The result of this conservative approach to evaluating the impact of including FAC-only and IGSCC-only welds in the delta risk calculations demonstrates that even without taking credit for the IGSCC and FAC augmented inspections programs, the impact on system and overall delta risk is insignificant.

4. IMPLEMENTATION AND MONITORING PROGRAM

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

RI-ISI welds and the balance of inspections will remain on the same ten-year inservice inspection interval so that a consistent code of record required by 10CFR50.55a, "Codes and Standards," will be applied. The code required percentages for the first inspection period have already been satisfied by the completion of code required inspections under the existing ASME Section XI ISI program. The required percentages for the second inspection periods will be satisfied with either RI-ISI or the Section XI required inspections. Inspection of 50-67% of the RI-ISI welds for each Unit will be performed by the end of the second inspection period of the current inspection interval. The remaining RI-ISI welds will be examined before the end of the inspection interval. This approach will result in all RI-ISI welds being examined by either the existing Section XI program or the RI-ISI program before the end of the second inspection interval.

The following table provides the current examination distribution of RI-ISI welds during the second ten-year inservice inspection interval. The total number of welds includes six socket welds in Unit 1 and four socket welds in Unit 2. Since these welds are required to be examined during each refueling outage, they are not included in the percentages calculated in the table. Therefore the percentages are based on 60 welds for both units. The current ASME Section XI Program inspections column provides the cumulative inspections and percentages completed to date. The RI-ISI Program columns provide the minimum cumulative inspections required in future outages.

		Current ASME Section XI Program		RI-ISI Program	
Unit	RI-ISI Welds Total	Period 1	Period 2	Period 2	Period 3
1	66	11 (18%)	N/A	30 (50%)	60 (100%)
2	64	14 (23%)	26 (43%)	30 (50%)	60 (100%)

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

implementing procedures will be retained and modified to address the RI-ISI process, as appropriate.

The RI-ISI program will be maintained as a living program requiring feedback of new relevant information to ensure the appropriate identification of high safety significant piping locations. Changes that could impact the RI-ISI program include major changes to the LaSalle PRA or changes to weld selection. Our Risk Management program requires a review of past applications following a PRA update. This requirement will be applied to the RI-ISI program. As a minimum, risk ranking of piping segments and element selections will be reviewed and adjusted during the mandatory program update required at the end of each 10-year ISI Interval. In addition, changes may occur more frequently as requested by NRC Bulletin or Generic Letter provisions, or by industry and plant-specific service experience feedback.

5. PROPOSED ISI PROGRAM PLAN CHANGE

A comparison between the proposed RI-ISI program and the current 1989 Edition, ASME Section XI Code program requirements for in-scope piping is provided in Table 5 and Table 6 for LaSalle Unit 1 and Unit 2, respectively. The number of exams for Unit 1 is reduced from 186 Section XI program exams to 66 RI-ISI program exams. Unit 2 is reduced from 195 exams to 64 exams. As shown in Tables 7 and 8, 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 $1x10^{-9}$ per year for both risk measures. These risk impacts are acceptably small in relation to the risk significance thresholds of the EPRI Topical Report and those in RG 1.174.

6. REFERENCES

1. EPRI, "Revised Risk-Informed Inservice Inspection Evaluation Procedure," TR-112657, Rev. B-A, December 1999.

2. EGC "Risk Informed Inservice Inspection Evaluation," LaSalle Nuclear Power Plant Units 1 and 2 Final Report February 2001.

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, 1999. *EPRI Licensed Material*

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

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

System Description	Number of Segments
Main Steam (MS, MSA, MSB, MSC, MSD)	45 / 39
Reactor Core Isolation Cooling (RI)	6 / 7
Feedwater (FWA, FWB)	14 / 14
Reactor Recirculation System (RR, RRA, RRB)	28 / 29
Low Pressure Core Spray (LP)	7 / 7
High Pressure Core Spray (HP)	8 / 8
Nuclear Boiler (NB)	8 / 8
Reactor Water Cleanup System (RT)	4 / 5
Residual Heat Removal (RHA, RHB, RHC, RHSDC)	60 / 60
Standby Liquid Control (SC)	1 / 1
Control Rod Drive and Scram Discharge Volume (RD, RDA, RDB)	5 / 5
Jet Pump Instrument Nozzles (JPI)	1 / 1
Reactor Pressure Vessel (RPV)	1 / 1
Hydrogen Recombiner (HG)	4 / 4
Total	192 / 189

NOTES: 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
CRD[1]			X								
ECCS[2]	X		X							X	X
FW	X	X									X
HPCS	X		X								
MS											
RCS[3]			X								X
RWCU											X
RCIC		X	X								

1. Includes scram discharge volume.
2. Includes residual heat removal (RHR), low pressure core spray (LP), and hydrogen recombiner (HG).
3. Includes reactor recirculation (RR), reactor pressure vessel (RPV), and jet pump instrument nozzles (JPI), nuclear boiler (NB), and standby liquid control (SC).

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 RI-ISI Program addresses the cumulative impact of all mechanisms that were identified in each system.

Table 3[5]
Number of Elements (Welds) by Risk Category for Unit 1

System	High Risk[4]			Medium Risk[4]		Low Risk[4]	TOTAL
	Category 1	Category 2	Category 3	Category 4	Category 5	Category 6 or 7	All Categories
CRD[1]		1				83	84
ECCS[2]	1	27	150	60	15	463	716
FW	74	12	10				96
HPCS		6		16		75	97
MS				142		151	293
RCS[3]	23	114		63		13	213
RWCU	25		38	3			66
RCIC		10		18		44	72
TOTAL	123	170	198	302	15	829	1637

1. Includes scram discharge volume.
2. Includes residual heat removal (RHR), low pressure core spray (LP), and hydrogen recombiner (HG).
3. Includes reactor recirculation (RR), reactor pressure vessel (RPV), and jet pump instrument nozzles (JPI), nuclear boiler (NB), and standby liquid control (SC).
4. See Figure 1 for definition of EPRI Risk Categories.
5. This table is developed considering all welds within the initial scope assessed by the RI-ISI analysis. Welds having only IGSCC or only FAC as a degradation mechanism are removed from the population for element selection prior to applying the 25% and 10% sampling percentages.

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[5]
Number of Elements (Welds) by Risk Category for Unit 2

System	High Risk[4]			Medium Risk[4]		Low Risk[4]	TOTAL
	Category 1	Category 2	Category 3	Category 4	Category 5	Category 6 or 7	All Categories
CRD[1]		1				83	84
ECCS		25	157	65	12	454	713
FW	79	6	10				95
HPCS		6		14		73	93
MS				140		158	298
RCS[3]	23	105		67	1	12	208
RWCU	24		39	2		1	66
RCIC		12		17		41	70
TOTAL	126	155	206	305	13	822	1627

1. Includes scram discharge volume.
2. Includes residual heat removal (RHR), low pressure core spray (LP), and hydrogen recombiner (HG).
3. Includes reactor recirculation (RR), reactor pressure vessel (RPV), and jet pump instrument nozzles (JPI), nuclear boiler (NB), and standby liquid control (SC).
4. See Figure 1 for definition of EPRI Risk Categories.
5. This table is developed considering all welds within the initial scope assessed by the RI-ISI analysis. Welds having only IGSCC or only FAC as a degradation mechanism are removed from the population for element selection prior to applying the 25% and 10% sampling percentages.

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 minor differences are due to slight differences in the number of welds in these systems.

Table 5[5,6]
Number of Inspections by Risk Category for Unit 1

	High Risk[4]						Medium Risk[4]				Low Risk[4]		All Risk Categories	
	Category 1		**Category 2**		**Category 3**		**Category 4**		**Category 5**		**Category 6 or 7**			
System	**Sec. XI**	**RI-ISI**	**Sec. XI**	**RI-ISI**	**Sec. XI**	**RI-ISI**	**Sec. XI**	**RI-ISI**	**Sec. XI**	**RI-ISI**	**Sec. XI**	**RI-ISI**	**Sec. XI**	**RI-ISI**
CRD[1]											6		6	
ECCS[2]			3	2		2	9	9	1	2	36		49	15
FW	7	8	12	4	3	2							22	14
HPCS			3	2				2			13		16	4
MS							27	18			18		45	18
RCS[3]							16	9					16	9
RWCU								1						1
RCIC			2	3			13	2			17		32	5
TOTAL	7	8	20	11	3	4	65	41	1	2	90		186	66

1. Includes scram discharge volume.
2. Includes residual heat removal (RHR), low pressure core spray (LP), and hydrogen recombiner (HG).
3. Includes reactor recirculation (RR), reactor pressure vessel (RPV), and jet pump instrument nozzles (JPI), nuclear boiler (NB), and standby liquid control (SC).
4. See Figure 1 for definition of EPRI RI-ISI risk categories.
5. Welds having only IGSCC (147) or only FAC (278) as a degradation mechanism are removed from the population for element selection prior to applying the 25% and 10% sampling percentages.
6. Two welds in the RHR system are pipe class 2, all other RI-ISI selected welds are class 1.

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

Table 6[5,6]
Number of Inspections by Risk Category for Unit 2

System	High Risk[4]						Medium Risk[4]				Low Risk[4]		All Risk Categories	
	Category 1		Category 2		Category 3		Category 4		Category 5		Category 6 or 7			
	Sec. XI	RI-ISI	Sec. XI	RI-ISI	Sec. XI	RI-ISI	Sec. XI	RI-ISI	Sec. XI	RI-ISI	Sec. XI	RI-ISI	Sec. XI	RI-ISI
CRD[1]											6		6	
ECCS[2]			3	2		2	18	9		2	33		54	15
FW	9	8	6	2	4	2							19	12
HPCS			2	2				2			13		15	4
MS							23	18			18		41	18
RCS[3]							22	9			2		24	9
RWCU								1			1		1	1
RCIC			4	3			12	2			19		35	5
TOTAL	9	8	15	9	4	4	75	41		2	92		195	64

1. Includes scram discharge volume.
2. Includes residual heat removal (RHR), low pressure core spray (LP), and hydrogen recombiner (HG).
3. Includes reactor recirculation (RR), reactor pressure vessel (RPV), and jet pump instrument nozzles (JPI), nuclear boiler (NB), and standby liquid control (SC).
4. See Figure 1 for definition of EPRI RI-ISI Risk Categories.
5. Welds having only IGSCC (134) or only FAC (288) as a degradation mechanism are removed from the population for element selection prior to applying the 25% and 10% sampling percentages.
6. Two feedwater system welds, one LP system weld & 2 RHR system welds are pipe class 2, all other RI-ISI selected welds are class 1.

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

Table 7
Impact of RI-ISI and No Inspections on CDF and LERF Due to Pipe Ruptures for LaSalle County Station Unit 1 Systems

System	System CDF Events/Reactor-Year			Δ CDF Events/Reactor-Year			Δ LERF Events/Reactor-Year		
	Section XI	RI-ISI	No Inspection	RI-ISI	No Inspection	Acceptance Criterion	RI-ISI	No Inspection	Acceptance Criterion
RCIC	4.45E-10	6.04E-10	6.70E-10	1.60E-10	2.26E-10	<1.00E-07	1.43E-10	1.73E-10	<1.00E-08
RWCU	6.25E-09	6.24E-09	6.25E-09	-1.44E-11	0.00E+00	<1.00E-07	-1.89E-12	0.00E+00	<1.00E-08
RCS[3]	1.98E-08	1.99E-08	2.09E-08	1.62E-10	1.14E-09	<1.00E-07	2.64E-11	2.45E-10	<1.00E-08
MS	3.82E-09	4.04E-09	4.45E-09	2.16E-10	6.28E-10	<1.00E-07	1.60E-10	4.62E-10	<1.00E-08
HPCS	1.65E-09	2.02E-09	2.02E-09	3.66E-10	3.66E-10	<1.00E-07	2.03E-10	2.03E-10	<1.00E-08
FW	1.26E-08	1.22E-08	1.44E-08	-3.38E-10	1.87E-09	<1.00E-07	-3.68E-10	1.44E-09	<1.00E-08
ECCS[2]	4.94E-09	4.60E-09	5.33E-09	-3.45E-10	3.92E-10	<1.00E-07	-3.11E-10	2.35E-10	<1.00E-08
CRD[1]	6.76E-10	7.05E-10	7.05E-10	2.92E-11	2.92E-11	<1.00E-07	2.62E-12	2.62E-12	<1.00E-08
Total	5.01E-08	5.04E-08	5.48E-08	2.36E-10	4.65E-09	<1.00E-06	-1.47E-10	2.77E-09	<1.00E-07

NOTES:

1. Includes scram discharge volume.
2. Includes residual heat removal (RHR), low pressure core spray (LP), and hydrogen recombiner (HG).
3. Includes reactor recirculation (RR), reactor pressure vessel (RPV), and jet pump instrument nozzles (JPI), nuclear boiler (NB), and standby liquid control (SC).

Table 8
Impact of RI-ISI and No Inspections on CDF and LERF due to Pipe Ruptures for LaSalle County Station Unit 2 Systems

System	System CDF Events/Reactor-Year			Δ CDF Events/Reactor-Year			Δ LERF Events/Reactor-Year		
	Section XI	RI-ISI	No Inspection	RI-ISI	No Inspection	Acceptance Criterion	RI-ISI	No Inspection	Acceptance Criterion
RCIC	2.05E-10	3.48E-10	3.97E-10	1.43E-10	1.92E-10	<1.00E-07	1.42E-10	1.74E-10	<1.00E-08
RWCU	6.24E-09	6.24E-09	6.25E-09	0.00E+00	1.44E-11	<1.00E-07	0.00E+00	1.89E-12	<1.00E-08
RCS[3]	1.68E-08	1.73E-08	1.77E-08	5.41E-10	9.18E-10	<1.00E-07	1.04E-10	1.81E-10	<1.00E-08
MS	3.89E-09	4.08E-09	4.47E-09	1.91E-10	5.80E-10	<1.00E-07	1.79E-10	4.51E-10	<1.00E-08
HPCS	8.69E-10	8.73E-10	8.89E-10	3.66E-12	2.02E-11	<1.00E-07	-1.01E-12	2.86E-12	<1.00E-08
FW	1.43E-08	1.44E-08	1.59E-08	1.11E-10	1.62E-09	<1.00E-07	1.12E-10	9.84E-10	<1.00E-08
ECCS[2]	3.67E-09	3.55E-09	4.20E-09	-1.14.E-10	5.31E-10	<1.00E-07	-2.77E-10	1.52E-10	<1.00E-08
CRD[1]	7.36E-10	7.65E-10	7.65E-10	2.92E-11	2.92E-11	<1.00E-07	2.62E-12	2.62E-12	<1.00E-08
Total	4.66E-08	4.75E-08	5.05E-08	9.04E-10	3.91E-09	<1.00E-06	2.61E-10	1.95E-09	<1.00E-07

NOTES:
1. Includes scram discharge volume.
2. Includes residual heat removal (RHR), low pressure core spray (LP), and hydrogen recombiner (HG).
3. Includes reactor recirculation (RR), reactor pressure vessel (RPV), and jet pump instrument nozzles (JPI), nuclear boiler (NB), and standby liquid control (SC).

Table 9
Comparison of Risk Impact Results for Unit 1

System	CDF			LERF		
	Simplified ΔCDF for All Welds	Simplified ΔCDF Excluding Low Risk Welds	Realistic ΔCDF - Markov Model	Simplified ΔLERF for All Welds	Simplified ΔLERF Excluding Low Risk Welds	Realistic ΔLERF Markov Model
CRD	5.16E-11	0.00E+00	2.92E-11	4.64E-12	0.00E+00	2.62E-12
ECCS	-5.82E-11	-1.25E-10	-3.45E-10	-9.95E-11	-1.11E-10	-3.11E-10
FW	2.38E-10	2.33E-10	-3.38E-10	5.71E-11	5.48E-11	-3.68E-10
HPCS	6.52E-10	6.30E-10	3.66E-10	3.60E-10	3.60E-10	2.03E-10
MS	3.68E-10	0.00E+00	2.16E-10	2.70E-10	0.00E+00	1.60E-10
RCIC	2.85E-10	2.81E-10	1.60E-10	2.32E-10	2.32E-10	1.43E-10
RCS	5.75E-10	0.00E+00	1.62E-10	1.08E-10	0.00E+00	2.64E-11
RWCU	-1.11E-11	0.00E+00	-1.44E-11	-1.46E-12	0.00E+00	-1.89E-12
Total	2.10E-09	1.02E-09	2.36E-10	9.32E-10	5.36E-10	-1.47E-10

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

Table 10
Comparison of Risk Impact Results for Unit 2

System	CDF			LERF		
	Simplified ΔCDF for All Welds	Simplified ΔCDF Excluding Low Risk Welds	Realistic ΔCDF Markov Model	Simplified ΔLERF for All Welds	Simplified ΔLERF Excluding Low Risk Welds	Realistic ΔLERF using Markov Model
CRD	5.16E-11	5.16E-11	2.92E-11	4.64E-12	4.64E-12	2.62E-12
ECCS	2.64E-10	1.82E-10	-1.14E-10	-1.01E-10	-1.21E-10	-2.77E-10
FW	7.41E-10	7.37E-10	1.11E-10	4.28E-10	4.25E-10	1.12E-10
HPCS	1.40E-11	0.00E+00	3.66E-12	-4.65E-14	0.00E+00	-1.01E-12
MS	3.27E-10	0.00E+00	1.91E-10	2.97E-10	0.00E+00	1.79E-10
RCIC	2.43E-10	2.38E-10	1.43E-10	2.35E-10	2.35E-10	1.42E-10
RCS	9.65E-10	0.00E+00	5.41E-10	1.84E-10	0.00E+00	1.04E-10
RWCU	1.44E-11	0.00E+00	0.00E+00	1.89E-12	0.00E+00	0.00E+00
SBLC	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Total	2.62E-09	1.21E-09	9.04E-10	1.05E-09	5.43E-10	2.61E-10

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

Table 11
Unit 1 Impact of FAC and IGSCC Welds in CDF Change in Risk Calculations

System	RISI (see Table 9)		No FAC Inspections		No IGSCC Inspections		No FAC and IGSCC Inspections	
	Markov	Bounding	Markov	Bounding	Markov	Bounding	Markov	Bounding
RWCU	-1.44E-11	-1.11E-11	-5.26E-12	5.15E-12	N/A	N/A	N/A	N/A
ECCS	-3.45E-10	-5.82E-11	-3.14E-10	1.52E-12	-1.48E-10	2.93E-10	-1.14E-10	3.52E-10
FW	-3.38E-10	2.38E-10	7.34E-10	2.14E-09	N/A	N/A	N/A	N/A
RCS[1]	1.62E-10	5.75E-10	N/A	N/A	3.44E-09	6.42E-09	N/A	N/A

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

Notes:
1. For Unit 1, The evaluation considered CRD as a part of the RCS system.

Table 12
Unit 2 Impact of Including FAC and IGSCC Welds in CDF Delta Risk Calculations

System	RISI (see Table 10)		No FAC Inspections		No IGSCC Inspections		No FAC and IGSCC Inspections	
	Markov	Bounding	Markov	Bounding	Markov	Bounding	Markov	Bounding
RWCU	0.00E+00	1.44E-11	8.87E-10	1.58E-09	N/A	N/A	N/A	N/A
ECCS	-1.14E-10	2.64E-10	-8.21E-11	3.24E-10	-3.23E-11	4.10E-10	1.85E-12	4.70E-10
FW	1.11E-10	7.41E-10	1.10E-09	2.49E-09	N/A	N/A	N/A	N/A
CRD	2.92E-11	5.16E-11	N/A	N/A	5.81E-11	1.03E-10	N/A	N/A
RCS	5.41E-10	9.65E-10	N/A	N/A	2.09E-09	3.73E-09	N/A	N/A

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