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

ComEd

October 18, 2000

PSLTR# 00-0144

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

Dresden Nuclear Power Station, Units 2 and 3
Facility Operating License Nos. DPR-19 and DPR-25
NRC Docket Nos. 50-237 and 50-249

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), Dresden Nuclear Power Station (Dresden) is submitting, for U. S. Nuclear Regulatory Commission (NRC) approval, a proposed alternative to 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 Dresden 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-21 and the summary of the RI-ISI Program Plan 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 Dresden's RI-ISI submittal is consistent with the Nuclear Energy Institute and industry template developed for applications of the RI-ISI methodology. Additional supporting documentation is available at Dresden Station for NRC review.

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Dresden plans to incorporate the RI-ISI program during the third period of the third Inservice Inspection Interval for both Units 2 and 3. For Unit 2, the third Inservice Inspection Interval began on March 1, 1992 and the projected end date is January 19, 2003. For Unit 3, the third Inservice Inspection Interval began on March 1, 1992 and the projected end date is October 31, 2002.

We intend to incorporate the risk based approach to the selection and examination of Class 1 and 2 piping welds for the remaining period of this third Inservice Inspection Interval for both units. It is our intent to complete 100 percent of the RI-ISI locations over the current third Inservice Inspection Interval by either the current ASME Section XI ISI program or by the proposed RI-ISI Program.

In order to effectively incorporate this risk informed alternative in the fall 2001 outage for Unit 2, we are requesting approval by March 2001.

Should you have any questions regarding this letter, please contact Mr. Dale F. Ambler at (815) 942-2920, extension 3800.

Respectfully,

Preston Swafford

Preston Swafford
Site Vice President,
Dresden Nuclear Power Station

Attachments:

1) Relief Request CR-21, "Alternate Selection and Examination Criteria for Category B-F, B-J, C-F-1, and C-F-2 Pressure Retaining Piping Welds"

2) Risk-Informed Inservice Inspection Program Plan– Dresden Nuclear Power Station Units 2 and 3

cc: Regional Administrator - NRC Region III
NRC Senior Resident Inspector - Dresden Nuclear Power Station
Office of Nuclear Facility Safety - Illinois Department of Nuclear Safety

Attachment 1

ISI Program Plan

Dresden Nuclear Power Station

Units 2 and 3

Third Interval

Relief Request CR-21

Relief Request CR-21
(Page 1 of 4)

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, B5.20, 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 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 - Dresden Nuclear Power Plants Units 2 and 3, Volumes 1, 2, and 3 (Dated July 2000) 4) American Society of Mechanical Engineers (ASME) Code Case N-578-1, "Risk-Informed Requirements for Class 1, 2, or 3 Piping, Method B"

CODE REQUIREMENT

Table IWB 2500-1, Examination Category B-F, requires a volumetric and/or surface examination on all welds for Items B5.10, B5.20, 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. Dresden Nuclear Power Station (DNPS) Third Interval Inspection Plan Relief Request CR-14, "Exemption from the Section XI Selection Criteria for Category B-J Welds" provides an approved alternative to the Table IWB 2500-1, Examination Category B-J extent and

Relief Request CR-21

CODE REQUIREMENT (Continued)

frequency requirements for Items B9.11, B9.12, B9.21, B9.31, B9.32, and B9.40. As stated in CR-14, DNPS selects Category B-J welds such that 25% of the nonexempt welds are examined during the interval and subsequent intervals. The weld population selected for inspection includes the following:

1. All terminal ends in pipe or branch runs connected to vessels.
2. All terminal ends in each pipe or branch run connected to other components
3. Additional piping welds so that the total number of circumferential butt welds, branch connections, or socket welds selected for examination equals 25% of the total number of nonexempt (i.e., per IWB-1220) circumferential butt welds, branch connections, or socket welds in the reactor coolant piping system. These additional piping welds shall be distributed as follows:
   a. The examinations shall be distributed among the Class 1 systems prorated, to the degree practicable, on the number of nonexempt welds in each system (i.e., if a system contains 30% of the nonexempt welds, then 30% of the nondestructive examinations required by Category B-J should be performed on that system);
   b. Within a system, the examinations shall be distributed among structural discontinuities prorated, to the extent practicable, on the number of nonexempt structural discontinuities in that system;
   c. Within each system, examinations shall be distributed between line sizes prorated to the degree practicable.

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

1. 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). Some welds not exempted by IWC-1220 do not require 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:

Relief Request CR-21

CODE REQUIREMENT (Continued)

a. the examinations shall be distributed among the Class 2 systems prorated, to the degree practicable, on the number of nonexempt austenitic or high alloy 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 utilizing Reference 1 along with two enhancements identified in Reference 4 will provide an acceptable level of quality and safety.

As stated in "Safety Evaluation Report Related to EPRI Risk-Informed Inservice Inspection Evaluation Procedure (EPRI TR-112657, Revision B, July 1999)" (Reference 2):

> "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, DNPS will be utilize the requirements of Subarticle-2430, "Additional Examinations" which is contained in Reference 4. The alternative criteria for additional examinations contained in Code Case N-578-1 (Reference 4) provides a more refined methodology for implementing necessary additional examinations.

Relief Request CR-21
(Page 4 of 4)

BASIS FOR RELIEF (Continued)

To supplement the requirements listed in Table 4-1, "Summary of Degradation-Specific Inspection Requirements and Examination Methods" of EPRI TR-112657, DNPS will utilize the provisions listed in Table 1, Examination Category R-A, "Risk-Informed Piping Examinations" of Reference 4. Table 1 of Code Case N-578-1 provides more refined guidance for examination method and categorization for parts to be examined.

PROPOSED ALTERNATE PROVISIONS

The proposed alternative described in the attached "Risk Informed Inservice Inspection Program Plan, Dresden Units 2 and 3," (Reference 3), along with the two enhancements noted previously provide an acceptable level of quality and safety as required by 10 CFR 50.55a(a)(3)(i).

Our application of the Risk Informed ISI, per EPRI TR-112657, Rev. B-A (Reference 1), requires that 25% of the elements that are categorized as "High" risk (i.e., Risk Category 1, 2, or 3) and 10% of the elements that are categorized as "Medium" risk (i.e., Risk Categories 4 and 5) be selected for inspection. 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 DNPS pressure and leak test program, which remains unaffected by the risk informed ISI program.

APPLICABLE TIME PERIOD

Relief is requested for DNPS for the third inspection period of the third ten-year interval for Unit 2 and for the third inspection period of the third ten-year interval for Unit 3.

Risk-Informed Inservice Inspection Program Plan

Dresden Nuclear Power Station

Units 2 and 3

Table of Contents

1. INTRODUCTION

2. PROPOSED ALTERNATIVE TO CURRENT ISI PROGRAM

   2.1 ASME Section XI

   2.2 Alternative RI-ISI Program

   2.3 Augmented Programs

   2.4 Multiple Damage Mechanisms

3. RISK-INFORMED ISI PROCESS

   3.1 Definition of RI-ISI Program Scope

   3.2 Consequence Analysis

   3.3 Degradation Mechanism Assessment

   3.4 Risk Categorization

   3.5 Inspection Location Selection and NDE Selection

   3.6 Program Relief Requests

   3.7 Risk Impact Assessment

4. IMPLEMENTATION AND MONITORING PROGRAM

5. PROPOSED ISI PROGRAM PLAN CHANGE

6. REFERENCES

1. INTRODUCTION

The objective of this program is to allow the use of a risk-informed inservice inspection (RI-ISI) program for American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," Class 1 and Class 2 piping requirements. This piping is currently inspected as part of the ASME Section XI based ISI (Inservice Inspection) program, 1989 Edition. The risk-informed process used in this submittal is described in Electric Power Research Institute (EPRI) Risk-Informed Inservice Inspection (RI-ISI) Topical Report (TR) (Reference 1). To strengthen the technical basis for this RI-ISI program beyond the minimum requirements implied by the EPRI RI-ISI TR, a number of enhancements were made to the process that are described in the paragraphs below.

Commonwealth Edison (ComEd) Company plans to incorporate the RI-ISI inspection program during the third period of the third inspection interval for Dresden Nuclear Power Station (DNPS), Units 2 and 3. The Third Inservice Inspection Interval started on March 1, 1992, for DNPS Unit 2, and the projected end date is January 19, 2003. This includes all extensions currently being taken. The Third Inservice Inspection Interval started on March 1, 1992, for DNPS Unit 3, and the projected end date is October 31, 2002. This includes all extensions currently being taken.

As a risk-informed application, this submittal meets the intent and principles of Regulatory Guides 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis,” and 1.178, “An Approach For Plant-Specific Risk-informed Decisionmaking Inservice Inspection of Piping” as well as those set forth in the EPRI RI-ISI TR 112657 Revision B-A “Revised Risk-Informed Inservice Inspection Evaluation Procedure.”

PRA Quality

The DNPS probabilistic risk assessment (PRA) model revision used to evaluate the consequences of pipe rupture for the RI-ISI assessment is documented in Calculations DRE99-0030, for Core Damage Frequency (CDF)(Reference 2) and DRE00-0042, for Large Early Release Frequency (LERF) (Reference 3). The base CDF and corresponding LERF from the DNPS PRA model are 2.6×10^{-6} per year and 1.4×10^{-6} per year, respectively. While separate PRA models were prepared to identify any significant differences in PRA results for each reactor unit, the differences that were identified do not impact key sequences cutsets, or system and component importance to any appreciable degree. Hence, these results apply to both reactor units.

A ComEd PRA Maintenance and Update Procedure, NEP-17-04, formalizes the PRA update process. The procedure defines the process for regular and interim updates and for tracking issues identified as potentially affecting the PRA.

The NRC Staff reviewed the DNPS IPE 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 IPE is complete with regard to the information requested by GL 88-20 (and associated guidance, NUREG-1335) and concludes that the licensee's IPE process meets the intent of GL 88-20." The staff concluded in its summary, "The licensee explicitly addressed the staff's concerns in the modified IPE submittal." The staff did note that Common Cause Factors (CCF) were lower than generic. ComEd has since enhanced the DNPS 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 DNPS PRA since the NRC Staff Evaluation Report on the IPE was issued in October 1997. Much of this upgrade was based on the results of the BWROG PRA Certification Peer Review of the DNPS PRA in January 1998. The upgrade of the DNPS PRA was done in parallel with an upgrade of the Quad Cities Nuclear Power Station PRA and has produced PRAs of comparable quality. Quad Cities and DNPS Nuclear Plants are sister plants with similar designs. ComEd had essentially the same personnel working on each of the PRA upgrades. Common enhancements to both plants PRAs included conversion to linked fault trees, addition of special initiators, update of initiating events data, revision of human reliability analysis, update of equipment failure rate, unavailability data and Common Cause Factors, and upgrading Event Tree Analysis. For these reasons, ComEd believes that the results from the BWROG PRA Certification Peer Review of the DNPS PRA scheduled for November 2000 will be essentially the same as the results obtained from the November 1999 BWROG PRA Certification Peer Review of the Quad Cities PRA.

The BWROG 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 Quad Cities PRA was graded 3 in ten of the PRA elements and 4 in the eleventh. As stated above, the results of the forthcoming BWROG PRA Certification Peer Review of the DNPS PRA are expected to be essentially the same.

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 past NRC Staff reviews and the BWROG PRA Certification Peer Reviews, ComEd believes that the level of detail and quality of the DNPS PRA fully supports the DNPS 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 Alternative RI-ISI Program

The alternative RI-ISI program for piping is described in EPRI TR 112657. 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 TR 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 TR, certain augmented inspection programs may be integrated into the RI-ISI program. In accordance with Table 6-2 of the EPRI TR, the issues raised by NRC Bulletin 88-08 are all 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:

- Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping (i.e., Generic Letter 88-01, except for Category A. Both Unit 2 and Unit 3 have completed the 25% population required for the current ten year interval required by Generic Letter 88-01
- NUREG-0313, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping,"
- Service Water Integrity Program (i.e., Generic Letter 89-13)
- Flow Accelerated Corrosion (FAC) (i.e., Generic Letter 89-08)
- High Energy Line Breaks (i.e., 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 RI-ISI element selection population and retained in the appropriate augmented inspection program. If another damage mechanism was identified, the element was retained within the scope of consideration for element selection as part of the RI-ISI program. In the Reactor 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 not to be susceptible to the damage mechanisms addressed in the EPRI RI-ISI methodology. A number of elements were found to be susceptible to one specific damage mechanism (approximately 20% for Unit 2 and approximately 15% for Unit 3), and a relatively small number were identified to be subject to the potential for two or more damage mechanisms (approximately 5.5% for Unit 2 and approximately 5% for Unit 3). Specific examples are welds in the Reactor Feedwater System that are subject to both FAC and thermal fatigue, as well as welds in the Shutdown Cooling and Low Pressure Coolant Injection 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 TR. 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 TR for ASME Code Case N-578-1 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 and as-operated 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 TR. 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 TR. 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 RI-ISI 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 TR 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 TR.

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 6 and described in the Tier 2 documentation (Reference 4).

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.

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)

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 DNPS Unit 2 and Unit 3, respectively.

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 TR, requires that 25% of the elements that are categorized as "High" risk (i.e., Risk Category 1, 2, or 3) and 10% of the elements that are categorized as "Medium" risk (i.e., Risk Categories 4 and 5) be selected for inspection and appropriate non-destructive examination (NDE). Inspection locations are generally selected on a system-by-system basis, so that each system with "High" risk category elements will have approximately 25% of the system's "High" risk elements selected for inspection and similarly 10% of the elements in systems having "Medium" risk category welds will be selected. During the selection process, an attempt is made to ensure that all damage mechanisms and all combinations of damage mechanisms are represented in the elements selected for inspection. An element ranking process was used to incorporate several factors into the selection of specific elements to satisfy the above sampling percentages. These factors include whether the element has been previously selected for ISI exams, whether previous exams had indications of possible damage, presence of radiation fields in the vicinity of the elements, accessibility of the element for inspection, and numerical estimates of the pipe rupture frequencies at these locations. The results of the selection are presented in Tables 5 and 6 for Units 2 and 3, respectively. Section 4 of the EPRI TR and ASME Code Case N-578-1 (Reference 7) were used as guidance in determining the examination methods and requirements for these locations. From the Class 1 butt welded elements that were considered within the scope of the RI-ISI evaluation at Unit 2, a total of 13.3% were selected for volumetric examination as part of the risk informed inspection program. Of the Class 1 socket welded elements that were considered within the scope of the RI-ISI evaluation, a total of 5.5% were selected for a risk informed examination using VT-2 inspections in accordance with ASME Code Case N-578-1. The corresponding percentages for Unit 3 were 12.0% and 6.7%, respectively. 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. This approach, which was approved under Code Case N-524 (approved for use in Regulatory Guide 1.147), is considered to be an acceptable alternative method of scheduling and examining longitudinal welds.

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

<u>Examination Method and Parts Examined</u>

Code Case N-578-1 Table 1, "Examination Category R-A, "Risk-Informed Piping Examinations" will be used in conjunction with of Table 4-1 of EPRI TR-112657 to provide supplemental guidance for the examination method applicable to socket welds. N-578-1 allows a VT-2 examination of socket welds to be performed each refuel outage in lieu of a volumetric or surface examination, regardless of the degradation mechanism. DNPS believes the VT-2 examination method is a more meaningful examination method when the nature of the flaw propogation 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 examination shall be further extended to include additional examinations.

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

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

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

3.6 Program Relief Requests

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 TR will be followed.

DNPS has determined that Relief Request CR-14, "Exemption from the Section XI Selection Criteria for Category B-J," can be withdrawn due to changes that occur from implementing the RI-ISI program. CR-14 will be withdrawn upon approval of this relief request CR-21.

In addition, DNPS's reference to adopting Code Case N-524, "Alternative Examination Requirements for Longitudinal Welds in Class 1 and 2 Piping, Section XI, Division 1," will be removed from the ISI Plan upon approval of relief request (CR-21).

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

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 TR, 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 in the scope of the RI-ISI evaluation. This is another enhancement that was made that goes well beyond the limited quantitative analyses that are needed to implement the methods described in the EPRI TR.

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 TR. 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 TR.

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 methodology (TR-112657) to ensure that the change in risk of implementing the RI-ISI program meets the requirements of Regulatory Guides 1.174 and 1.178. The criteria established require that the cumulative increase in CDF and LERF be less than $1x10^{-7}$ and $1x10^{-8}$ per year per system, respectively. Meeting these limits is consistent with meeting Regulatory Guide 1.174 risk significant thresholds of $1x10^{-6}$ per year and $1x10^{-7}$ per year for changes in CDF and LERF 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 4). These parameters include a set of failure rates and rupture frequencies for piping systems in General Electric BWR plants subject to several degradation mechanisms that were identified for these systems as part of the degradation mechanism assessment. The failure rates and rupture frequencies that were used in this evaluation are those developed in Table A-11, in EPRI TR-111880.

Separate Markov calculations were performed for the change in CDF and the change in LERF. This calculation was performed so that pipe elements whose failure could create a potential bypass concern were factored into the LERF evaluation. Due to the relatively high LERF to CDF ratios for these BWR Mark I 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 DNPS Unit 2 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 TR. A similar set of results is presented in Table 8 and Figures 4 and 5 for Unit 3. As seen in these figures and tables, most of the 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 10% of the EPRI RI-ISI risk significance threshold of $1x10^{-8}$/system-year, and a change in CDF that is less than 2.5% 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 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 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 Regulatory Guide 1.174 and the EPRI RI-ISI TR.

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 choosing 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 ASME Code regardless of its risk classification.

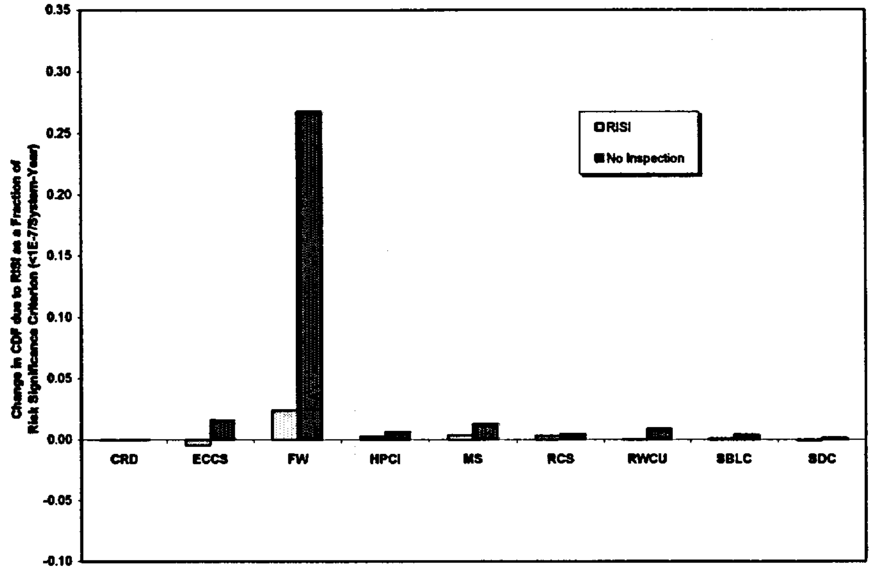

Figure 2
Change in Pipe Rupture CDF for Dresden Unit 2 Systems

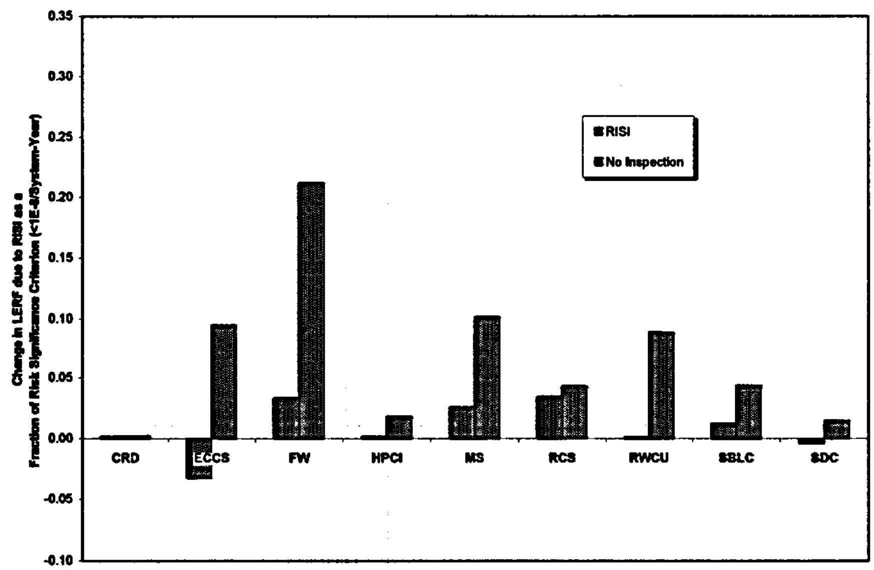

Figure 3
Change in Pipe Rupture LERF for Dresden Unit 2 Systems

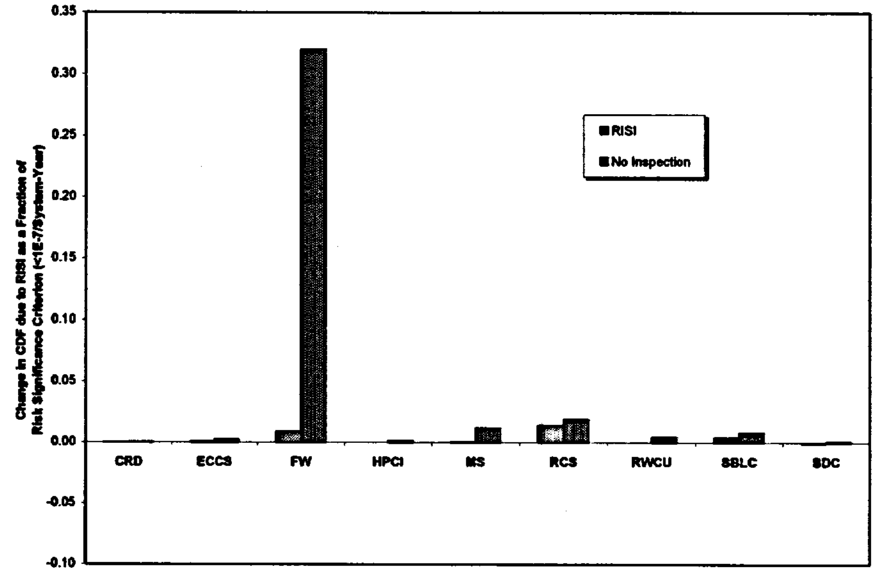

Figure 4
Change in Pipe Rupture CDF for Dresden Unit 3 Systems

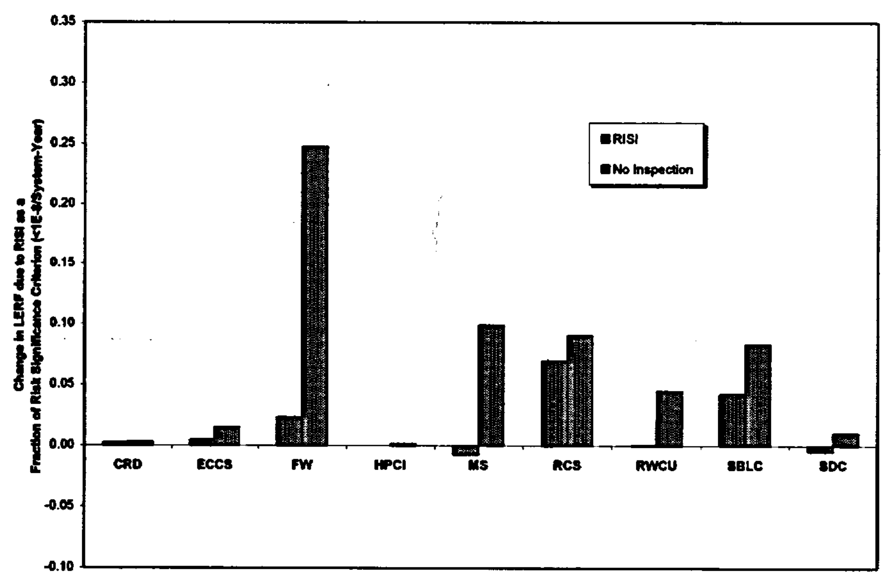

Figure 5
Change in Pipe Rupture LERF for Dresden Unit 3 Systems

4. IMPLEMENTATION AND MONITORING PROGRAM

Upon approval of the RI-ISI program, procedures that comply with the guidelines described in EPRI RI-ISI TR will be prepared to implement and monitor the program. The new program will be integrated into the third period of the third inservice inspection interval for DNPS Units 2 and 3. 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 RI-ISI process, as appropriate.

The RI-ISI 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 DNPS Units 2 and 3 PRA models which could impact both the risk characterization and risk impact assessments, any new trends in service experience with piping systems at DNPS and across the industry, and new information on element accessibility that will be obtained as the risk informed inspections are implemented. As a minimum, risk ranking of piping segments and element selections will be reviewed and adjusted on an ASME ISI interval basis. In addition, changes may occur more frequently as directed by NRC Bulletin or Generic Letter requirements, or by industry and plant-specific service experience feedback.

5. PROPOSED ISI PROGRAM PLAN CHANGE

A comparison between the RI-ISI program and 1989 ASME Section XI Code Edition program requirements for in-scope piping is provided in Table 5 and Table 6 for Unit 2 and Unit 3, respectively. The number of exams at Unit 2 is reduced from 244 Section XI program exams to 95 RI-ISI program exams, a net reduction of 149 exams (61% reduction in number of exams). Unit 3 is reduced from 270 exams to 94 exams, a net reduction of 176 exams (65% reduction in number of 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 $1x10^{-7}$ per year, and $1x10^{-8}$ per year, respectively. These risk impacts are acceptably small in relation to the risk significance thresholds of the EPRI TR and those in Regulatory Guide 1.174.

6. REFERENCES

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

2. ComEd Calculation DRE99-0030, "Dresden PRA Model Update and OPSRE Version 2.0 Development for Unit 2 and Unit 3"

3. ComEd Calculation DRE00-0042, "Dresden LERF (Large Early Release Frequency) Model"

4. ComEd Risk Informed Inservice Inspection Evaluation, Dresden Nuclear Power Plant Units 2 and 3 – Final Report, July 2000

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"

Table 1
System Selection and Segment Definition for Unit 2 / Unit 3

System Description	Number of Segments
Main Steam and MS Drains (MSA, MSB, MSC, MSD, MSDN)	19 / 19
Reactor Head Vent (RHV)	5 / 6
Feedwater (FWA, FWB, FW2)	16 / 16
Reactor Recirculation System (RRAD, RRBD, RRAS, RRBS, RVBD)	35 / 34
Core Spray System (CSAD, CSAS, CSBD, CSBD)	38 / 38
Isolation Condenser (ICSOCR, ISCOSS)	15 / 13
High Pressure Coolant Injection (HPCIPD, HPCIPS, HPCISS, HPCITE)	30 / 27
Low Pressure Coolant Injection System (LPCIAD, LPCIAS, LPCIBD, LPCIBS, LPCISR, LPCITR, LPCIX)	67 / 65
Shutdown Cooling (SDC)	18 / 15
Reactor Water Cleanup System (RWCU)	9 / 8
Reactor Head Spray and Extra Nozzle (RHS, RHSP)	3 / 5
Standby Liquid Control (SBLC)	5 / 4
Control Rod Drive and Scram Discharge Volume (CRD, CRDSD)	9 / 9
ECCS Common Suction – Torus Ring Header (ECCS)	2 / 2
Jet Pump Instrument Nozzles (JPIA, JPIB)	2 / 2
RPV Level Instrument Nozzles (LVLA, LVLB, UVLA, UVLB)	4 / 4
Total	277 / 267

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.

Suppression Pool Cooling (SPC) is modeled as part of LPCITR.

Table 2
Failure Potential Assessment Summary for Unit 2 and Unit 3

	Thermal Fatigue		Stress Corrosion Cracking				Localized Corrosion			Flow Sensitive	
System	TASCS	TT	IGSCC	TGSCC	ECSCC	PWSCC	MIC	PIT	CC	E-C	FAC
CRD[1]											
ECCS[2]	X	X	X								
FW	X	X									X
HPCI		X									
MS[3]	X	X	X								X
RHV			X								
RR[4]	X		X								
RWCU			X								X
SBLC	X										
SDC	X	X	X								

(1) Includes Scram Discharge Volume (CRDSD).
(2) Includes CS, LPCI, RHS, and RHSP.
(3) Includes Isolation Condenser (ISCOCR, ISCOSS).
(4) Includes Jet Pump and RPV Level Instrument nozzles (JPIA, JPIB, LVLA, LVLB, UVLA, UVLB).

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

System	High Risk[5]			Medium Risk[5]		Low Risk[5]	TOTAL
	Category 1	Category 2	Category 3	Category 4	Category 5	Category 6 or 7	All Categories
CRD[1]				2		64	66
ECCS[2]		53		150	2	434	639
FW	55		1				56
HPCI				76	25	78	179
MS[3]		16	21	121	37	49	244
RHV					1	73	74
RR[4]		110		18	13	85	226
RWCU	5	10	6	13		17	51
SBLC				23	13	13	49
SDC		47				24	71
TOTAL	60	236	28	403	91	837	1655

(1) Includes Scram Discharge Volume (CRDSD).
(2) Includes CS, LPCI, RHS, and RHSP.
(3) Includes Isolation Condenser (ISCOCR, ISCOSS).
(4) Includes Jet Pump and RPV Level Instrument nozzles (JPIA, JPIB, LVLA, LVLB, UVLA, UVLB).
(5) See Figure 1 for definition of EPRI Risk Categories

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).

Table 4
Number of Elements (Welds) by Risk Category for Unit 3

System	High Risk[5]			Medium Risk[5]		Low Risk[5]	TOTAL
	Category 1	Category 2	Category 3	Category 4	Category 5	Category 6 or 7	All Categories
CRD[1]				2		51	53
ECCS[2]		25		149	2	484	660
FW	56		1				57
HPCI		2		69	22	87	180
MS[3]		10	37	113	39	55	254
RHV				1	3	80	84
RR[4]				84	6	59	149
RWCU	4		6	13		19	42
SBLC				38	17	20	75
SDC		44				19	63
TOTAL	60	81	44	469	89	874	1617

(1) Includes Scram Discharge Volume (CRDSD).
(2) Includes CS, LPCI, RHS, and RHSP.
(3) Includes Isolation Condenser (ISCOCR, ISCOSS).
(4) Includes Jet Pump and RPV Level Instrument nozzles (JPIA, JPIB, LVLA, LVLB, UVLA, UVLB). See Figure 1 for definition of EPRI Risk Categories
(5) See Figure 1 for definition of EPRI Risk Categories

NOTE: This table shows the results of the Risk Categorization for Unit 3. 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
Number of Inspections by Risk Category for Unit 2

System	High Risk						Medium Risk				Low Risk		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]							1	1			4	0	5	1
ECCS[2]			19	14			13	16	0	0	39	0	71	30
FW	8	7			0	0							8	7
HPCI							11	8	2	3	6	0	19	11
MS[3]			3	3	5	6	32	13	1	1	19	0	60	23
RHV									0	0	18	0	18	0
RR[4]			0	0			1	2	6	2	19	0	26	4
RWCU	0	0	0	0	0	0	4	2			3	0	7	2
SBLC							4	3	5	2	4	0	13	5
SDC			12	12							5	0	17	12
TOTAL	8	7	34	29	5	6	66	45	14	8	117	0	244	95

(1) Includes Scram Discharge Volume (CRDSD).
(2) Includes CS, LPCI, RHS, and RHSP.
(3) Includes Isolation Condenser (ISCOCR, ISCOSS).
(4) Includes Jet Pump and RPV Level Instrument nozzles (JPIA, JPIB, LVLA, LVLB, UVLA, UVLB).
(5) See Figure 1 for definition of EPRI RI-ISI risk categories

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
Number of Inspections by Risk Category for Unit 3

System	High Risk[5]						Medium Risk[5]				Low Risk[5]		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]							2	1			4	0	6	1
ECCS[2]			11	7			18	15	0	0	41	0	70	22
FW	9	7			0	0							9	7
HPCI			0	0			6	7	2	3	10	0	18	10
MS[3]			1	1	8	10	28	12	2	1	15	0	54	24
RHV							1	1	0	0	19	0	20	1
RR[4]							34	9	3	1	13	0	50	10
RWCU	0	0			0	0	2	2			6	0	8	2
SBLC							8	4	5	2	6	0	19	6
SDC			10	11							6	0	16	11
TOTAL	9	7	22	19	8	10	99	51	12	7	120	0	270	94

(1) Includes Scram Discharge Volume (CRDSD).
(2) Includes CS, LPCI, RHS, and RHSP.
(3) Includes Isolation Condenser (ISCOCR, ISCOSS).
(4) Includes Jet Pump and RPV Level Instrument nozzles (JPIA, JPIB, LVLA, LVLB, UVLA, UVLB).
(5) See Figure 1 for definition of EPRI RI-ISI Risk Categories

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 Dresden 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
CRD[1]	6.23E-10	6.42E-10	6.60E-10	1.94E-11	3.74E-11	<1.00E-07	1.94E-11	2.65E-11	<1.00E-08
ECCS[2]	2.59E-08	2.55E-08	2.76E-08	-4.21E-10	1.62E-09	<1.00E-07	-3.22E-10	9.50E-10	<1.00E-08
FW	1.39E-07	1.42E-07	1.66E-07	2.41E-09	2.68E-08	<1.00E-07	3.33E-10	2.12E-09	<1.00E-08
HPCI	4.37E-09	4.66E-09	5.02E-09	2.90E-10	6.51E-10	<1.00E-07	2.08E-11	1.81E-10	<1.00E-08
MS[3]	8.74E-09	9.13E-09	1.01E-08	3.84E-10	1.33E-09	<1.00E-07	2.59E-10	1.02E-09	<1.00E-08
RCS[4]	1.28E-08	1.32E-08	1.33E-08	3.34E-10	4.85E-10	<1.00E-07	3.43E-10	4.33E-10	<1.00E-08
RWCU	8.44E-09	8.47E-09	9.35E-09	3.63E-11	9.11E-10	<1.00E-07	1.44E-11	8.89E-10	<1.00E-08
SBLC	3.60E-09	3.72E-09	4.03E-09	1.25E-10	4.36E-10	<1.00E-07	1.25E-10	4.35E-10	<1.00E-08
SDC	9.06E-10	8.69E-10	1.06E-09	-3.73E-11	1.58E-10	<1.00E-07	-3.55E-11	1.50E-10	<1.00E-08
Total	2.05E-07	2.08E-07	2.37E-07	3.14E-09	3.24E-08	<1.00E-06	7.57E-10	6.20E-09	<1.00E-07

(1) Includes Scram Discharge Volume (CRDSD).
(2) Includes CS, LPCI, RHS, and RHSP.
(3) Includes Isolation Condenser (ISCOCR, ISCOSS).
(4) Includes Reactor Recirculation (RR), Reactor Head Vent (RHV), Jet Pump Instrument (JPIA, JPIB), and RPV Level Instrument nozzles (LVLA, LVLB, UVLA, and UVLB).

Table 8
Impact of RI-ISI and No Inspections on CDF and LERF due to Pipe Ruptures for Dresden Unit 3 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
CRD[1]	4.47E-10	4.84E-10	5.02E-10	3.74E-11	5.53E-11	<1.00E-07	2.65E-11	3.35E-11	<1.00E-08
ECCS[2]	1.24E-09	1.34E-09	1.53E-09	9.76E-11	2.87E-10	<1.00E-07	4.90E-11	1.51E-10	<1.00E-08
FW	1.71E-07	1.72E-07	2.03E-07	9.02E-10	3.20E-08	<1.00E-07	2.32E-10	2.47E-09	<1.00E-08
HPCI	2.12E-09	2.12E-09	2.29E-09	1.05E-12	1.63E-10	<1.00E-07	5.60E-13	1.60E-11	<1.00E-08
MS[3]	6.21E-09	6.27E-09	7.42E-09	5.98E-11	1.20E-09	<1.00E-07	-7.19E-11	9.88E-10	<1.00E-08
RCS[4]	6.95E-09	8.36E-09	8.85E-09	1.41E-09	1.90E-09	<1.00E-07	6.95E-10	9.02E-10	<1.00E-08
RWCU	5.51E-09	5.51E-09	5.97E-09	3.20E-12	4.59E-10	<1.00E-07	3.15E-12	4.48E-10	<1.00E-08
SBLC	5.87E-09	6.30E-09	6.70E-09	4.24E-10	8.31E-10	<1.00E-07	4.24E-10	8.31E-10	<1.00E-08
SDC	9.17E-10	8.73E-10	1.03E-09	-4.37E-11	1.09E-10	<1.00E-07	-4.26E-11	1.04E-10	<1.00E-08
Total	2.00E-07	2.03E-07	2.37E-07	2.89E-09	3.70E-08	<1.00E-06	1.32E-09	5.95E-09	<1.00E-07

(1) Includes Scram Discharge Volume (CRDSD).
(2) Includes CS, LPCI, RHS, and RHSP.
(3) Includes Isolation Condenser (ISCOCR, ISCOSS).
(4) Includes Reactor Recirculation (RR), Reactor Head Vent (RHV), Jet Pump Instrumentation (JPIA, JPIB), and RPV Level Instrument nozzles (LVLA, LVLB, UVLA, and UVLB)