FPL

JUL 08 2002

L-2002-022
10 CFR 50.55a

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Re: Turkey Point Unit 4
Docket No. 50-251
Risk-Informed Inservice Inspection Program

In accordance with 10 CFR 50.55a(a)(3)(i) Florida Power and Light Company (FPL) requests to revise the Turkey Point Unit 4 Inservice Inspection (ISI) Piping Program for Class 1 piping only, through the use of the attached Risk-Informed Inservice Inspection Program (RI-ISI) as an alternative to the current requirements of the ASME Boiler and Pressure Vessel Code Section XI, 1989 Edition, as required by 10CFR50.55a.

The proposed revisions to the current ISI Program for Class 1 piping only are based on the risk-informed process described in Westinghouse Owners Group (WOG) WCAP-14572, Revision 1-NP-A, "Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report," and WCAP-14572, Revision 1-NP-A, Supplement 1, "Westinghouse Structural Reliability and Risk Assessment (SRRA) Model for Piping Risk-Informed Inservice Inspection."

The attached Risk-Informed Inservice Inspection Program supports the conclusion that the proposed alternative provides an acceptable level of quality and safety as required by 10 CFR 50.55a(a)(3)(i). This program also meets the intent and principles of Regulatory Guides 1.174 and 1.178. This program submittal has been reviewed by the Plant Nuclear Safety Committee.

FPL requests NRC approval of this WOG Risk-Informed Inservice Inspection Pilot Program within 6 months to support timely implementation of the Program. Please note also that FPL considers implementation of this RI-ISI program to be a Cost Beneficial Licensing Action (CBLA).

Should there be any questions concerning this submittal, please contact us.

Very truly yours,

Terry Jones for

J. P. McElwain
Vice President
Turkey Point Plant

CLM

Attachment

NRC Regulatory Issue Summary 2001-05 waived the requirements that multiple copies of documents be submitted to the NRC.

A047

ATTACHMENT TO L-2002-022

Florida Power and Light Company
Turkey Point Unit 4
Risk-Informed Inservice Inspection Piping Program Submittal Using the Westinghouse Owners Group (WOG) Methodology (WCAP-14572, Revision 1-NP-A, February 1999)

Revision 0

December 2001

RISK-INFORMED INSERVICE INSPECTION PROGRAM PLAN

Table of Contents

1. INTRODUCTION/RELATION TO NRC REGULATORY GUIDE RG-1.174

1.1 Introduction

Inservice Inspections (ISI) are currently performed on piping to the requirements of the ASME Boiler and Pressure Vessel Code Section XI, 1989 Edition as required by 10CFR50.55a. The unit is currently in the third inspection interval as defined by the Code for Program B.

The objective of this submittal is to request a change to the ISI program plan for piping through the use of a risk-informed ISI (RI-ISI) program for Turkey Point Unit 4. The risk-informed process used in this submittal is described in Westinghouse Owners Group (WOG) WCAP-14572, Revision 1-NP-A, "Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report," and WCAP-14572, Revision 1-NP-A, Supplement 1, "Westinghouse Structural Reliability and Risk Assessment (SRRA) Model for Piping Risk-Informed Inservice Inspection," (referred to as "WCAP-14572, A-version" for the remainder of this document). "

The Turkey Point Unit 4 RI-ISI program builds off the work done by the WOG for Turkey Point Unit 3. Plant drawings and input data were compared between the two units to identify the appropriate changes to develop the Unit 4 program.

As a risk-informed application, this submittal meets the intent and principles of Regulatory Guide 1.174. Further information is provided in Section 3.10 relative to defense-in-depth.

1.2 Probabilistic Risk Assessment (PRA) Quality

The Turkey Point (PTN) Units 3&4 Level 1 and Level 2 PRA model, PTN Probabilistic Safety Assessment (PSA) Model Update (2000), was used to evaluate the consequences of pipe ruptures during operation in Modes 1 and 2. The base core damage frequency (CDF) and base large, early release frequency (LERF) from this version of the PRA model are 9.84E-06 per year and 3.85E-08 per year, respectively. The PRA model has been updated since the Turkey Point Unit 3 RI-ISI submittal.

The Turkey Point PRA model is an updated version of the original Turkey Point Individual Plant Examination (IPE) submittal. Prior to the IPE being submitted to the U.S. Nuclear Regulatory Commission (NRC), a peer review was conducted by an outside contractor. All review findings were addressed prior to the IPE submittal to NRC on June 25, 1991. Following the submittal, the NRC chose to apply a "Step 1" and a "Step 2" review of the IPE. The "Step 2" portion consisted of a more detailed review, including an on-site visit by an NRC review team. After resolving the findings of this review, a revised IPE was submitted to NRC in March 1992. The NRC safety evaluation report (SER) for the IPE was issued thereafter in October 1992. The SER and the associated technical report were very positive in their assessment of the Turkey Point IPE. The few comments on the submittal were minor and were addressed by Florida Power and Light (FPL) and closed out.

Since the IPE, the FPL Reliability and Risk Assessment Group (RRAG) has maintained the Turkey Point PRA model consistent with the plant configuration as it has evolved. The PRA computer models are updated on an as-needed basis for various reasons, such as plant changes and modifications, procedure changes, accrual of new plant data, modeling improvements, advances in PRA technology, and issuance of new industry PSA standards.

These changes are implemented and documented in a timely manner to ensure that risk analyses performed in support of plant operation reflect the current plant configuration, operating philosophy, transient history, and system and component performance.

The PRA maintenance and update process is governed by the RRAG PRA procedures. Updates to the Turkey Point PRA model are documented and reviewed via engineering calculations and evaluations in accordance with the FPL Engineering Department's Quality Instructions and RRAG procedures. As further verification for this application, the RI-ISI evaluation included a determination that the PRA model and supporting documentation accurately reflect the current Turkey Point plant configuration.

2. PROPOSED ALTERNATIVE TO ISI PROGRAM

2.1 ASME Section XI

ASME Section XI Categories B-F and B-J currently contain the requirements for examining (via non-destructive examination (NDE)) Class 1 piping components. This risk-informed program is limited to ASME Class 1 piping. The alternative RI-ISI program for piping is described in WCAP-14572, A-Version. The Class 1 RI-ISI program will substitute the current examination program on piping in accordance with 10 CFR 50.55a(a)(3)(i) by alternatively providing an acceptable level of quality and safety. Other non-related portions of the ASME Section XI Code will be unaffected. WCAP-14572, A-version, provides the requirements defining the relationship between the risk-informed examination program and the remaining unaffected portions of ASME Section XI.

2.2 Augmented Programs

There are no augmented programs in the Class 1 piping. Therefore, the augmented inspection programs remain unchanged.

3. RISK-INFORMED ISI PROCESSES

The processes used to develop the RI-ISI program are consistent with the methodology described in WCAP-14572, A-Version.

The process that is being applied, involves the following steps:

- Scope Definition
- Segment Definition
- Consequence Evaluation
- Failure Assessment
- Risk Evaluation
- Expert Panel Categorization
- Element/NDE Selection
- Implement Program
- Feedback Loop

Deviations

There are no significant deviations from the process described in WCAP-14572, A-Version.

As part of the risk evaluation described in Section 3.5, the uncertainty analysis as described on WCAP page 125 was performed and is now included as part of the base process.

The change in risk methodology described in Section 3.10 deviated from the methodology for segments located inside containment and that interface with the RCS such that radiation monitors and sump level will detect a leak and thus leak detection is credited for these segments. For these segments, the failure probability "with ISI" for those being inspected by NDE and without ISI for those not being inspected is used along with credit for leak detection.

3.1 Scope of Program

The scope of this program is limited to the Class 1 piping, including piping exempt from current requirements. The Class 1 piping systems included in the RI-ISI program are provided in Table 3.1-1, System Selection and Segment Definition for Class 1 Piping. For Turkey Point Unit 4, because of the vintage of the plant, the Class 1 piping boundaries are from the Reactor Coolant System up to, in most cases, the second isolation valve. This includes piping through the excess letdown and regenerative heat exchangers in the Chemical and Volume Control System.

3.2 Segment Definitions

Once the systems to be included in the program are determined, the piping for these systems is divided into segments.

The number of pipe segments defined for the Class 1 piping (3 systems) is summarized in Table 3.1-1. The as-operated piping and instrumentation diagrams were used to define the segments.

3.3 Consequence Evaluation

The consequences of pressure boundary failures are measured in terms of core damage and large early release. The impact on these measures due to both direct and indirect effects was considered. Table 3.3-1, Summary of Postulated Consequences by System, summarizes the postulated consequences for each system, both the direct and indirect effects.

A review of the design bases of Turkey Point (primarily, the Turkey Point Units 3 and 4 Updated Final Safety Analysis Report and supporting documents) was performed to determine the potential impact of the indirect effects of pipe leak or rupture inside containment. As a result of the review, FPL concluded that the containment structure and the safety related components inside containment are adequately protected from pipe failures such that the effects of a failure are limited to direct effects.

3.4 Failure Assessment

Failure estimates were generated utilizing industry failure history, plant specific failure history and other relevant information. An engineering team was established that had access to expertise from ISI, NDE, materials, stress analysis and system engineering. The team was trained in the failure probability assessment methodology and the Westinghouse Structural Reliability and Risk Assessment (SRRA) code, including identification of the capabilities and limitations as described in WCAP-14572, Revision 1-NP-A, Supplement 1. The SRRA code was used to calculate failure probabilities for the failure modes, materials, degradation mechanisms, input variables and uncertainties it was programmed to consider as discussed in the WCAP Supplement 1. All the piping configurations included in the RI-ISI program could be adequately modeled using the SRRA code.

The engineering team assessed industry and plant experience, plant layout, materials, operating conditions and identifies the potential failure mechanisms and causes. Information was gathered from various sources by the Engineering team to provide input for the SRRA model.

Sensitivity studies were performed to aid in determining representative input values when sufficient information was not available. Snubber failure history was also reviewed to identify any potential effects that could increase piping failure probability.

Table 3.4-1, Failure Probability Estimates (without ISI), summarizes the failure probability estimates for the dominant potential failure mechanism(s)/ combination(s) by system. Table 3.4-1 also describes why the degradation mechanisms could occur at various locations within the system. Full break cases are shown only when pipe whip is of concern.

No augmented inspections are performed for the Class 1 piping. The failure probabilities used in the risk-informed process are documented and maintained in the plant records.

3.5 Risk Evaluation

Each piping segment within the scope of the program was evaluated to determine its core damage frequency (CDF) and large, early release frequency (LERF) due to the postulated piping failure. Calculations were also performed with and without operator action.

Once this evaluation was completed, the total pressure boundary core damage frequency and large early release frequency were calculated by summing across the segments for each system.

The uncertainty analysis as described on WCAP-14572, A-Version page 125 was performed and is now included as part of the base process. The results of these calculations are presented in Table 3.5-1, Number of Segments and Piping Risk Contribution by System (without ISI). The core damage frequency due to piping failure without operator action is 1.46E-05/year, and with operator action is 1.40E-05/year. The large early release frequency due to piping failure without operator action is 2.57E-08/year, and with operator action is 2.46E-08/year.

To assess safety significance, the risk reduction worth (RRW) and risk achievement worth (RAW) were calculated for each piping segment.

3.6 Expert Panel Categorization

The final safety determination, i.e. high and low safety significance, of each piping segment was made by the expert panel using both probabilistic and deterministic insights. The expert panel was composed of personnel who have expertise in the following fields: probabilistic safety assessment, Inservice examination, nondestructive examination, stress and material considerations, plant operations, plant and industry maintenance, repair, and failure history, system design and operation, and SRRA methods, including uncertainty. Members associated with the Maintenance Rule were used to ensure consistency with the other PRA applications. Alternates were used if their expertise and training were sufficient.

The expert panel had the following positions represented by either the permanent or alternate member at all times during an expert panel meeting.

- Probabilistic Risk Assessment (PRA engineer)
- Operations (Senior Reactor Operator or Shift Technical Advisor)
- Inservice Inspection (ISI Engineer)
- Plant & Industry Maintenance , Repair, and Failure History (System Engineer)

A minimum of 4 members or alternates filling the above positions constituted a quorum. This core team of panel members was supplemented by other experts, including a metallurgist and piping stress engineer, as required for the piping system under evaluation.

The expert panel chairperson was appointed by the Turkey Point Engineering Manager. The chairperson conducted and ruled on the proceedings of the meeting.

Members received training and indoctrination in the RI-ISI selection process. They were indoctrinated in the application of risk analysis techniques for ISI. These techniques included risk importance measures, threshold values, failure probability models, failure mode assessments, PRA modeling limitations and the use of expert judgment. Training documentation is maintained with the expert panel's records.

Worksheets were provided to the panel on each system for each piping segment, containing information pertinent to the panel's selection process. This information, in conjunction with each panel member's own expertise and other documents as appropriate, were used to determine the safety significance of each piping segment.

A consensus process was used by the expert panel. Consensus is defined as unanimous during first consideration and 2/3 (rounding conservatively) of members or alternates present in the second or subsequent considerations. The chairperson allowed appropriate time duration between considerations for deliberation.

Meeting minute records were generated. The minutes included the names of members in attendance and whether a quorum was present. The minutes contained relevant discussion summaries and the results of membership voting.

3.7 Identification of High Safety Significant Segments

The number of high safety significant segments for each system, as determined by the expert panel, is shown in Table 3.7-1, Summary of Risk Evaluation and Expert Panel Categorization Results, along with a summary of the risk evaluation identification of high safety significant segments.

3.8 Structural Element and NDE Selection

The structural elements in the high safety significant piping segments were selected for inspection and appropriate NDE methods were defined.

The initial program being submitted addresses the high safety significant (HSS) piping components placed in Regions 1 and 2 of Figure 3.7-1 in WCAP-14572, A-Version. Segments considered as "high failure importance" (Region 1) were defined as all segments identified as being affected by an active failure mechanism or analyzed to be highly susceptible to a failure mechanism (probability of large leak at 40 years exceeds 1E-04). Region 3 piping components, which are low safety significant, are to be considered for an Owner Defined Program and are not considered part of the program requiring approval. Region 1, 2, 3 and 4 piping components will continue to receive Code required pressure testing, as part of the current ASME Section XI program. For the 228 piping segments that were evaluated in the RI-ISI program, Region 1 contains 5 segments, Region 2 contains 13 segments, Region 3 contains 50 segments, and Region 4 contains 160 segments (includes 22 segments categorized as "not used").

The number of locations to be inspected in a HSS segment was determined using a Westinghouse statistical (Perdue) model as described in section 3.7 of WCAP-14572, A-Version. One HSS piping segment in Region 1 and 12 of the HSS piping segments in Region 2 were evaluated using the Perdue model. The 5 segments that were not evaluated using the Perdue model contained only socket welds.

The 5 segments categorized as HSS by the plant expert panel that contain only socket welds consist of piping with a nominal diameter of 2 inches or less. The socket welds in these segments can not be individually examined by any currently available NDE methods that are appropriate for the degradation mechanism of intent. Therefore, for these segments, a VT-2 visual exam will be performed during the system pressure test each refueling outage.

Table 4.1-1 in WCAP-14752, A-Version, was used as guidance in determining the examination requirements for the HSS piping segments. VT-2 visual examinations are scheduled in accordance with the station's pressure test program that remains unaffected by RI-ISI program.

<u>Additional Examinations</u>

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

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

3.9 Program Relief Requests

Alternate methods are specified to ensure structural integrity in cases where examination methods cannot be applied due to limitations such as inaccessibility or radiation exposure hazard.

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

At this time, all the risk-informed examination locations that have been selected provide >90% coverage. In instances where a location may be found at the time of the examination that it does not meet >90% coverage, the process outlined in Section 4.1 of WCAP-14572, A-Version will be followed.

3.10 Change in Risk

The RI- ISI program has been done in accordance with Regulatory Guide 1.174, and the risk from implementation of this program is expected to decrease slightly when compared to that estimated from current requirements.

The change in risk calculations were performed according to all the guidelines provided on page 213 of the WCAP. All four criteria for accepting the results discussed on page 214 and 215 in the WCAP were met or adjustments were made (systems identified in Table 5-1 via a footnote). The change in risk methodology deviated from the methodology for segments located inside containment and that interface with the RCS such that radiation monitors and sump level will detect a leak. For these segments, the failure probability "with ISI" for those being inspected by NDE, and without ISI for those not being inspected, is used along with credit for leak detection.

A comparison between the proposed RI-ISI program and the current ASME Section XI ISI program was made to evaluate the change in risk. The approach evaluated the change in risk with the inclusion of the probability of detection as determined by the SRRA model. This evaluation resulted in the identification of seven (7) additional piping segments for which examinations are now required to meet the four criteria identified in WCAP-14572, pages 214 and 215.

The results from the risk comparison are shown in Table 3.10-1, Comparison of CDF/LERF for Current Section XI and Risk Informed ISI Programs and the Systems which Contributed Significantly to the Change. As can be seen from the table, the RI-ISI program reduces the risk associated with piping CDF/LERF slightly more than the current Section XI program while reducing the number of examinations. Table 3.10-1 also includes the systems that are the main contributors to the risk reduction in moving from the current program to the RI-ISI program. The primary basis for this risk reduction is that examinations are now being placed on piping segments that are High Safety Significant and which are not inspected by NDE in the current ASME Section XI ISI program.

Defense-In-Depth

The reactor coolant piping will continue to receive a system pressure test and visual VT-2 examination as currently required by the Code. Surface and volumetric examinations are proposed on the main reactor coolant piping and main safety injection lines (downstream of first check valve) as part of the RI-ISI program (segments categorized as HSS). These locations, which include reactor vessel, steam generator, and pressurizer dissimilar metal welds determined by the RI-ISI program for Turkey Point Unit 4, assure that "defense in depth" is maintained.

New Information

None

4. IMPLEMENTATION AND MONITORING PROGRAM

Upon approval of the RI-ISI program, procedures that comply with the guidelines described in WCAP-14572, A-Version, will be prepared to implement and monitor the program. The new program will be integrated into the existing ASME Section XI interval. No changes to the Updated Final Safety Analysis Report are necessary for program implementation.

The applicable aspects of the Code not affected by this change would 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 would be retained and would be modified to address the RI-ISI process, as appropriate. Additionally the procedures will be modified to include the high safety significant locations in the program requirements regardless of their current ASME class.

The proposed monitoring and corrective action program will contain the following elements:

A. Identify
B. Characterize
C. (1) Evaluate, determine the cause and extent of the condition identified
   (2) Evaluate, develop a corrective action plan or plans
D. Decide
E. Implement
F. Monitor
G. Trend

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. As a minimum risk ranking of piping segments will be reviewed and adjusted on an ASME period basis. Significant changes may require more frequent adjustment as directed by NRC Bulletin or Generic Letter requirements, or by plant specific feedback.

5. PROPOSED ISI PROGRAM PLAN CHANGE

A comparison between the RI-ISI program and the current ASME Section XI program requirements for piping is given in Table 5-1, Comparison of CDF/LERF for Current Section XI and Risk Informed ISI Programs and the Systems which Contributed Significantly to the Change. An identification of piping segments that are part of plant augmented programs is also included in Table 5-1.

The plant will be performing examinations on elements not currently required to be examined by ASME Section XI. An example of these additional examinations is that several elements (segments) currently classified as exempt from examination as nominal pipe size (NPS) 1" and smaller shall be included into the program plan, for Class 1 only.

The initial program will be started in the inspection period current at the time of program approval. Turkey Point Unit 4 is currently in its third ten-year interval, which ends in April 2004.

6. SUMMARY OF RESULTS AND CONCLUSIONS

A partial scope (Class 1 piping) risk-informed ISI application has been completed for Turkey Point Unit 4. Upon review of the proposed risk-informed ISI examination program given in Table 5-1, an appropriate number of examinations are proposed for the high safety significant segments across the Class 1 portions of the plant piping systems. Resources to perform examinations currently required by ASME Section XI in the Class 1 portions of the plant piping systems, though reduced, are distributed to address the greatest amount of risk within the scope. Thus, the change in risk principle of Regulatory Guide 1.174 is maintained. Additionally, the examinations performed will address specific damage mechanisms postulated for the selected locations through appropriate examination selection and increase volume of examination.

The PRA model has been updated since the Turkey Point Unit 3 RI-ISI submittal.

A comparison between Turkey Point Unit 3 and Turkey Point Unit 4 was made. From a piping layout perspective, the most significant difference identified for Unit 4 was the location of the Residual Heat Removal system hot leg suction line (Loop A versus Loop C) and the corresponding relation to the high head safety injection recirculation lines (Loops A and B). This change affects the combinations of plant components potentially disabled by the failure of several piping segments. There are also several differences in the location of small bore piping branch connections in the SI and CH systems. Finally, the alternate low injection path (located on Loop C for Unit 3) is located on Loop A for Unit 4.

In addition, plant-specific stress inputs were used for each unit's failure probability evaluation. This resulted in different failure probabilities across many segments.

As a risk-informed application, this submittal meets the intent and principles of Regulatory Guide 1.174.

7. REFERENCES/DOCUMENTATION

WCAP-14572, Revision 1-NP-A, "Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report," February 1999

WCAP-14572, Revision 1-NP-A, Supplement 1, "Westinghouse Structural Reliability and Risk Assessment (SRRA) Model for Piping Risk-Informed Inservice inspection," February 1999

Supporting Onsite Documentation

Westinghouse Calculation Note: CN-RRA-01-14, Revision 0, "Turkey Point Unit 4 Risk-Informed ISI Program – Segment Definition"

Westinghouse Calculation Note: CN-RRA-01-09, Revision 0, "Turkey Point Unit 4 SRRA for RI-ISI"

Westinghouse Calculation Note: CN-RRA-01-27, Revision 0, "Turkey Point 4 Risk Evaluation for RI-ISI (Revision 0)"

Westinghouse Calculation Note: CN-RRA-01-28, Revision 0, "Turkey Point Unit 4 RI-ISI Expert Panel Database"

Westinghouse Calculation Note: CN-RRA-01-44, Revision 0, "Turkey Point Unit 4 RI-ISI Delta Risk Evaluation"

Westinghouse Calculation Note: CN-RRA-01-33, Revision 0, "RI-ISI Perdue Model Calculations for Turkey Point Unit 4"

Table 3.1-1 System Selection and Segment Definition for Class 1 Piping			
System Description	**PRA**	**Section XI**	**Number of Segments**
CH - Chemical & Volume Control	Yes	Yes	75
RC - Reactor Coolant	Yes	Yes	117
SI - Safety Injection	Yes	Yes	36
Total			228
Note: The table includes 22 segments categorized as "not used" for Unit 4 only to track differences between Units 3 and 4.			

Table 3.3-1 Summary of Postulated Consequences by System	
System	Summary of Consequences
CH - Chemical & Volume Control	The direct consequences associated with piping failures are small LOCA, small-small LOCA, loss of letdown, loss of charging flow from CVCS, and loss of HHSI to one cold leg loop, loss of CVCS injection to RCP seals, and loss of auxiliary spray .
RC - Reactor Coolant	The direct consequences associated with piping failures are large (>13.5"), medium (6" to 13.5"), small (2" to 6") or small-small (3/8" to 2") loss of coolant accidents (LOCAs) and loss of HHSI, RHR/alternate low head, and accumulator injection flow to one cold leg loop, CVCS injection to one cold leg loop, HHSI recirculation to two hot leg loops, loss of letdown, loss of normal and auxiliary pressurizer spray and loss of RHR hot leg suction.
SI - Safety Injection	The direct consequences associated with piping failures are the small-small LOCA, loss of HHSI recirculation to two hot leg loops, loss of HHSI to all cold legs, loss of RHR, alternate low head, and accumulator injection to one cold leg loop.

Table 3.4-1 Failure Probability Estimates (without ISI)				
System	Dominant Potential Degradation Mechanism(s)/ Combination(s)	Failure Probability range at 40 years with no ISI		Comments
		Small leak	Disabling leak (by disabling leak rate)*	
CH	• Thermal Fatigue • Thermal Fatigue and Vibratory Fatigue	5.9E-10 – 1.9E-04 1.4E-05 – 1.0E-03	2.7E-11 – 1.9E-04 6.3E-06 – 4.5E-04	Vibration occurs near the reactor coolant pumps and letdown orifices.
RC	• Thermal Fatigue	3.4E-09 – 1.0E-04	• LLOCA 4.5E-11 – 2.9E-05 • MLOCA 6.8E-11 – 2.9E-05 • SLOCA 6.8E-11 – 2.9E-05 • SSLOCA 6.8E-11 – 1.0E-04	Thermal striping or stratification occurs in the pressurizer surge line. Vibration occurs near the reactor coolant pumps.
	• Thermal Fatigue & Vibratory Fatigue	1.7E-08 – 3.7E-07	• SLOCA 1.3E-08 • SLOCA 1.2E-08 • SYS 9.0E-10	
	• Thermal Fatigue & Striping/Stratification	4.0E-04	• MLOCA 1.6E-04 • SLOCA 1.6E-04 • SSLOCA 1.7E-04	
SI	• Thermal Fatigue	1.0E-05 – 1.9E-04	• SSLOCA 5.6E-05 • SYS 6.3E-06 – 1.9E-04	Vibration occurs on small branch lines connected to the main loop.
	• Thermal Fatigue & Vibratory Fatigue	7.3E-06 – 5.4E-04	• SYS 2.9E-06 – 2.6E-04	

* - Disabling leak rate – LLOCA, MLOCA, SLOCA, SSLOCA and SYS (system disabling leak). When no leak rate is shown, this is the system disabling leak rate.

Table 3.5-1 Number of Segments and Piping Risk Contribution by System (without ISI)					
System	# of Segments	CDF without Operator Action (/yr)	CDF With Operator Action (/yr)	LERF without Operator Action (/yr)	LERF with Operator Action (/yr)
CH	75	7.90E-07	6.02E-07	1.19E-09	9.06E-10
RC	117	1.36E-05	1.32E-05	2.42E-08	2.33E-08
SI	36	2.09E-07	2.09E-07	3.16E-10	3.16E-10
TOTAL	228	1.46E-05	1.40E-05	2.57E-08	2.46E-08

Table 3.7-1 Summary of Risk Evaluation and Expert Panel Categorization Results						
System	Number of segments with any RRW >1.005	Number of segments with any RRW between 1.005 and 1.001	Number of segments with all RRW < 1.001	Number of segments with any RRW between 1.005 and 1.001 placed in HSS	Number of segments with all RRW < 1.001 selected for inspection	Total number of segments selected for inspection (High Safety Significant Segments)
CH	3	30	42	0	0	3
RC	14	49	54	0	0	14
SI	1	10	25	0	0	1
Total	18	89	121*	0	0	18

Note:

* - includes 22 segments categorized as "not used" for Unit 4 only to track differences between Units 3 and 4.

Table 3.10-1 COMPARISON OF CDF/LERF FOR CURRENT SECTION XI AND RISK-INFORMED ISI PROGRAMS AND THE SYSTEMS WHICH CONTRIBUTED SIGNIFICANTLY TO THE CHANGE		
Case (Systems Contributing to Change)	Current Section XI	Risk-Informed
<u>CDF No Operator Action</u>	1.6E-06	1.5E-06
• CH	5.6E-07	5.1E-07
• RC	9.5E-07	9.5E-07
• SI	5.2E-08	6.2E-08
<u>CDF with Operator Action</u>	1.4E-06	1.3E-06
• CH	4.0E-07	3.4E-07
• RC	9.4E-07	9.4E-07
• SI	5.2E-08	6.2E-08
<u>LERF No Operator Action</u>	2.4E-09	2.3E-09
• CH	8.4E-10	7.6E-10
• RC	1.5E-09	1.5E-09
• SI	7.9E-11	9.3E-11
<u>LERF with Operator Action</u>	2.2E-09	2.1E-09
• CH	6.1E-10	5.1E-10
• RC	1.5E-09	1.5E-09
• SI	7.9E-11	9.3E-11

Table 5-1

STRUCTURAL ELEMENT SELECTION RESULTS AND COMPARISON TO ASME SECTION XI 1989 EDITION REQUIREMENTS

System	Number of High Safety Significant Segments (No. of HSS in Augmented Program / Total No. of Segments in Augmented Program)	Degradation Mechanism(s)	Class	ASME Code Category	Weld Count[j]		ASME XI Examination Methods (Volumetric (Vol) and Surface (Sur))		RI-ISI[a]	
					Butt	Socket	Vol & Sur	Sur Only	SES Matrix Region	Number of Exam Locations
CH	3 (0/0)	TF, VF	Class 1	B-F	0	0	0	0	1	3[b]
				B-J	124	166	10	44		
RC	14 (0/0)	TF, VF, Strip/Strat	Class 1	B-F	18	0	9	0	1, 2	23 + 4[b,d]
				B-J	176	28	60	19		
SI	1 (0/0)	TF, VF	Class 1	B-F	0	0	0	0	1	1[b] + 2[b,d] + 1[d]
				B-J	60	172	17	47		
TOTAL	18		Class 1	B-F	18	0	9	0		23NDE + 11VIS
				B-J	360	366	87	110		

Summary: Current ASME Section XI selects a total of 206 non-destructive exams while the proposed RI-ISI program selects a total of 23 exams (34 – 11 visual exams), which results in a 89% reduction.

Degradation Mechanisms: VF – Vibratory Fatigue; TF – Thermal Fatigue; Strip/Strat – Striping/Stratification

Notes for Table 5-1
a. System pressure test requirements and VT-2 visual examinations shall continue to be performed in all ASME Code Class 1, 2, and 3 systems.
b. VT-2 exam for entire segment
c. UT thickness only
d. Seven (7) examinations added for change in risk considerations.