Florida Power & Light Company, 6501 S. Ocean Drive, Jensen Beach, FL 34957

FPL

January 16, 2003

L-2003-009
10 CFR 50.4
10 CFR 50.55a

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

Re: St. Lucie Unit 2
Docket No. 50-389
Inservice-Inspection Plan
Second Ten-Year Interval
Relief Request 29 Request for Additional Information Response

Pursuant to 10 CFR 50.55a (a)(3) on July 23, 2002, Florida Power and Light Company (FPL) requested approval of Relief Request 29, Risk Informed Inservice Inspection Program. On December 17, 2002, the NRC provided FPL a request for additional information (RAI) needed before the NRC could complete their review.

By letter L-2002-142 dated July 23, 2002, FPL requested a change to the St. Lucie Unit 2 ISI Program plan for Class 1 piping only, through the use of a Risk-Informed Inservice Inspection (RI-ISI) Program. The risk-informed process used to prepare the submittal was described in Westinghouse Owners Group WCAP-14572, Revision 1-NP-A, *Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report*. As a risk-informed application, the submittal met the intent and principles of Regulatory Guide (RG) 1.174, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis*, and RG 1.178, *An Approach for Plant-Specific Risk-Informed Decision Making: Inservice Inspection of Piping*. In accordance with 10 CFR 50.55a (a)(3)(i), FPL determined that the proposed alternatives provided an acceptable level of quality and safety.

Attachment 1 is the response to the NRC RAI dated December 17, 2002. Attachment 2 is Revision 2 of the Risk–Informed Inservice Inspection Program Plan for St. Lucie Unit 2. FPL has modified the requested approval of Relief Request 29 to February 28, 2003 to support its use during the spring 2003 St. Lucie Unit 2 refueling outage (SL2-14). Please contact George Madden at 772-467-7155, should there be any additional questions about this submittal.

Very truly yours,

Donald E. Jernigan
Vice President
St. Lucie Plant

DEJ/GRM

Attachment

A047

St. Lucie Unit 2
Request For Additional Information (RAI) Response
Risk-Informed Inservice Inspection (RI-ISI) Relief Request 29

NRC Request 1:

Were all Class 1 piping segments and all B-J and B-F welds included in the evaluation? If not, please provide the following information.

a) With reference to Table 3.1-1 "System Selection and Segment Definition for Class 1 Piping", please provide your rationale for excluding shutdown cooling system from risk evaluation.

b) Are all the piping and welds in the Chemical and Volume Control (CH), Reactor Coolant (RC), and Safety Injection (SI) system included in the scope of the RI-ISI submittal?

c) What percentage of Class 1 welds are excluded from the scope of the RI-ISI submittal?

d) What percentage of butt welds in the portions of the Class 1 piping excluded from the scope of the RI-ISI submittal will continue to be inspected with volumetric examination under the ASME Section XI program?

FPL Response 1:

Yes. The scope of the piping in this submittal is all Class 1 piping, including piping exempt from current Section XI examination requirements. This Class 1 piping scope boundary limits the program to consideration of the reactor coolant pressure boundary portions of the CH, RC, and SI systems, including those portions that provide the shutdown cooling function. No Class 1 welds were excluded.

NRC Request 2:

In Table 3.4-1 "Failure Probability Estimates (without ISI)," please provide justification for not addressing stress corrosion cracking (SCC) as a potential failure mechanism for the charging, safety injection and shutdown cooling systems. How will the failure probability be affected when SCC is considered as a potential degradation mechanism in addition to fatigue and thermal transients?

FPL Response 2:

Table 3.4-1 lists only the dominant potential degradation mechanisms. The parameters indicative of stress corrosion cracking were considered for all segments, and PWSCC was not identified as a potential degradation mechanism. For those segments subject to temperatures greater than 570°F, the failure rates due to fatigue and thermal transients were sufficient to select them as high safety significant.

NRC Request 3:

The industry experience based on cracking of alloy 600 safe-end material (Inconel 82/182) at V.C. Summer attributes the degradation mechanism to be primary water stress corrosion cracking. This degradation mechanism has not been addressed in the Topical Report WCAP-14572, Rev 1-NP-A. As discussed on page 3-87 in the Electric Power Research Institute (EPRI) Topical TR-112657, Rev. B-A, the EPRI location selection process includes guidance that additional locations should also be chosen from segments that are susceptible to a degradation mechanism that is not otherwise inspected and the examination effort should be directed to detect flaws or relevant condition as a result of this degradation mechanism. In Table 5-1, "Structural Element Selection," three welds are selected for volumetric examination in B-F examination category in the reactor coolant system. Are these welds made of Inconel 82/182?

FPL Response 3:

Yes. While none of these welds meet all of the criteria for being subject to PWSCC, all were further evaluated and one was specifically selected because it is greater than 570°F, and therefore, meets two of the criteria.

NRC Request 4:

Under "Deviations," it is stated that the selection of elements is based on that described in EPRI Topical Report TR-112657, Rev. B-A. Are the examination volumes and methods for each of the welds, heat-affected zones and base metal, specific to each degradation mechanism in the affected segment as proposed in EPRI-TR?

FPL Response 4:

The deviation referenced for the selection process is with respect only to the percentage to be selected; i.e., 25% of the high safety significant elements. This is the percentage designated in the EPRI Topical Report for high safety significant segments, and is also the percentage selected in the current Section XI criteria.

The examination volumes and methods are in accordance with Table 1 of Code Case N-577-1, which is identical to that of Code Case N-578-1. Code Case N-578-1 implements the EPRI method. Footnotes 1 and 8 to Table 1 of the Code Cases specify the examination volumes and examination criteria.

NRC Request 5:

On page 2 of the submittal dated July 23, 2002, FPL stated that the methodology used deviated from the methodology presented in Westinghouse Owners Group WCAP-14572, Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report" (WCAP) for selecting elements from Region 1B and 2 of the Structural Element Selection Matrix shown in Figure 3.7-1 of the WCAP. FPL indicated the use of the process described in EPRI TR-112657 to select elements from these regions instead of the one described in the WCAP. On page 10 of the submittal (Section 3.8, Structural Element and NDE [nondestructive examination] Selection), FPL stated that for the 205 piping segments that were evaluated, Region 1B contains 9 segments, Region 2 contains 2 segments, no segments are contained in Region 3, and Region 4 contains 194 segments. Explain why there are no parts of any segments in Region 1A (the placement of segments in Region 1A does not rely on the Perdue method). What failure probability value was used to distinguish elements between Region 1 and Region 2?

FPL Response 5:

Placement in Region 1A is dependent on the presence of an active degradation mechanism. FPL did not identify any active degradation mechanisms.

An annual failure rate of 2.5E-05 was used to differentiate between Regions 1 and 2 simply for reporting purposes. Both Regions 1B and 2 were subject to the same element selection requirements; therefore, this segregation had no impact on the final selection.

NRC Request 6:

Are there any piping segments that include piping of different diameters? If so, how were the failure frequencies estimated for these segments? How does the methodology for determining the failure frequency comport with the methodology described on page 71 of the WCAP?

FPL Response 6:

Each segment consists of a single pipe diameter. The technique used is in full agreement with that described in the WCAP; i.e., the worst case found anywhere in the segment was used for each of the individual parameters input into the calculation.

NRC Request 7:

Table 3.4-1 of the submittal provides the failure probability range for a small leak probability. Provide the disabling leak failure probabilities, and a breakdown in small break loss-of-coolant-accident (LOCA), medium break LOCA, and large break LOCA failure probabilities.

FPL Response 7:

The title of Table 3.4-1 is misleading. The probabilities presented in Table 3.4-1 are in fact those for what is called a "Big Leak" in WinPRAISE.

WinPRAISE requires the input of a specific leak rate, and the probability of a break large enough to result in that leak rate is calculated. For loss of a mitigating system, a leak rate equal to 10% of required system flow was designated. For those segments that could result in a LOCA of any size, a leak rate equal to the Technical Specification limit for leakage inside containment was designated. For conservatism, FPL did not allow any failure rate calculation to run to full break (LOCA) size. The results of the calculations are considered the system disabling leaks, and are the values presented in Table 3.4-1.

NRC Request 8:

Section 6 of the submittal, states that St. Lucie, Unit 2, was designed to ASME Section III and, therefore, has an improved level of fatigue analysis. Most plants were designed to ASME Section III and do not have leak probabilities as low as those presented in Table 3.4-1. Explain what physical characteristics of the St. Lucie reactor cause the estimation of small leak probabilities that are significantly lower than those given in other RI-ISI submittals.

FPL Response 8:

In the WinPRAISE calculations used by FPL, specific values are entered for each parameter, as determined by research of the individual stress calculations and physical specifications for each segment. Since the FPL calculations are based on specific parameters determined for each segment, it is concluded that the specifications for piping at St. Lucie Unit 2 are appropriately stringent to produce lower failure rates.

NRC Request 9:

Under what conditions would the RI-ISI program be resubmitted to the Nuclear Regulatory Commission (NRC) prior to the end of any 10-year interval.

FPL Response 9:

FPL intends to comply with the standards developed by the industry and accepted by the NRC. Specifically, FPL will utilize the guidance in the topical report currently drafted by NEI, upon endorsement by the NRC. The current FPL proposal calls for NRC notification and approval for:

- Changing from one methodology to another
- Changing scope of application
- Plant-specific impact of revised methodology or safety evaluations
- ASME Section XI ten year update
- Changes that impact the basis for NRC approval in the FPL St. Lucie Unit 2 specific Safety Evaluation

NRC Request 10:

Identify the version of the Probabilistic Safety Assessment (PSA) model that was used for the RI-ISI application and when it was last updated. Include when, and which version, of your PSA has been peer reviewed by the Combustion Engineering Owner's Group. Please also provide the Category A and Category B Facts and Observations obtained from your peer review and how these may impact your RI-ISI evaluation.

FPL Response 10:

The version of the Level 1 model used for input to the RI-ISI submittal is dated February 1999. The version of the Level 2 evaluation used is dated May 2001.

The St. Lucie CEOG peer review was conducted the week of May 20,2002. The model reviewed by the peer review team was the draft version of a 2002 update.

The St. Lucie peer review report has not been finalized. The Facts and Observations (F&Os), therefore, are still draft and subject to change. The draft F&Os were reviewed for potential impact on the RI-ISI submittal results. Some of the F&Os are recommendations related only to documentation improvements and thus have no impact on the PSA results. Other F&Os are not related to LOCA scenarios and have an insignificant impact on the Class 1 piping RI-ISI applications. The rest of the F&Os affecting the CDF results are similar to those weaknesses of the IPE submittal and the response to Question 11 below is applicable. The review of the F&Os related to potential model enhancements concluded that the issues addressed by the F&Os would not have a significant impact on the results and conclusions of the RI-ISI evaluation.

NRC Request 11:

The safety evaluation (SE) report on the St. Lucie Individual Plant Examination (IPE), dated July 21, 1997, concluded that the IPE met the intent of Generic Letter 88-20. The SE also stated that "the staff identified weaknesses in the front-end, HRA [human reliability analysis] and back-end portions of the IPE which, we believe, limit its future usefulness." The weaknesses stated in the SE are briefly outlined below. Explain how each of the weaknesses has been removed by modifications to the PRA or otherwise addressed during the RI-ISI evaluation.

NRC Request 11 a):

Some initiating event frequencies appeared low and some initiating event frequencies which relied on generic values should have received a plant-specific analysis.

FPL Response 11 a):

Data update has been performed since the IPE and before the RI-ISI evaluation. The data update included re-quantification of the LOCA initiating event (IE) frequencies based on a CEOG Technical Position Paper. Initiating event fault trees were also developed for loss of CCW, loss of ICW, loss of TCW, loss of DC bus, and loss of instrument air. Plant specific data was used for other initiating events where available.

The impact of the IEs on the RI-ISI is judged to be small as RI-ISI focuses on Class 1 piping and the LOCA frequencies are estimated using probabilistic fracture mechanics. As a sensitivity study, the loss of CCW, loss of ICW, loss of TCW, loss of DC bus, and loss of instrument air IE frequencies were changed to reflect values obtained from quantifying each of the IE fault trees. These revised IE frequencies, along with revised HEPs (discussed in the response to 11(c) below), were used to requantify the "Change in CDF" and CCDP values used to derive the relative risk of each segment.

There is no significant difference between the sensitivity study values for "Change in CDF" and CCDP and those used for the submittal (the difference in the CCDP ranges from -2.3% to +2.5%). It is judged that this IE data issue does not have a significant impact on the results and conclusions for the RI-ISI application.

NRC Request 11 b):

Some preinitiator human actions appeared in dominant accident sequences, with unexpected and uncommon results. It appears that a more detailed analysis of preinitiator human actions may appropriately reduce the human error probabilities (HEPs) for these events, thus reducing the likelihood that excessively conservative HEPs may distort the risk profile.

FPL Response 11 b):

Screening values have been used in all updates to date. The LOCA initiator related cutset files generated in support of the RI-ISI submittal were reviewed to determine the Fussel-Vesely (FV) values for the dominant pre-initiator actions. The largest FV value noted was 3E-02. Only in three other cases were values slightly greater than 1E-02. The risk reduction worth for the dominant pre-initiator would thus only be 3%. The potential impact on CDF due to more detailed analyses of the pre-initiators would thus be much less than 3% for the dominant contributor and much less than 1% for most. It is judged that the use of unrefined pre-initiator screening values does not have a significant impact on the results and conclusions for the RI-ISI application.

NRC Request 11 c):

It was not clear what basis was used to determine which postinitiator human actions were quantified with a time-independent technique and those postinitiator actions that were quantified with a time-dependent technique. Three postinitiator human actions (initiating once-through cooling, manually initiating recirculation actuation components following loss of the automatic signal, and securing the reactor coolant pumps after loss of seal cooling) are relatively short time frame events. Failure to consider time in these events might lead to unrealistic values.

FPL Response 11 c):

No changes to the Human Reliability Analysis (HRA) to address this issue have been implemented for the PSA updates to date. The St. Lucie IPE SER states that "the HEPs for the events modeled as slips were not unreasonable and several of the events modeled in this way still show up as being important. Therefore, there is no reason to believe that the approach necessarily precluded detection of HRA related vulnerabilities."

For Class 1 piping RI-ISI applications, the use of a different HRA method is not expected to affect the conditional core damage probability given pipe breaks. As discussed in the response to Question 11(a) above, a sensitivity study was performed using updated IE frequencies and updated HEPs. The specific HRA events identified in the IPE review and other potentially significant HRA events that were originally quantified using a time-independent technique were re-quantified as time-dependent actions. The resulting HEPs, along with the revised IE frequencies discussed in the response to Question 11(a) above, were used to re-quantify the "Change in CDF" and CCDP values used to derive the relative risk of each segment.

There is no significant difference between the sensitivity study values for "Change in CDF" and CCDP and those used for the submittal (the difference in the CCDP ranges from -2.3% to +2.5%). It is judged that this HRA issue does not have a significant impact on the results and conclusions for the RI-ISI application.

NRC Request 11 d):

The time-dependent human actions used likelihood indices at their default values. Therefore, the resulting human error probabilities may be generic rather than plant-specific.

FPL Response 11 d):

No changes to the HRA to address this issue have been implemented for the PSA updates to date. The St. Lucie IPE SER states that in general, the way in which the

SAIC time-dependent method was applied in the IPE did not appear to violate its basic tenets and that resulting HEPs would not be considered unusual. The SER also states that most of the HEP values themselves would not suggest that identification of human action vulnerabilities was precluded.

As discussed in the response to Question 11(a) above, a sensitivity study was performed using updated HEPs for events previously quantified as time-independent. The methodology used to calculate the revised HEPs addresses plant specific factors.

The process for evaluating individual human interactions breaks down the detection, diagnosis, and decision-making aspects into different failure mechanisms, with causes of failure delineated for each. Eight different potential failure mechanisms are identified:

- Availability of information
- Failure of attention
- Misread/miscommunicate data
- Information misleading
- Skip a step in procedure
- Misinterpret instruction
- Misinterpret decision logic
- Deliberate violation

A relatively simple decision tree is used for each of these mechanisms. Each of these decision trees identifies performance shaping factors that could cause the relevant mechanism to lead to failure to initiate the proper action. The analyst selects branch points in the decision trees that correspond to the aspects of the interaction being analyzed (e.g., the number and quality of cues for the operators, the ease of use of the procedures, etc.). For each outcome in the decision trees, there is a nominal probability of failure.

Depending on the failure cause, certain recovery mechanisms may come into play. The potential for recovery may arise as follows:

- due to self-review by the operator initially responsible for the misdiagnosis or error in decision-making, as additional cues become available or additional procedural steps provide opportunity to review actions that have been taken and the resulting effects on the plant;
- as a result of review by other crew members who would be in a position to recognize the lack of proper response;
- by the STA, whose review might identify errors in response;
- by the technical support center (TSC) when it is staffed and actively involved in reviewing the situation; and
- by oncoming crewmembers when there is a shift turnover (when the time window is very long).

Thus, after processing each of the decision trees to arrive at estimates for the basic failure mechanisms, the analyst must identify and characterize the appropriate recovery factors.

There are other considerations besides time that affect the treatment of the non-recovery potential. These included the degree to which new or repeated cues and recurring procedural steps would give rise to considering the action that had not been successfully taken.

Another element represents failure to implement the action correctly, given that the decision is made to initiate the action. A basic task analysis is performed to identify the essential steps that must be accomplished to implement a decision. The corresponding failures to perform them properly are noted. These failures are then quantified.

In considering the execution errors, three levels of stress were identified: optimal, moderately high, and extremely high. Optimal stress would apply for actions that are part of a normal response to a reactor trip, and for which the operators would be alert. Moderately high stress would apply when the operators are responding to unusual events, including multiple failures. Extremely high stress would apply for scenarios in which there is a significant threat, such as the potential that core damage is imminent if the actions are not successful, or when actions must be accomplished under significantly less than optimal conditions.

The execution errors may be subject to review and recovery as well. This is particularly true for actions taken in the control room, where additional observers may be able to identify the need for corrective action. As in the case of the initiation errors, a set of guidelines for considering review and recovery by other crewmembers has been developed.

Based on the discussion above, it can be seen that the revised HEPs used in the sensitivity study takes into account plant specific factors.

It is judged that this HRA issue does not have a significant impact on the results and conclusions for the RI-ISI application.

<u>NRC Request 11 e):</u>

An additional sensitivity analysis should have been performed regarding the probability of in-vessel recovery since the licensee assumed a very high probability of in-vessel recovery due to ex-vessel cooling.

<u>FPL Response 11 e):</u>

For the Class 1 piping RI-ISI application, conservatism embedded in the IPE with respect to other dominant early containment failure mechanisms (e.g., direct containment heating, steam explosion, and the vessel acting like a rocket) outweigh the risk impact of variations in the probability used for in-vessel recovery. The revised Level 2 analysis, incorporating insights since the IPE submittal, indicates that the large early containment failure probability, assuming 25% for ex-vessel cooling, is less than 1%.

NRC Request 12:

On Table 3.5-1 of the submittal, you present the piping risk contribution by system without ISI. Explain how the risk contribution can be so low in the SI system given that the system contains 59 segments. Also, please provide the conditional core damage probability for small, medium, and large break LOCAs.

FPL Response 12:

The risk contribution of each individual segment in the SI system was evaluated. The relative low ranking of the segments in the SI system was due to the low failure rates for each segment. Reference the response to Item 8 for a discussion of the failure probability calculations.

LOCAs at PSL are designated as small-small loca, small LOCA, and large LOCA. Their CCDPS are:

Description	Designator	CDF	Freq	CCDP
Small-Small LOCA	SSLOCA	8.15E-06	3.01E-03	2.71E-03
Small LOCA	SLOCA	6.30E-08	2.16E-05	2.92E-03
Large LOCA	LLOCA	7.51E-07	5.85E-05	1.28E-02

NRC Request 13:

Table 3.10-1 presents a comparison of core damage frequency/ large early release frequency for current Section XI and RI-ISI programs. Page 214 of the WCAP presents criteria for the evaluation of these results.

NRC Request 13 a):

Criterion 1 states that the total change in piping risk should be risk neutral or a risk reduction. The risk from the SI and RC systems remain the same but the risk from the CH system, although small, increases. Was an analysis performed of the dominant system and piping segment contributors to the RI-ISI? If so, provide any added inspection locations.

FPL Response 13 a):

The risk from the CH system was analyzed. The single segment that resulted in the difference between the current Section XI and the RI-ISI program is small bore piping with socket welds only. Since the RI-ISI examination specified for socket welds is a VT-2 examination that may be conducted during a system pressure test, there is actually no difference between the current Section XI program and the proposed examinations, since VT-2 visual examinations are scheduled in accordance with the plant's pressure test program, which remains unaffected by the RI-ISI program. This was not credited in Table 3.10-1, since the segment was not determined to be HSS and was not specified as a RI-ISI examination.

NRC Request 13 b):

Criterion 2 states that any system that contributes greater than 10 percent of the total risk for RI-ISI should be examined when the overall risk has not changed. The criteria further state that any such identified systems should be reevaluated in an attempt to identify additional examinations which would reduce the risk for these systems and, thus, the overall risk. It appears from Table 3.10-1 that RC system contributes greater than 10 percent of the total risk for RI-ISI and the overall risk when implementing RI-ISI has not changed. Explain why additional examinations were not added to reduce risk in the RC system and thus reduce overall risk.

FPL Response 13 b):

In the original program development, the Expert Panel reevaluated the reactor coolant system. Two additional segments were added to the initial list of HSS segments. These segments were selected as the next highest contributors to risk based on the sensitivity and uncertainty studies, and the fact that they contributed to the shutdown cooling function. These additional examinations were added, but did not reduce risk in the RC system or reduce overall risk. Selection of additional elements would not contribute to risk reduction.

NRC Request 14:

Section 3.5 discusses an uncertainty analysis done in support of the submittal.

NRC Request 14 a):

Explain how this uncertainty analysis comports or differs from the uncertainty analysis required on page 125 of the WCAP.

FPL Response 14 a):

The uncertainty analysis was performed in accordance with the guidance on page 125 of the WCAP. Reasonable variations were considered with respect to the conditional CDF/LERF probabilities by using the 5th and 95th percentile bounds of an uncertainty distribution on the appropriate initiating events to calculate new CDF/LERF values and the related RRWs. Reasonable variations in pipe failure were considered by varying the failure rate by plus and minus an order of magnitude. This simplified analysis verified that no low safety significance segments moved into high safety significance. The results of this evaluation were provided to the Expert Panel, and were part of the consideration that moved two segments from the medium to the high classification.

As a validation process, an uncertainty analysis using the @RISK program was performed. This analysis utilized the results of each WinPRAISE calculation as the point estimate for failure rate. Following an iteration process to refine bounding values in the original calculation, the @RISK results validated the results of the simplified analysis originally performed at St. Lucie Unit 2.

NRC Request 14 b):

Explain how the 5 percent and 95 percent percentile bounds were developed for the piping failure frequency and provide some representative ranges.

FPL Response 14 b):

The failure probability calculated by WinPRAISE was used as the point estimate (median). Ranges for the highest significance segment, and for the segment that moved from medium to high significance in the analysis are:

Segment	WinPRAISE point estimate (median)	Mean	5th percentile	95th percentile
2-RC-011	2.85E-03	7.59E-03	2.85E-04	2.85E-02
2-RC-001	2.45E-05	1.29E-04	1.22E-06	4.89E-04

NRC Request 15:

Section 3.8 of the licensee's submittal addresses additional examinations. It states, "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."

ASME Code directs licensees to perform these sample expansions in the current outage that the flaws or relevant conditions were identified. Verify in what time frame the sample expansions will be completed.

<u>FPL Response 15:</u>

The FPL PSL RI-ISI program was developed in accordance with the Westinghouse Owners Group WCAP-14572. This technique has been incorporated into the ASME Code through Code Case N-577-1 for trial use. A non-mandatory appendix is currently under development for permanent incorporation in Section XI. Both the code case and the proposed non-mandatory appendix state that an evaluation shall be performed to establish when additional examinations are to be conducted. It is the intent of FPL to comply with the version of the Code that is approved incorporating the WCAP technique, including any conditions that may be imposed.

Florida Power and Light Company

St. Lucie Unit 2

Risk-Informed Inservice Inspection Piping Program Submittal Using the Westinghouse Owners Group (WOG) Methodology (WCAP-14572, Revision 1-NP-A, February 1999)

December 2002

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. St. Lucie Unit 2 is currently in the second 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 Class 1 piping only through the use of a risk-informed inservice inspection (RI-ISI) program. The risk-informed process used in this submittal is described in Westinghouse Owners Group WCAP-14572, Revision 1-NP-A, *Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report*, (referred to as "WCAP-14572, A-Version" for the remainder of this document).

As a risk-informed application, this submittal meets the intent and principles of Regulatory Guides 1.174 and 1.178. Further information is provided in Section 3.10 relative to defense-in-depth.

1.2 PRA Quality

The St. Lucie Unit 2 Probabilistic Safety Assessment (PSA) baseline model was used to evaluate the consequences of pipe ruptures. The base core damage frequency (CDF) and the base large early release frequency (LERF) are 1.25E-05 and 6.00E-06, respectively.

The baseline model used for this RI-ISI evaluation was generated using the IPE model developed in response to Generic Letter (GL) 88-20, *Individual Plant Examination for Severe Accident Vulnerabilities*, and associated supplements. The original development work was classified and performed as "quality related" under the FPL 10 CFR Appendix B quality assurance program. The revision and applications of the PRA models and associated databases continue to be handled as quality related. Administrative controls include written procedures, independent review of all model changes, data updates, and risk assessments performed using PSA methods and models. Risk assessments are performed by one PSA engineer, independently reviewed by another PSA engineer, and approved by the department head or designee. The PSA group falls under the FPL Engineering Quality Instructions (QI) with written procedures derived from those QIs. Procedures, risk assessment documentation, and associated records are controlled and retained as QA records.

Since the approval of the IPE, the FPL Reliability and Risk Assessment Group (RRAG) has maintained the PSA models consistent with the current plant configuration such that they are considered "living" models. The PSA models are updated for different reasons, including plant changes and modifications, procedure changes, accrual of new plant data, discovery of modeling errors, advances in PSA technology, and issuance of new industry PSA standards. The update process ensures that the applicable changes are implemented and documented timely so that risk analyses performed in support of plant operation reflect the plant configuration, operating philosophy, and transient and component failure history. The PSA maintenance and update process is described in FPL RRAG standard *PSA Update and Maintenance Procedure*, This standard defines two different types of periodic updates: 1) a data analysis update, and 2) a model update. The data analysis update is performed at least every five years. Model updates consist of either single or multiple PSA changes and are performed at a frequency dependent on the estimated impact of the accumulated changes. Guidelines to determine the need for a model update are provided in the standard. The maintenance rule program developed to implement the requirements of 10 CFR 50.65 is also based on this PSA. The PSA model was also used to justify a risk-informed evaluation to support a technical specification change request to extend the diesel generator allowed outage time (AOT). The technical specification change request is currently under NRC review.

The St. Lucie Unit 2 PSA model uses a large fault tree/small event tree method of quantification. Event tree models were developed to define the logic for core damage sequences. The event tree models were converted to equivalent fault tree logic and linked to the frontline and support system fault tree models. The core damage sequence gates were combined into a single-top core damage gate using "OR" logic. The single-top core damage gate was quantified to obtain core damage cutsets in terms of basic events. The core damage cutsets were used to obtain the CDF values. Each quantification involves post-process operations on the quantified "raw" cutsets. Cutsets containing pre-defined, mutually exclusive event combinations, were removed from the final cutset listing. Finally, recovery events were applied to selected cutsets based on pre-defined recovery rules. EPRI's Risk & Reliability Software Package and the NURELMCS code were used to perform the quantification of CDF values.

For this RI-ISI application, the impact of pipe breaks were simulated by defining surrogate basic event in the fault tree modes and using the events to configure the fault tree models prior to the quantification process. If a pipe break did not result in an initiating event, the appropriate basic event(s) were set to a logical "TRUE" state prior to each fault tree quantification to simulate failure of a mitigation system or function due to the pipe break. If a pipe break resulted in an initiating event, the appropriate basic event(s) were set equal to the initiating event prior to each fault tree quantification to simulate the impact of the pipe break initiating event on mitigation systems or functions. Existing basic events in the model were used as the preferred method of simulating the postulated pipe break. New surrogate basic events were added to the model, as required, to properly simulate the impact of the postulated pipe break when existing events were not adequate.

The Level 2 evaluation determines that for Unit 2, LERF comprises 1% of CDF, except for those degradations that result in the inability to mitigate steam generator tube ruptures or interfacing systems LOCAs.

Since the St. Lucie Unit 2 PSA model has been used for Maintenance Rule risk ranking applications and risk-informed technical specification requests, it is concluded that, on a relative basis, the PSA method and model would yield meaningful rankings for RI-ISI evaluations when combined with deterministic insights.

2. PROPOSED ALTERNATIVE TO ISI PROGRAM

2.1 ASME Section XI

ASME Section XI Class 1 Categories B-F and B-J currently contain the requirements for examining (via non-destructive examination (NDE) Class 1 piping components. This RI-ISI program is limited to ASME Class 1 piping, including piping currently exempt from requirements. The alternative RI-ISI program for piping is described in WCAP-14572, A-Version. The Class 1 RI-ISI program will be substituted for 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 inspection programs for the St. Lucie Unit 2 Class 1 piping systems.

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 two deviations to the process described in WCAP-14572, A-Version:

WCAP-14572 uses the Westinghouse Structural Reliability and Risk Assessment Model (SRRA) to calculate failure rates. Since SRRA is a Westinghouse product and St. Lucie is a CE plant, FPL uses WinPRAISE, a Microsoft Windows based version of the PRAISE code used as the benchmark for SRRA in WCAP-14572, Supplement 1.

In WCAP-14572, selection of elements in Regions 1B and 2 of the Structural Element Selection Matrix shown in Figure 3.7-1 of the WCAP is determined by a statistical evaluation process. Since the statistical model used in the WCAP is a proprietary Westinghouse product and St. Lucie is a CE plant, an alternative selection process was used. The alternative is based on that described in EPRI Topical Report TR-112657, Revision B-A, approved in a safety evaluation report dated October 28, 1999 and on current ASME Section XI criteria. The alternative process selected 25% of the elements in each high safety significance segment. This resulted in the selection of 27.7% of the total population of elements in the high safety significance segments.

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 risk-informed ISI program are provided in Table 3.1-1.

3.2 Segment Definitions

Once the scope of the program is determined, the piping for these systems is divided into segments.

The numbers of pipe segments defined for the Class 1 piping systems are 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 frequency. The impact on these measures due to both direct and indirect effects was considered.

A review of the license basis of St. Lucie (Updated Final Safety Analysis Report Amendment No. 13) and the IPE internal events methodology was performed to determine the potential impact of the indirect effects of pipe leak or rupture inside containment. As a result of the review, it was 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. Table 3.3-1 summarizes the postulated consequences for each system.

3.4 Failure Assessment

Failure estimates were generated utilizing industry failure history, plant specific failure history, and other industry relevant information.

The engineering team that performed this evaluation used WinPRAISE, a Microsoft Windows based version of the PRAISE code used as the benchmark for SRRA in WCAP-14572, Supplement 1. The failure rate for each segment was based on an aggregate condition, utilizing a combination of the highest individual values of each parameter input to the calculation.

Table 3.4-1 summarizes the failure probability estimates for the dominant potential failure mechanism(s)/combination(s) by system. Table 3.4-1 also describes why the failure mechanisms could occur at various locations within the system.

No augmented inspections are performed for the Class 1 piping.

3.5 Risk Evaluation

Each piping segment within the scope of the program was evaluated to determine its CDF and 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 results of these calculations are presented in Table 3.5-1. The expected value for core damage frequency due to piping failure without operator action is 9.365E-05/year, and with operator action is 9.364E-05/year. The expected value for large early release frequency due to piping failure without operator action is 9.365E-07/year, and with operator action is 9.364E-07/year. This evaluation also included a fifth and ninty-fifth percentile uncertainty analysis.

To assess safety significance, the risk reduction worth (RRW) and risk achievement worth (RAW) importance measures 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 comprised 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. Maintenance rule expert panel members were used to ensure consistency with the other PSA applications.

The expert panel had the following positions represented during the expert panel meeting.

- Probabilistic Safety Assessment (PSA engineer)
- Maintenance Rule (Chairman)
- Operations (Senior Reactor Operator)
- Inservice Inspection (ISI & NDE)
- Plant & Industry Maintenance, Repair, and Failure History (System Engineer)
- Materials Engineer
- Stress Engineer

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

The system and component engineering manager is the chairman of the expert panel. The maintenance rule administrator may act as alternate chairman.

Members received training and indoctrination in the risk-informed inservice inspection 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, PSA modeling limitations, and the use of expert judgment. Training documentation is maintained with the expert panel's records.

Worksheets were provided to the panel 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.

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 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 non-destructive examination methods were defined.

The program being submitted addresses the high safety significant (HSS) piping components placed in regions 1 and 2 of Figure 3.7-1 and described in Section 3.7.1 in WCAP-14572, A-Version. Region 3 piping components, which are low safety significant, are to be considered in an owner defined program and is not considered part of the program requiring NRC 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 205 piping segments that were evaluated in the RI-ISI program, Region 1B contains 9 segments, Region 2 contains 2 segments, no segments are contained in Region 3, and Region 4 contains 194 segments.

The number of locations to be inspected in applicable HSS segments was determined using a selection process based on that described in EPRI Topical Report TR-112657, Revision B-A, approved in a safety evaluation report dated October 28, 1999 and on current ASME Section XI criteria. The process selected 25% of the elements in each high safety significance segment. This resulted in the selection of 27.7% of the total population of elements in the high safety significance segments.

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 which remains unaffected by the risk-informed inspection program.

Additional Examinations

The risk-informed inspection program in all cases will determine through an engineering evaluation the root cause of any unacceptable flaw or relevant condition found during examination. The evaluation will include the applicable service conditions and degradation mechanisms to establish that the element(s) will still perform their intended safety function during subsequent operation. Elements not meeting this requirement will be repaired or replaced.

The evaluation will include 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

An attempt shall be made to provide a minimum of >90% coverage criteria (per ASME Code Case N-460) when performing an exam. 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.

In instances where it may be found at the time of the examination that a location does not meet >90% coverage, the process outlined in Section 4.0 (Inspection Program Requirements) of WCAP-14572, A-Version will be followed.

3.10 Change in Risk

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

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 inservice inspections with a "good" probability of detection in the WinPRAISE model and followed the guidelines provided on page 213 of WCAP-14572.

The results from the risk comparison are shown in Table 3.10-1. As seen from the table, the overall RI-ISI program maintains the risk associated with piping CDF/LERF, with respect to the current Section XI program, while reducing the number of examinations. The primary basis for being able to maintain risk with a reduced number of examinations is that exams are now being placed on piping segments that are high safety significant, and in some cases elements are inspected that are not inspected by NDE in the current ASME Section XI ISI program.

<u>Defense-In-Depth</u>

The reactor coolant piping will continue to receive a system leakage test and visual VT-2 examination as currently required by the Code. Volumetric examinations will also continue on the main reactor coolant piping as part of the RI-ISI program (segments categorized HSS). These locations, which include main loop and pressurizer surge line piping welds determined by the RI-ISI program for St. Lucie Unit 2, assure that "defense-in-depth" is maintained. No additional inspection locations are required to meet "defense-in-depth."

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 Technical Specifications or 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.

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 Section XI inspection period basis. Significant changes may require more expedited 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. The plant will be performing examinations on elements not currently required to be examined by ASME Section XI. The current ASME Section XI program selects a prescribed percentage of examinations without regard to safety significance. The RI-ISI program focuses examinations on those high safety significant segments and subsequently examinations are required on inspection elements not currently scheduled for examination by the ASME Section XI program.

The program was be started in the third period of the second interval, starting in the outage that began November 19, 2001. Currently, 68% of the exams in the Section XI program have been performed, meeting the 50% requirement for the end of the second inspection period of the current interval.

6. SUMMARY OF RESULTS AND CONCLUSIONS

A partial scope Class 1 risk-informed ISI application has been completed for Unit 2. 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 increased volume of examination.

The construction permit for St. Lucie Unit 2 was issued May 1977. The plant is designed to ASME Section III for the Class 1 piping. The ASME Section III design provides an improved level of fatigue analysis and operating conditions scrutiny when compared to older vintage plants. This results in a larger percentage of the reactor coolant system piping constructed with butt welds as opposed to socket welds and more detailed information is available for input to the estimation of the failure probability.

From a risk perspective, the PRA dominant accident sequences include small LOCA; loss of offsite power, and large LOCA.

For the RI-ISI program, appropriate sensitivity and uncertainty evaluations have been performed to address variations in piping failure probabilities and PRA consequence values along with consideration of deterministic insights to assure that all high safety significant piping segments have been identified.

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
- Calculation Number PSL-BFJR-98-004, Revision 2*, St. Lucie Units 1 & 2 Baseline EOOS Models.*
- St. Lucie Units 1 & 2 Individual Plant Examination Submittal, Revision 0, dated December 1993.
- St. Lucie Unit 2 Probabilistic Safety Assessment Update, Revision 0, 1996.
- Procedure RRAG-GEN-002, *PSA Update and Maintenance Procedure*, Revision 2, Dated July 25, 1996.
- Calculation Number PSL-BFJR-96-007, Revision 4," *Documentation of St. Lucie Pre-Evaluated Maintenance Risk Assessments (PREMRA's)*
- Risk & Reliability Software developed for the electric power industry under sponsorship of EPRI, the Electric Power Research Institute.
- NURELMCS, SCIENTECH, Version 2.20, Revision 1.8

Supporting Onsite Documentation

The onsite documentation is contained within the following Engineering Evaluations:

- PSL-ENG-SEOS-01-002, *St. Lucie Unit 2 Risk-Informed ISI Program Development Analysis*
- PSL-ENG-SEOS-01-003, *St. Lucie Unit 2 Risk-Informed ISI Program – Failure Analysis*
- PSL-ENG-SEOS-01-004, *St. Lucie Unit 2 Risk-Informed ISI Program – Consequence Quantification*

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	20
RC - Reactor Coolant [1]	Yes	Yes	126
SI - Safety Injection [1,2]	Yes	Yes	59
Total			205

Notes:

1. Includes shutdown cooling flow paths
2. Includes flow paths for high pressure safety injection, low pressure safety injection, and the passive accumulator in portions of SI.

Table 3.3-1
Summary of Postulated Consequences by System

System	Summary of Consequences
CH – Chemical & Volume Control	The direct consequences postulated from piping failures in this system are: loss of auxiliary pressurizer spray flow path; loss of one or more trains for charging; and small-small loss of coolant accident (LOCA).
RC – Reactor Coolant	The direct consequences associated with piping failures are: large, small, and/or small-small LOCAs; loss of safety injection tank flow path; loss of cold or hot injection leg flow path; loss of alternate injection flow path; loss of auxiliary pressurizer spray flow path; loss of one or more charging flow paths; and loss of identified instrumentation; loss of shutdown cooling flow path.
SI – Safety Injection	The direct consequences associated with piping failures are: loss of safety injection tank flow path; loss of low pressure safety injection (LPSI) flowpath; loss of cold or hot leg injection flow path; loss of alternate injection flowpath; piping break outside primary containment; large and/or small-small LOCAs; loss of suction to LPSI pump; loss of identified instrumentation; loss of shutdown cooling flow path.

Table 3.4-1
Failure Probability Estimates (without ISI)

System	Dominant Potential Degradation Mechanism(s)/ Combination(s)	Failure Probability Range (Small Leak Probability @ 40 years, no ISI)	Comments
CH	-Fatigue	2.01E-12 – 6.80E-09	The charging path to the applicable RCS loop is potentially susceptible to thermal fatigue
RC	-Fatigue -Thermal Transients -Thermal and Vibratory Fatigue	1.66E-14 – 4.82E-06 5.67E-14 – 2.85E-03 1.4E-06 – 4.6E-05	Fatigue at instrument line connections to main loop. Piping where large thermal transients could occur: pressurizer surge line and charging nozzles The piping is located on the RCP pump or seal housing and is potentially subject to vibration.
SI	- Fatigue - Thermal Transients	1.85E-15 – 2.07E-11 8.21E-15 – 5.83E-14	Piping in flow path of alternate injection and SIT is potentially susceptible to thermal fatigue. Potential piping locations where thermal transients could occur in injection lines.

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	20	4.041E-11	3.963E-11	4.041E-13	3.963E-13
RC	126	9.365E-05	9.364E-05	9.365E-07	9.364E-07
SI	59	1.954E-16	1.954E-16	1.954E-18	1.954E-18
TOTAL	205	9.365E-05	9.364E-05	9.365E-07	9.364E-07

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	0	0	20	0	0	0
RC	9	2	115	2	0	11
SI	0	0	59	0	0	0
Total	9	2	194	2	0	11

Table 3.10-1 COMPARISON OF CDF/LERF FOR CURRENT SECTION XI AND RISK-INFORMED ISI PROGRAMS		
Case	Current Section XI	Risk-Informed
CDF No Operator Action	8.03E-05	8.03E-05
• CH	2.46E-11	2.46E-11
• RC	8.03E-05	8.03E-05
• SI	1.60E-18	1.60E-18
CDF with Operator Action	8.03E-05	8.03E-05
• CH	2.39E-11	2.39E-11
• RC	8.03E-05	8.03E-05
• SI	1.60E-18	1.60E-18
LERF No Operator Action	8.03E-07	8.03E-07
• CH	2.46E-13	2.46E-13
• RC	8.03E-07	8.03E-07
• SI	1.60E-20	1.60E-20
LERF with Operator Action	8.03E-07	8.03E-07
• CH	2.39E-13	2.39E-13
• RC	8.03E-07	8.03E-07
• SI	1.60E-20	1.60E-20

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 Aug. Program / Total No. of Segments in Aug. Program)	Degradation Mechanism(s)	Class	ASME Code Category	Weld Count		ASME XI Examination Methods (Volumetric (Vol) and Surface (Sur))		RI-ISI	
					Butt	Socket	Vol & Sur	Sur Only	SES Matrix Region	Number of Exam Locations
CH	0	Thermal Fatigue	1	B-F	3	0	0	3	-	0
				B-J	30	115	0	81		
RC	11 (0/0)	Thermal Fatigue, Thermal Transients, Vibration Fatigue	1	B-F	20	0	12	8	1B, 2	3 volumetric
				B-J	202	20	50	24		21 volumetric
SI	0	Thermal Fatigue, Thermal Transients	1	B-F	6	0	6	0	-	0
				B-J	143	22	18	5		
TOTAL	11 (0/0)		CL. 1	B-F	29	0	18	11		3 NDE
				B-J	375	157	68	110		20 NDE
			TOTAL		404	157	86	121		22 NDE

Summary: Current ASME Section XI selects a total of 86 non-destructive exams (surface only exams not included), while the proposed RI-ISI program selects a total of 24 non-destructive exams. This results in a 72% reduction of non-destructive exams.

General Note:
System pressure test requirements and VT-2 visual examinations shall continue to be performed in ASME Class 1 systems.