Progress Energy

Crystal River Nuclear Plant
Docket No. 50-302
Operating License No. DPR-72

Ref: 10 CFR 50.55a

November 10, 2004
3F1104-02

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Subject: Crystal River Unit 3 – Third 10-Year Inservice Inspection Program – Request for Approval of Risk-Informed Inservice Inspection Program for Class 1, ASME Code, Category B-J and B-F Piping Welds

Reference:
1. EPRI TR-112657, Rev. B-A, Final Report, "Revised Risk-Informed Inservice Inspection Evaluation Procedure," July 1999
2. NRC letter dated August 5, 1999, "Evaluation of Third 10-Year Interval Inservice Inspection Program Requests For Relief For Crystal River 3" (TAC No. MA1642)

Dear Sir:

Florida Power Corporation, doing business as Progress Energy Florida, Inc. (PEF), is requesting relief, in accordance with 10 CFR 50.55a(a)(3)(i), from the requirements of the 1989 Edition of the ASME Code, Section XI in order to implement a Risk-Informed Inservice Inspection (RI-ISI) Program for Class 1, Code Category B-J and B-F piping welds. The program provided in Attachment A has been developed in accordance with the Electric Power Research Institute (EPRI) methodology contained in EPRI Topical Report TR-112657, "Revised Risk-Informed Inservice Inspection Evaluation Procedure," Revision B-A (Reference 1). EPRI Report TR-112657 was approved by the NRC in a Safety Evaluation Report dated October 28, 1999. The proposed alternative described in the Attachment provides an acceptable level of quality and safety as required in 10 CFR 50.55a(a)(3)(i).

Crystal River Unit 3 (CR-3) plans to implement the proposed RI-ISI Program during the third period of the third 10-year inservice inspection interval. The third 10-year inspection interval began on August 14, 1998. CR-3 is requesting approval of the RI-ISI by August 1, 2005, in order to support planning for Refueling Outage 14, scheduled for Fall 2005.

In accordance with EPRI Topical Report TR-112657 (Reference 1), CR-3 has reviewed the impact of the RI-ISI on previous relief requests. As a result of the review, CR-3 has identified the need to withdraw Relief Request 98-001-II upon NRC approval of the attached RI-ISI Program since RI-ISI does not require an outer surface examination. Relief Request 98-001-II allows performance of volumetric examination of the Core Flood Nozzle-to-safe end welds in lieu of the ASME Code required surface examination. The relief request was approved by the NRC as part of the Third 10-Year Interval ISI Program Requests for Relief (Reference 2).

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The NRC has previously approved RI-ISI Programs, based on EPRI Topical Report TR-112657, for Comanche Peak (TAC Nos. MB 1201 and MB 1202) and South Texas Project (TAC Nos. MA 7789 and 7790).

Regulatory commitments made in this submittal are provided in Attachment B.

If you have any questions regarding this submittal, please contact Mr. Sid Powell, Supervisor, Licensing and Regulatory Programs at (352) 563-4883.

Sincerely,

Michael J. Annacone
Manager Engineering

MJA/lvc

Attachments:

A. Risk-Informed Inservice Inspection (RI-ISI) Program Plan
B. List of Regulatory Commitments

xc: NRR Project Manager
Regional Administrator, Region II
Senior Resident Inspector

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72

ATTACHMENT A

Risk Informed Inservice Inspection (RI-ISI) Program Plan

RISK-INFORMED INSERVICE INSPECTION PROGRAM PLAN CRYSTAL RIVER UNIT 3, REVISION 0

Table of Contents

1. INTRODUCTION

Crystal River Unit 3 (CR-3) is currently in the third inservice inspection (ISI) interval as defined by the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Section XI Code for Inspection Program B. CR-3 plans to start implementing a risk-informed inservice inspection (RI-ISI) program during the third inspection period. Initial RI-ISI Program implementation is planned for the plant's Refueling Outage 14 (RFO14) scheduled for Fall 2005. The ASME Section XI Code of Record for the third ISI interval at CR-3 is the 1989 Edition.

The objective of this submittal is to request the use of a risk-informed process for the inservice inspection of Class 1 piping. The RI-ISI process used in this submittal is described in Electric Power Research Institute (EPRI) Topical Report (TR) 112657, Rev. B-A, Final Report, "Revised Risk-Informed Inservice Inspection Evaluation Procedure." The RI-ISI application was also conducted in a manner consistent with ASME Code Case N-578, "Risk-Informed Requirements for Class 1, 2, and 3 Piping, Method B."

1.1 Relation to NRC Regulatory Guides 1.174 and 1.178

As a risk-informed application, this submittal meets the intent and principles of Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis," and Regulatory Guide 1.178, "An Approach for Plant-Specific Risk-Informed Decisionmaking for Inservice Inspection of Piping." Further information is provided in Section 3.6.2 relative to defense-in-depth.

1.2 PSA Quality

The PSA model used in the RI-ISI consequence evaluation is "CR-3 PSA Model of Record – MOR02." The base case Core Damage Frequency (CDF) and Large Early Release Frequency (LERF) values are given below:

- CDF = 6.83×10^{-6} per year
- LERF = 3.59×10^{-7} per year

These numbers are generally lower than the industry average. A review of the results and the model has provided some reasons for a low CDF and LERF:

- Byron Jackson N-9000 Reactor Coolant Pump (RCP) seals are installed and are assumed to maintain their integrity as long as they have seal injection, or seal cooling, or the RCPs are tripped. This greatly reduces the likelihood of an RCP seal failure causing a Loss-of-Coolant Accident (LOCA).
- Offsite power is supplied from a 230 kV switchyard that has feeds from the grid and from three fossil plants onsite. CR-3 outputs to a separate 500 kV switchyard. Based on this, dependent loss of offsite power events occurring due to trip initiators is not considered a credible event.

- CR-3 maintains a diverse secondary cooling capability, including automatically actuated steam and diesel driven emergency feedwater pumps, a backup motor driven pump powered from the Engineered Safeguards (ES) bus, and a backup motor driven pump that is powered from normal offsite power or a dedicated diesel generator.

- CR-3 has three high head injection/makeup pumps each capable of providing adequate primary cooling via the pressurizer power-operated relief valve or pressurizer safeties at full Reactor Coolant System (RCS) pressure. The High Pressure Injection (HPI) pumps also have diverse support systems. Two of the pumps have backup cooling and one can be powered from either ES 4160 kV bus.

- CR-3 has separate safety-related service water systems for the decay heat removal system and nuclear services support for other systems. The nuclear services system also has a third non-safety related train that can cool normal loads.

- CR-3 has a dedicated chiller installed for 10CFR50 Appendix R (fire) considerations that is not dependent on service water.

The PSA inputs used for this application were generated using updated Individual Plant Examination (IPE) models developed in response to Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities," and associated supplements. The original development work was a level one Probabilistic Risk Assessment (PRA) study completed in 1987 (Crystal River Unit 3 Probabilistic Risk Assessment, Florida Power Corporation, Science Applications Intl. Corporation, July 1987), which was submitted to the NRC and reviewed by Argonne National Laboratory (NUREG/CR-5245). This study was subsequently updated for the Generic Letter 88-20 IPE submittal to include a level two containment analysis and an internal flooding analysis. The study was subject to reviews by the relevant CR-3 system engineers, and review of the event sequence analysis, quantification, and recovery analysis by the Nuclear Safety Supervisor at CR-3, a former Senior Reactor Operator.

Revisions to the models have been made to maintain the models consistent with plant design changes and operational changes. Current administrative controls include written procedures and review of all model changes, data updates, and risk assessments performed using PSA methods and 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, and advances in PSA technology. The PSA maintenance and update process is described in administrative procedure ADM-NGGC-0004, "Updates to PSA Models." Guidance to determine the need for a model update is provided in the procedure.

Computer programs that process PSA model inputs are verified and validated in accordance with administrative procedure CSP-NGGC-2505, "Software Quality Assurance and Configuration Control of Business Computer Systems." This procedure provides for software verification and validation to ensure the software meets the software requirement specifications and functional requirements.

Since the submittal of the IPE, there have been several significant plant design changes incorporated into the PSA model that have resulted in a reduction in the core damage frequency. A summary of significant model changes incorporated due to these plant changes includes the following:

- BEST added ("A" and "B" safeguards trains powered from separate transformers)
- FWP-7 with dedicated diesel generator MTDG-1 installed
- Appendix R chiller installed
- EFP-3 installed
- Low pressure injection suction valves changed to be normally open
- High pressure injection discharge throttle valves and cross-ties added
- Revision of emergency operating procedures reflected in human action probabilities

In addition to these plant changes, updates have been made to plant-specific data (through 1999) and initiating events data, as well as updates to the methods used for human reliability, common cause, internal flooding, and level two analyses.

The CR-3 PSA model and documentation was subjected to the industry peer certification review process in September 2001. The industry peer certification review was conducted by a diverse group of PSA engineers from other Babcock & Wilcox (B&W) plants, industry PSA consultants familiar with the B&W plant design, and a representative from the Institute of Nuclear Power Operations (INPO). The certification review covered all aspects of the PSA model and the administrative processes used to maintain and update the model. This review generated specific recommendations for model changes, as well as guidance for improvements to processes and methodologies used in the CR-3 PSA model, and enhancements to the documentation of the model and the administrative procedures used for model updates.

Following completion of this review, the CR-3 PSA model was revised to address each issue identified which affected the model. The significant changes identified included:

- Update of the plant-specific thermal-hydraulic analyses that provides the bases for accident sequences, system success criteria, and timing for operator actions.
- Revision of accident sequence logic for steam generator tube rupture (SGTR) and anticipated transient without scram (ATWS) mitigation.
- Development of an initiating event to address the loss of all raw water pumps (loss of ultimate heat sink).
- Update of the interfacing systems loss of coolant accident (ISLOCA) analyses.

- Update of the human reliability analysis including the dependency analysis for multiple operator action responses to an event, and
- Update of the level two analysis.

All peer review items which impact the PSA model have been addressed and are reflected in the PSA model used in this submittal.

At the time of the peer review, the level two model was not yet completed, and only a preliminary draft version, along with the original IPE level two results, were available for review. The level two model is now complete, and the findings identified from the peer certification review of the preliminary results and the IPE model have been addressed.

Subsequent to the completion of the RI-ISI effort at CR-3, but prior to this submittal, an update to the CR-3 PRA had been conducted. An evaluation of the impact of the CR-3 PRA update on the RI-ISI Program was performed. The review concluded that there is no impact on the results of the RI-ISI evaluations and therefore no change to the number, type, or locations of examinations.

2. PROPOSED ALTERNATIVE TO CURRENT ISI PROGRAMS

2.1 ASME Section XI

ASME Section XI Examination Categories B-F and B-J currently contain the requirements for the nondestructive examination (NDE) of Class 1 piping components. The alternative RI-ISI Program for piping is described in EPRI TR-112657. The RI-ISI Program will be substituted for the current program for Class 1 piping (Examination Categories B-F and B-J) 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. EPRI TR-112657 provides the requirements for defining the relationship between the RI-ISI Program and the remaining unaffected portions of ASME Section XI.

2.2 Augmented Programs

The following plant augmented inspection programs were considered during the RI-ISI application:

- A plant augmented inspection program is currently implemented at CR-3 that requires a volumetric examination of Category B-J HPI system piping ≥ 2 inches Nominal Pipe Size (NPS) and ≤ 4 inches NPS. This augmented volumetric examination is performed in addition to the Code required surface examination of these small-bore Class 1 HPI piping weld locations. The RI-ISI Program effectively subsumes this plant augmented inspection program requirement since each of the HPI element selections is subject to a volumetric examination per the RI-ISI process.
- The plant augmented inspection program for the high pressure injection and normal makeup nozzle thermal sleeves and nozzle knuckle transition regions is not affected or changed by the RI-ISI Program.

- The plant augmented inspection program pertaining to thermal stratification concerns in the Class 1 piping pressure boundary is subsumed by the RI-ISI Program.

- The plant augmented inspection program pertaining to Inconel I-600 components is not affected or changed by the RI-ISI Program.

3. RISK-INFORMED ISI PROCESS

The process used to develop the RI-ISI Program conformed to the methodology described in EPRI TR-112657 and consisted of the following steps:

- Scope Definition
- Consequence Evaluation
- Failure Potential Assessment
- Risk Characterization
- Element and NDE Selection
- Risk Impact Assessment
- Implementation Program
- Feedback Loop

A deviation to the EPRI RI-ISI methodology has been implemented in the failure potential assessment for CR-3. Table 3-16 of EPRI TR-112657 contains criteria for assessing the potential for thermal stratification, cycling and striping (TASCS). Key attributes for horizontal or slightly sloped piping greater than 1inch NPS include:

1. Potential exists for low flow in a pipe section connected to a component allowing mixing of hot and cold fluids, or
2. Potential exists for leakage flow past a valve, including in-leakage, out-leakage and cross-leakage allowing mixing of hot and cold fluids, or
3. Potential exists for convective heating in dead-ended pipe sections connected to a source of hot fluid, or
4. Potential exists for two phase (steam/water) flow, or
5. Potential exists for turbulent penetration into a relatively colder branch pipe connected to header piping containing hot fluid with turbulent flow,

AND

- $\Delta T > 50°F$,

AND

- Richardson Number > 4 (*this value predicts the potential buoyancy of a stratified flow)*

These criteria, based on meeting a high cycle fatigue endurance limit with the actual change in Temperature (ΔT) assumed equal to the greatest potential ΔT for the transient, will identify locations where stratification is likely to occur, but allows for no assessment of severity. As such, many locations will be identified as subject to TASCS where no significant potential for thermal fatigue exists. The critical attribute missing from the existing methodology that would allow consideration of fatigue severity is a criterion that addresses the potential for fluid cycling. The impact of this additional consideration on the existing TASCS susceptibility criteria is presented below.

- **Turbulent penetration TASCS**

  Turbulent penetration typically occurs in lines connected to piping containing hot flowing fluid. In the case of downward sloping lines that then turn horizontal, significant top-to-bottom cyclic ΔTs can develop in the horizontal sections if the horizontal section is less than about 25 pipe diameters from the reactor coolant piping. Therefore, TASCS is considered for this configuration.

  For upward sloping branch lines connected to the hot fluid source that turn horizontal or in horizontal branch lines, natural convective effects combined with effects of turbulence penetration will keep the line filled with hot water. If there is no potential for in-leakage towards the hot fluid source from the outboard end of the line, this will result in a well-mixed fluid condition where significant top-to-bottom ΔTs will not occur. Therefore TASCS is not considered for these configurations. Even in fairly long lines, where some heat loss from the outside of the piping will tend to occur and some fluid stratification may be present, there is no significant potential for cycling as has been observed for the in-leakage case. The effect of TASCS will not be significant under these conditions and can be neglected.

- **Low flow TASCS**

  In some situations, the transient startup of a system (e.g., Residual Heat Removal (RHR) suction piping) creates the potential for fluid stratification as flow is established. In cases where no cold fluid source exists, the hot flowing fluid will fairly rapidly displace the cold fluid in stagnant lines, while fluid mixing will occur in the piping further removed from the hot source and stratified conditions will exist only briefly as the line fills with hot fluid. As such, since the situation is transient in nature, it can be assumed that the criteria for thermal transients (TT) will govern.

- **Valve leakage TASCS**

  Sometimes a very small leakage flow of hot water can occur outward past a valve into a line that is relatively colder, creating a significant temperature difference. However, since this is generally a "steady-state" phenomenon with no potential for cyclic temperature changes, the effect of TASCS is not significant and can be neglected.

- **Convection heating TASCS**

  Similarly, there exists the potential for heat transfer across a valve to an isolated section beyond the valve, resulting in fluid stratification due to natural convection. However,

since there is no potential for cyclic temperature changes in this case, the effect of TASCS is not significant and can be neglected.

In summary, these additional considerations for determining the potential for thermal fatigue as a result of the effects of TASCS provide an allowance for the consideration of cycle severity. The above criteria have previously been submitted by EPRI for generic approval (Letters dated February 28, 2001 and March 28, 2001, from P.J. O'Regan (EPRI) to Dr. B. Sheron (USNRC), "Extension of Risk-Informed Inservice Inspection Methodology"). The methodology used in the CR-3 RI-ISI application for assessing TASCS potential conforms to the updated criteria described in the EPRI letter to NRC dated March 28, 2001. Final materials reliability program (MRP) guidance on the subject of TASCS will be incorporated into the CR-3 RI-ISI application if warranted. It should be noted that the NRC has granted approval for RI-ISI relief requests incorporating these TASCS criteria at several facilities, including Comanche Peak (SER dated September 28, 2001) and South Texas Project (SER dated March 5, 2002).

3.1 Scope of Program

The systems included in the RI-ISI Program are provided in Table 3.1. The piping and instrumentation diagrams and additional plant information including the existing plant ISI Program were used to define the Class 1 piping system boundaries.

3.2 Consequence Evaluation

The consequence(s) of pressure boundary failures were evaluated and ranked based on their impact on core damage and containment performance (i.e., isolation, bypass and large early release). The consequence evaluation included an assessment of shutdown and external events. The impact on these measures, due to both direct and indirect effects, was considered using the guidance provided in EPRI TR-112657.

3.3 Failure Potential Assessment

Failure potential estimates were generated utilizing industry failure history, plant specific failure history, and other relevant information. These failure estimates were determined using the guidance provided in EPRI TR-112657, with the exception of the previously stated deviation.

Table 3.3 summarizes the failure potential assessment by system for each degradation mechanism that was identified as potentially operative.

3.4 Risk Characterization

In the preceding steps, each run of piping within the scope of the program was evaluated to determine its impact on core damage and containment performance (i.e., isolation, bypass and large, early release) as well as its potential for failure. Given the results of these steps, piping segments are then defined as continuous runs of piping potentially susceptible to the same type(s) of degradation and whose failure will result in similar consequence(s). Segments are then ranked based upon their risk significance as defined in EPRI TR-112657.

The results of these calculations are presented in Table 3.4.

3.5 Element and NDE Selection

In general, EPRI TR-112657 requires that 25% of the locations in the high risk region and 10% of the locations in the medium risk region be selected for inspection using appropriate NDE methods tailored to the applicable degradation mechanism. In addition, per Section 3.6.4.2 of EPRI TR-112657, if the percentage of Class 1 piping locations selected for examination falls substantially below 10%, then the basis for selection needs to be investigated. For CR-3, the percentage of Class 1 piping welds selected strictly for RI-ISI purposes was 10.4%.

A brief summary is provided in the following table, and the results of the selections are presented in Table 3.5. Section 4 of EPRI TR-112657 was used as guidance in determining the examination requirements for these locations.

Totals	Description
539	Class 1 Piping Welds(1)
56	RI-ISI Program Selections

Notes

1. Includes all Category B-F and B-J locations. All in-scope piping components, regardless of risk classification, will continue to receive Code required pressure testing, as part of the current ASME Section XI Program. VT-2 visual examinations are scheduled in accordance with the station's pressure test program that remains unaffected by the RI-ISI Program.

3.5.1 Additional Examinations

The RI-ISI 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 in the segment or additional segments are subject to the same root cause conditions. Additional examinations will be performed on those elements with the same root cause conditions or degradation mechanisms. The additional examinations will include high risk significant elements and medium risk significant elements, if needed, up to a number equivalent to the number of elements required to be inspected on the segment or segments during the current outage. If unacceptable flaws or relevant conditions are again found similar to the initial problem, the remaining elements identified as susceptible will be examined during the current outage. No additional examinations will be performed if there are no additional elements identified as being susceptible to the same root cause conditions.

3.5.2 Program Relief Requests

An attempt has been made to select RI-ISI locations for examination such that a minimum of >90% coverage (i.e., Code Case N-460 criteria) is attainable. However, some limitations will not be known until the examination is performed, since some locations may be examined for the first time by the specified techniques.

In instances where locations are found at the time of the examination that do not meet the >90% coverage requirement, the process outlined in EPRI TR-112657 will be followed.

The following relief request can be withdrawn for the reasons provided below with all other relief requests remaining in place.

Relief Request	Relief Request Description
98-001-II[1]	Pertains to the volumetric examination of the core flood nozzles outer surface during the performance of the automated reactor vessel examination from the inside diameter.

Notes

1. The core flood nozzle locations selected for RI-ISI examination are subject to a volumetric examination of only the inner 1/3rd volume. Since an outer surface examination is not required for RI-ISI purposes, Relief Request 98-001-II can be withdrawn.

3.6 Risk Impact Assessment

The RI-ISI Program has been conducted in accordance with Regulatory Guide 1.174 and the requirements of EPRI TR-112657, and the risk from implementation of this program is expected to remain neutral or decrease when compared to that estimated from current requirements.

This evaluation identified the allocation of segments into High, Medium, and Low risk regions of the EPRI TR-112657 and ASME Code Case N-578 risk ranking matrix, and then determined for each of these risk classes what inspection changes are proposed for each of the locations in each segment. The changes include changing the number and location of inspections within the segment and in many cases improving the effectiveness of the inspection to account for the findings of the RI-ISI degradation mechanism assessment. For example, for locations subject to thermal fatigue, examinations will be conducted on an expanded volume and will be focused to enhance the probability of detection (POD) during the inspection process.

3.6.1 Quantitative Analysis

Limits are imposed by the EPRI methodology to ensure that the change in risk of implementing the RI-ISI Program meets the requirements of Regulatory Guides 1.174 and 1.178. The EPRI criterion requires that the cumulative change in core

damage frequency (CDF) and large early release frequency (LERF) be less than 1E-07 and 1E-08 per year per system, respectively.

CR-3 conducted a risk impact analysis per the requirements of Section 3.7 of EPRI TR-112657. The analysis estimates the net change in risk due to the positive and negative influence of adding and removing locations from the inspection program. A risk quantification was performed using the "Simplified Risk Quantification Method" described in Section 3.7 of EPRI TR-112657. The conditional core damage probability (CCDP) and conditional large early release probability (CLERP) used for high consequence category segments was based on the highest evaluated CCDP (3.5E-02) and CLERP (1.6E-05), whereas, for medium consequence category segments, bounding estimates of CCDP (1E-04) and CLERP (1E-05) were used. The likelihood of pressure boundary failure (PBF) is determined by the presence of different degradation mechanisms and the rank is based on the relative failure probability. The basic likelihood of PBF for a piping location with no degradation mechanism present is given as x_o and is expected to have a value less than 1E-08. Piping locations identified as medium failure potential have a likelihood of $20x_o$. These PBF likelihoods are consistent with References 9 and 14 of EPRI TR-112657. In addition, the analysis was performed both with and without taking credit for enhanced inspection effectiveness due to an increased POD from application of the RI-ISI approach.

Table 3.6-1 presents a summary of the RI-ISI Program versus 1989 ASME Section XI Code Edition program requirements and identifies, on a per system basis, each applicable risk category. It should be noted that no degradation mechanisms (e.g., Flow Accelerated Corrosion (FAC)) managed by plant augmented inspection programs exist in the scope of this Class 1 piping application for CR-3. As such, no adjustments were required in the performance of the quantitative analysis to account for the impact of plant augmented inspection program managed degradation mechanisms on the risk ranking.

As indicated in the following table, this evaluation has demonstrated that unacceptable risk impacts will not occur from implementation of the RI-ISI Program, and satisfies the acceptance criteria of Regulatory Guide 1.174 and EPRI TR-112657.

Risk Impact Results

System(1)	$\Delta Risk_{CDF}$		$\Delta Risk_{LERF}$	
	w/ POD	w/o POD	w/ POD	w/o POD
RC	-2.1E-08	1.1E-09	-9.8E-12	4.8E-13
DH	-6.3E-09	-3.5E-09	-2.9E-12	-1.6E-12
MU	-2.8E-08	-1.6E-08	-1.5E-11	-9.5E-12
CF	-2.3E-09	5.3E-10	-1.0E-12	2.4E-13
Total	**-5.8E-08**	**-1.8E-08**	**-2.9E-11**	**-1.0E-11**

Note
1. Systems are described in Table 3.1.

3.6.2 Defense-in-Depth

The intent of the inspections mandated by ASME Section XI for piping welds is to identify conditions such as flaws or indications that may be precursors to leaks or ruptures in a system's pressure boundary. Currently, the process for picking inspection locations is based upon structural discontinuity and stress analysis results. As depicted in ASME White Paper 92-01-01 Rev. 1, "Evaluation of Inservice Inspection Requirements for Class 1, Category B-J Pressure Retaining Welds," this method has been ineffective in identifying leaks or failures. EPRI TR-112657 and Code Case N-578 provide a more robust selection process founded on actual service experience with nuclear plant piping failure data.

This process has two key independent ingredients, that is, a determination of each location's susceptibility to degradation and secondly, an independent assessment of the consequence of the piping failure. These two ingredients assure defense in depth is maintained. First, by evaluating a location's susceptibility to degradation, the likelihood of finding flaws or indications that may be precursors to leak or ruptures is increased. Secondly, the consequence assessment effort has a single failure criterion. As such, no matter how unlikely a failure scenario is, it is ranked High in the consequence assessment, and at worst Medium in the risk assessment (i.e., Risk Category 4), if as a result of the failure, there is no mitigative equipment available to respond to the event. In addition, the consequence assessment takes into account equipment reliability and less credit is given to less reliable equipment.

All locations within the Class 1 pressure boundary will continue to receive a system pressure test and visual VT-2 examination as currently required by the Code regardless of its risk classification.

4. IMPLEMENTATION AND MONITORING PROGRAM

Upon approval of the RI-ISI Program, procedures that comply with the guidelines described in EPRI TR-112657 will be prepared to implement and monitor the program. The new program will be integrated into the third inservice inspection interval. No changes to the Improved Technical Specifications or Final Safety Analysis Report are necessary for program implementation.

The applicable aspects of the ASME Code not affected by this change will 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 will be retained and modified to address the RI-ISI process, as appropriate.

The CR-3 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. In addition, significant changes may require more frequent adjustment as directed by NRC Bulletin or Generic Letter requirements, or by industry and plant specific feedback.

5. PROPOSED ISI PROGRAM PLAN CHANGE

A comparison between the RI-ISI Program and ASME Section XI 1989 Code Edition program requirements for in-scope piping is provided in Tables 5-1 and 5-2. Table 5-1 provides a summary comparison by risk region. Table 5-2 provides the same comparison information, but in a more detailed manner by risk category, similar to the format used in Table 3.6-1.

CR-3 intends to start implementing the RI-ISI Program during the plant's Refueling Outage 14 (RFO14) scheduled for Fall 2005. Beginning with RFO14, inspection locations selected per the RI-ISI process will replace those formerly selected per ASME Section XI criteria. At the midpoint of the second period, 35.4% of the piping weld examinations required by ASME Section XI, IWB-2412, Inspection Program B have been completed thus far in the third ISI interval for Examination Categories B-F and B-J. To ensure the performance of 100% of the required examinations during the current ten-year ISI interval, 64.6% of the inspection locations selected for examination per the RI-ISI process will be examined over the remainder of the third ISI interval.

Subsequent ISI intervals will implement 100% of the inspection locations selected for examination per the RI-ISI Program. Examinations shall be performed such that the period percentage requirements of ASME Section XI, paragraph IWB-2412 are met.

6. REFERENCES/DOCUMENTATION

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

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

Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis"

Regulatory Guide 1.178, "An Approach for Plant-Specific Risk-Informed Decisionmaking for Inservice Inspection of Piping"

Supporting Onsite Documentation

Risk-Informed Inservice Inspection, Consequence Evaluation of Class 1, Crystal River 3, Revision 2

EPRI-156-340, Degradation Mechanism Evaluation for the Class 1 (Category B-J/B-F) Piping at Crystal River Unit 3, Revision 2

Engineering Change 55725R0, CR-3 Service History and Susceptibility Review

EPRI-156-341, Crystal River 3 Risk Ranking, Revision 0

EPRI-156-342, Minutes of the Element Selection Meeting for the RI-ISI Project at Crystal River Unit 3 (CR-3), Revision 0

Crystal River Unit 3, RI-ISI Risk Impact Analysis, Revision 0

"Evaluation of the impact of the CR3 PRA Update on the RI-ISI Program, Revision 0, October 2, 2004"

Table 3.1 System Selection and Segment / Element Definition		
System Description	**Number of Segments**	**Number of Elements**
RC – Reactor Coolant System	59	240
DH – Decay Heat Removal System	4	16
MU – Makeup and Purification System	21	253
CF – Core Flood System	8	30
Totals	**92**	**539**

Table 3.3

Failure Potential Assessment Summary

System[1]	Thermal Fatigue		Stress Corrosion Cracking				Localized Corrosion			Flow Sensitive	
	TASCS	TT	IGSCC	TGSCC	ECSCC	PWSCC	MIC	PIT	CC	E-C	FAC
RC	X	X				X					
DH	X										
MU	X	X	X								
CF		X				X					

Note

1. Systems are described in Table 3.1.

Table 3.4

Number of Segments by Risk Category With and Without Impact of FAC(1)

System(2)	High Risk Region						Medium Risk Region				Low Risk Region			
	Category 1		Category 2		Category 3		Category 4		Category 5		Category 6		Category 7	
	With	Without	With	Without	With	Without	With	Without	With	Without	With	Without	With	Without
RC			20	20			32	32			1	1	6	6
DH			1	1			2	2			1	1		
MU			4	4			6	6	4	4	4	4	3	3
CF			4	4			2	2			2	2		
Total			29	29			42	42	4	4	8	8	9	9

Notes

1. The Flow Accelerated Corrosion (FAC) Program is not applicable for Class 1 piping at CR-3. As such, the FAC Program has no impact on the segment counts shown in the table. The table format and reference to the FAC Program has been retained solely for uniformity purposes with other RI-ISI application template submittals.
2. Systems are described in Table 3.1.

Table 3.5

Number of Elements Selected for Inspection by Risk Category Excluding Impact of FAC[1]

System[2]	High Risk Region						Medium Risk Region				Low Risk Region			
	Category 1		Category 2		Category 3		Category 4		Category 5		Category 6		Category 7	
	Total	Selected	Total	Selected	Total	Selected	Total	Selected	Total	Selected	Total	Selected	Total	Selected
RC			42	11			183	19			2	0	13	0
DH			2	1			7	1			7	0		
MU			13	4			140	14	13	2	84	0	3	0
CF			4	2			11	2			15	0		
Total			61	18			341	36	13	2	108	0	16	0

Notes

1. The Flow Accelerated Corrosion (FAC) Program is not applicable for Class 1 piping at CR-3. As such, the FAC Program has no impact on the element counts shown in the table. The table format and reference to the FAC Program has been retained solely for uniformity purposes with other RI-ISI application template submittals.
2. *Systems are described* in Table 3.1.

Table 3.6-1
Risk Impact Analysis Results

System(1)	Category	Consequence Rank	Failure Potential		Inspections			CDF Impact(3)		LERF Impact(3)	
			DMs	Rank	SXI(2)	RI-ISI	Delta	w/ POD	w/o POD	w/ POD	w/o POD
RC	2	High	TASCS, TT, PWSCC	Medium	2	1	-1	3.5E-09	3.5E-09	1.6E-12	1.6E-12
RC	2	High	TASCS, TT	Medium	6	5	-1	-1.9E-08	3.5E-09	-8.6E-12	1.6E-12
RC	2	High	TASCS	Medium	0	0	0	no change	no change	no change	no change
RC	2	High	TT	Medium	0	0	0	no change	no change	no change	no change
RC	2	High	PWSCC	Medium	3	5	2	-7.0E-09	-7.0E-09	-3.2E-12	-3.2E-12
RC	4	High	None	Low	25	19	-6	1.1E-09	1.1E-09	4.8E-13	4.8E-13
RC	6a	Medium	None	Low	0	0	0	no change	no change	no change	no change
RC	7a	Low	None	Low	0	0	0	no change	no change	no change	no change
RC Total								**-2.1E-08**	**1.1E-09**	**-9.8E-12**	**4.8E-13**
DH	2	High	TASCS	Medium	0	1	1	-6.3E-09	-3.5E-09	-2.9E-12	-1.6E-12
DH	4	High	None	Low	1	1	0	no change	no change	no change	no change
DH	6a	Medium	None	Low	3	0	-3	negligible	negligible	negligible	negligible
DH Total								**-6.3E-09**	**-3.5E-09**	**-2.9E-12**	**-1.6E-12**
MU	2	High	TASCS, TT	Medium	0	2	2	-1.3E-08	-7.0E-09	-5.8E-12	-3.2E-12
MU	2	High	TT	Medium	0	2	2	-1.3E-08	-7.0E-09	-5.8E-12	-3.2E-12
MU	4	High	None	Low	0	14	14	-2.5E-09	-2.5E-09	-1.1E-12	-1.1E-12
MU	5a	Medium	TT, IGSCC	Medium	0	1	1	-1.0E-11	-1.0E-11	-1.0E-12	-1.0E-12
MU	5a	Medium	TT	Medium	0	1	1	-1.8E-11	-1.0E-11	-1.8E-12	-1.0E-12
MU	6a	Medium	None	Low	0	0	0	no change	no change	no change	no change
MU	7a	Low	None	Low	0	0	0	no change	no change	no change	no change
MU Total								**-2.8E-08**	**-1.6E-08**	**-1.5E-11**	**-9.5E-12**

Table 3.6-1 (Cont'd)
Risk Impact Analysis Results

System[1]	Category	Consequence Rank	Failure Potential		Inspections			CDF Impact[3]		LERF Impact[3]	
			DMs	Rank	SXI[2]	RI-ISI	Delta	w/ POD	w/o POD	w/ POD	w/o POD
CF	2	High	TT, PWSCC	Medium	2	1	-1	3.5E-09	3.5E-09	1.6E-12	1.6E-12
CF	2	High	TT	Medium	0	1	1	-6.3E-09	-3.5E-09	-2.9E-12	-1.6E-12
CF	4	High	None	Low	5	2	-3	5.3E-10	5.3E-10	2.4E-13	2.4E-13
CF	6a	Medium	None	Low	5	0	-5	negligible	negligible	negligible	negligible
CF Total								**-2.3E-09**	**5.3E-10**	**-1.0E-12**	**2.4E-13**
Grand Total								**-5.8E-08**	**-1.8E-08**	**-2.9E-11**	**-1.0E-11**

Notes

1. Systems are described in Table 3.1.
2. Only those ASME Section XI Code inspection locations that received a volumetric examination in addition to a surface examination are included in the count. Inspection locations previously subjected to a surface examination only were not considered in accordance with Section 3.7.1 of EPRI TR-112657.
3. Per Section 3.7.1 of EPRI TR-112657, the contribution of low risk categories 6 and 7 need not be considered in assessing the change in risk. They are excluded from analysis because they have an insignificant impact on risk. Hence, the word "negligible" is given in these cases in lieu of values for CDF and LERF Impact. For those cases in high, medium or low risk region piping where no impact to CDF or LERF exists, "no change" is listed.

Table 5-1

Inspection Location Selection Comparison Between ASME Section XI Code and EPRI TR-112657 by Risk Region

System[1]	Code Category[2]	High Risk Region					Medium Risk Region					Low Risk Region				
		Weld Count	Section XI		EPRI TR-112657		Weld Count	Section XI		EPRI TR-112657		Weld Count	Section XI		EPRI TR-112657	
			Vol/Sur	Sur Only	RI-ISI	Other[3]		Vol/Sur	Sur Only	RI-ISI	Other[3]		Vol/Sur	Sur Only	RI-ISI	Other[3]
RC	B-F	2	2	0	2											
	B-J^{DMWs}	7	4	3	4		15	8	7	0						
	B-J	33	5	6	5		168	17	20	19		15	0	2	0	
DH	B-J	2	0	0	1		7	1	0	1		7	3	0	0	
MU	B-J^{DMWs}	4	0	4	0		1	0	1	0						
	B-J	9	0	9	4		152	0	37	16		87	0	20	0	
CF	B-F	2	2	0	1											
	B-J	2	0	0	1		11	5	0	2		15	5	0	0	
Total	B-F	4	4	0	3											
	B-J^{DMWs}	11	4	7	4		16	8	8	0						
	B-J	46	5	15	11		338	23	57	38		123	8	22	0	

Notes

1. Systems are described in Table 3.1.
2. CR-3 is taking guidance from a later Code requirement pertaining to the classification of piping dissimilar metal welds. Starting with the 1989 Addenda, piping dissimilar metal welds (DMWs) are classified as Category B-J instead of B-F. Category B-F pertains only to vessel dissimilar metal welds, which for CR-3, consists of the pressurizer surge and spray nozzles, and two core flood nozzle connections to the reactor vessel.
3. The column labeled "Other" is generally used to identify plant augmented inspection program locations credited per Section 3.6.5 of EPRI TR-112657. The EPRI methodology allows plant augmented inspection program locations to be credited if the inspection locations selected strictly for RI-ISI purposes produce less than a 10% sampling of the overall Class 1 weld population. This option was not applicable for the CR-3 RI-ISI application. The "Other" column has been retained in this table solely for uniformity purposes with the other RI-ISI application template submittals.

Table 5-2
Inspection Location Selection Comparison Between ASME Section XI Code and EPRI TR-112657 by Risk Category

System[1]	Risk		Consequence Rank	Failure Potential		Code Category[2]	Weld Count	Section XI		EPRI TR-112657	
	Category	Rank		DMs	Rank			Vol/Sur	Sur Only	RI-ISI	Other[3]
RC	2	High	High	TASCS, TT, PWSCC	Medium	B-J^{DMWs}	2	2	0	1	
RC	2	High	High	TASCS, TT	Medium	B-J^{DMWs}	1	1	0	0	
						B-J	19	5	2	5	
RC	2	High	High	TASCS	Medium	B-J	6	0	1	0	
RC	2	High	High	TT	Medium	B-J	8	0	3	0	
RC	2	High	High	PWSCC	Medium	B-F	2	2	0	2	
						B-J^{DMWs}	4	1	3	3	
RC	4	Medium	High	None	Low	B-J^{DMWs}	15	8	7	0	
						B-J	168	17	20	19	
RC	6a	Low	Medium	None	Low	B-J	2	0	1	0	
RC	7a	Low	Low	None	Low	B-J	13	0	1	0	
DH	2	High	High	TASCS	Medium	B-J	2	0	0	1	
DH	4	Medium	High	None	Low	B-J	7	1	0	1	
DH	6a	Low	Medium	None	Low	B-J	7	3	0	0	
MU	2	High	High	TASCS, TT	Medium	B-J^{DMWs}	3	0	3	0	
						B-J	7	0	7	2	
MU	2	High	High	TT	Medium	B-J^{DMWs}	1	0	1	0	
						B-J	2	0	2	2	
MU	4	Medium	High	None	Low	B-J^{DMWs}	1	0	1	0	
						B-J	139	0	32	14	
MU	5a	Medium	Medium	TT, IGSCC	Medium	B-J	9	0	1	1	
MU	5a	Medium	Medium	TT	Medium	B-J	4	0	4	1	
MU	6a	Low	Medium	None	Low	B-J	84	0	20	0	
MU	7a	Low	Low	None	Low	B-J	3	0	0	0	

Table 5-2 (Cont'd)
Inspection Location Selection Comparison Between ASME Section XI Code and EPRI TR-112657 by Risk Category

System[1]	Risk		Consequence Rank	Failure Potential		Code Category[2]	Weld Count	Section XI		EPRI TR-112657	
	Category	Rank		DMs	Rank			Vol/Sur	Sur Only	RI-ISI	Other[3]
CF	2	High	High	TT, PWSCC	Medium	B-F	2	2	0	1	
CF	2	High	High	TT	Medium	B-J	2	0	0	1	
CF	4	Medium	High	None	Low	B-J	11	5	0	2	
CF	6a	Low	Medium	None	Low	B-J	15	5	0	0	

Notes

1. Systems are described in Table 3.1.
2. CR-3 is taking guidance from a later Code requirement pertaining to the classification of piping dissimilar metal welds. Starting with the 1989 Addenda, piping dissimilar metal welds (DMWs) are classified as Category B-J instead of B-F. Category B-F pertains only to vessel dissimilar metal welds, which for CR-3, consists of the pressurizer surge and spray nozzles, and two core flood nozzle connections to the reactor vessel.
3. The column labeled "Other" is generally used to identify plant augmented inspection program locations credited per Section 3.6.5 of EPRI TR-112657. The EPRI methodology allows plant augmented inspection program locations to be credited if the inspection locations selected strictly for RI-ISI purposes produce less than a 10% sampling of the overall Class 1 weld population. This option was not applicable for the CR-3 RI-ISI application. The "Other" column has been retained in this table solely for uniformity purposes with the other RI-ISI application template submittals.

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72

ATTACHMENT B

List of Regulatory Commitments

List of Regulatory Commitments

The following table identifies those actions committed to by Florida Power Corporation, doing business as Progress Energy Florida, Inc., in this document. Any other actions discussed in the submittal represent intended or planned actions by Florida Power Corporation. They are described to the NRC for the NRC's information and are not regulatory commitments. Please notify the Supervisor, Licensing and Regulatory Programs, of any questions regarding this document or any associated regulatory commitments.

ID Number	Commitment	Commitment Date
3F1104-02-1	The methodology used in the CR-3 RI-ISI application for assessing TASCS potential conforms to the updated criteria described in the EPRI letter to NRC dated March 28, 2001. Final materials reliability program (MRP) guidance on the subject of TASCS will be incorporated into the CR-3 RI-ISI application if warranted.	Once final guidance is approved by the NRC
3F1104-02-2	To ensure the performance of 100% of the required examinations during the current ten-year ISI interval, 64.6% of the inspection locations selected for examination per the RI-ISI process will be examined over the remainder of the third ISI interval.	Over the remainder of the third ISI Interval