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U S Nuclear Regulatory Commission
ATTN: Document Control Desk
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Prairie Island Nuclear Generating Plant Units 1 and 2
Dockets 50-282 and 50-306
License Nos. DPR-42 and DPR-60

Relief Request to Implement Risk-Informed Inservice Inspection (ISI) Scheduling for the Fourth 10-Year Inspection Interval for Prairie Island Units 1 and 2

Prairie Island submitted its Inservice Inspection Plan for the Fourth 10-Year Interval on June 21, 2004. The interval runs from December 21, 2004 through December 20, 2014, for both Unit 1 and Unit 2.

Enclosed is a relief request to allow a risk-informed weld selection and examination volumes as an alternative to the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI. The alternative follows ASME Code Case N-578, "Risk-Informed Requirements for Class 1, 2, and 3 Piping, Method B" and Electric Power Research Institute Topical Report TR-112657 Rev. B-A, "Revised Risk-Informed Inservice Inspection Evaluation Procedure."

We are requesting relief pursuant to 10 CFR Part 50, Section 50.55a(a)(3)(i) because the proposed alternative would provide an acceptable level of quality and safety. Nuclear Management Company requests approval by December 31, 2005. A similar alternative was approved for the Point Beach Nuclear Plant, Units 1 and 2, by NRC letter dated July 2, 2003.

Summary of Commitments

This letter contains no new commitments and no revisions to existing commitments.

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Enclosures(2)

cc: Administrator, Region III, USNRC
Project Manager, Prairie Island, USNRC
Resident Inspector, Prairie Island, USNRC

ENCLOSURE 1

Prairie Island Unit 1 – RELIEF REQUEST NUMBER 1-RR-4-5: (Rev. 0)
Prairie Island Unit 2 – RELIEF REQUEST NUMBER 2-RR-4-5: (Rev. 0)

2 pages follow

Prairie Island Unit 1 - RELIEF REQUEST NUMBER 1-RR-4-5: (Rev. 0)
Prairie Island Unit 2 - RELIEF REQUEST NUMBER 2-RR-4-5: (Rev. 0)

Risk Informed Examination of Class 1 and 2 Piping Welds
(Code Case N-578 and EPRI TR-112657)

SYSTEM/COMPONENT (S) FOR WHICH RELIEF REQUEST WILL BE USED

Code Class:	Class 1 and 2
Reference:	ASME, Section XI, 1998 Edition with Addenda through 2000, Tables IWB-2500-1 and IWC-2500-1 ASME Section XI Code Case N-578
Examination Categories:	B-F, B-J, C-F-1, C-F-2
Item Number:	B5.10, B5.20, B5.40, B5.50, B5.70, B9.11, B9.21, B9.31, B9.32, B9.40, C5.11, C5.21, C5.30, C5.41, C5.51, C5.61, C5.70, C5.81
Description:	All pressure retaining piping welds
Component Numbers:	See attached Risk-Informed Inservice Inspection Plan, Prairie Island Nuclear Generating Plant, Units 1 and 2, Revision 0

CODE REQUIREMENTS:

The 1998 Edition of Section XI with Addenda through 2000, IWB-2500(a) states, "components shall be examined and tested as specified in Table IWB-2500-1. The method of examination for the components and parts of the pressure retaining boundaries shall comply with those tabulated in Table IWB-2500-1 except where alternate examination methods are used that meet the requirements of IWA-2240."

Table IWB-2500-1, Categories B-F and B-J requires 100% and 25% respectively of the total number of non-exempt welds.

The 1998 Edition of Section XI with Addenda through 2000, IWC-2500(a) states, "components shall be examined and pressure tested as specified in Table IWC-2500-1. The method of examination for the components and parts of the pressure retaining boundaries shall comply with those tabulated in Table IWC-2500-1 except where alternate examination methods are used that meet the requirements of IWA-2240."

Table IWC-2500-1, Categories C-F-1 and C-F-2 requires 7.5% of the total number of non-exempt and exempt welds, but not less than 28 welds.

In addition, Tables IWB-2500-1 and IWC-2500-1 reference figures that convey the examination volume for each configuration that could be encountered.

PROPOSED ALTERNATIVE:

As an alternative to existing ASME Section XI requirements for piping weld selection and examination volumes, Prairie Island Nuclear Generating Plant (PINGP) will implement the alternative methods as specified in Code Case N-578 and EPRI TR-112657B-A. "Risk-Informed Inservice Inspection Program Plan, Prairie Island Nuclear Generating Plant, Units 1 and 2, Revision 0 details the methodology for application of this alternative.

BASIS FOR RELIEF:

The scope for the ASME Section XI ISI Programs is largely based on deterministic results contained in design stress reports. These reports are normally very conservative and may not be an accurate representation of failure potential. Since the stress reports for PINGP do not contain all the information required to select welds in accordance with the later editions of ASME Section XI, PINGP has been utilizing the alternative selection methodology of the 1974 Edition of Section XI with addenda through summer 1975 for Category B-J welds as allowed by 10CFR50.55a(b)(2)(ii). Industry service experience has shown that piping weld failures are due to either corrosion or fatigue and typically occur in areas not included in the plant's ISI program. Consequently, nuclear plants are devoting significant resources to inspection programs that provide minimum benefit.

As an alternative, significant industry attention has been devoted to the risk-informed selection criteria in order to determine a more appropriate scope for ISI Programs at nuclear power plants. EPRI studies indicate that the application of Risk Informed techniques will allow operating plants to reduce the examination scope of current ISI Programs by as much as 60% to 80%, significantly reduce costs and radiation exposure, and continue to maintain an equivalent or better safety level as current ASME Section XI selection criteria.

PINGP has reviewed the EPRI Methodology as documented in the NRC approved Topical Report TR-112657B-A and referenced in Code Case N-578. Utilizing this methodology for the selection and subsequent examination of PINGP Class 1 and 2 piping welds will provide an acceptable level of quality and safety.

IMPLEMENTATION SCHEDULE:

Relief is requested for the 4th 10 year interval of the Inservice Inspection Program for Prairie Island Units 1 and 2.

REFERENCE:

By letter dated July 2, 2003, the NRC Staff authorized similar relief to Point Beach Nuclear Plant Units 1 and 2 (TAC Nos. MB5553 and MB5554).

ENCLOSURE 2

Risk-Informed Inservice Inspection Program Plan
Prairie Island Nuclear Generating Plant
Units 1 and 2 – Revision 0

35 pages follow

RISK-INFORMED INSERVICE INSPECTION PROGRAM PLAN

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

UNITS 1 AND 2 - REVISION 0

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1. INTRODUCTION

The Prairie Island Nuclear Generating Plant (PINGP) is currently nearing the end of its 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. PINGP plans to implement a risk-informed inservice inspection (RI-ISI) program during the fourth inservice inspection interval, which begins December 21, 2004 for both units. The ASME Section XI Code used during the third interval was the 1989 Edition. PINGP is using the 1998 Edition of Section XI with addenda through 2000 for the fourth inservice inspection interval.

The objective of this submittal is to request the use of a risk-informed process for the inservice inspection of Class 1 and 2 piping. The RI-ISI process used in this submittal is described in Electric Power Research Institute (EPRI) Topical Report (TR) 112657 Rev. B-A "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 Inservice Inspection of Piping". Further information is provided in Section 3.6.2 relative to defense-in-depth.

1.2 PSA Quality

The Prairie Island Level 1 and Level 2 Probabilistic Risk Assessment (PRA) results based on the Revision 1.2 update were used to evaluate the consequences of pipe ruptures for the RI-ISI assessment during power operation. A summary of the Level 1 and Level 2 results (in terms of dominant contributors to Core Damage Frequency and Large Early Release Frequency) are discussed below.

Summary of the PRA Level 1 Results: The Prairie Island Level 1 PRA model calculated a Core Damage Frequency (CDF) for internal events of 2.20E-5/year for each PINGP unit. The dominant accident sequences by initiating events are:

- Loss of Offsite Power (LOOP) including station blackout (SBO) (23.9%)
- Loss of Coolant Accidents (LOCA) (23.8%)
- Internal Flooding (22.5%)
- Steam Generator Tube Rupture (SGTR) (14.8%)
- Transients excluding LOOP (15.0%)

Summary of the PRA Level 2 Results: The Prairie Island Level 2 calculated a Large Early Release Frequency (LERF) of 6.9E-7/year for each PINGP unit, resulting in a LERF/CDF ratio of 0.031. The dominant contributors to LERF are:

- SGTR (87%)
- Intersystem LOCA (13%)
- Core damage sequence followed by failure of containment isolation (0.2%)

PRA MODEL HISTORY:

IPE Results (Level 1 and Level 2, Revision 0)

The first full-scope PRA analysis done for the Prairie Island Nuclear Generating Plant was performed to satisfy the Individual Plant Examination (IPE) requirements, and was completed in February 1994. This was a study to determine vulnerabilities to severe accidents from at-power operation. It was based on a Level 1 and Level 2 PRA model performed for Unit 1. Unit 2 vulnerabilities were qualitatively evaluated based on the Unit 1 results and consideration of asymmetries in plant design and operation that exist between the units. The study found no vulnerabilities to severe accidents at the PINGP. This PRA study is now considered to be Revision 0 of the Level 1 and 2 PRA models.

Level 1 and Level 2, Revision 1.0

Revision 1.0 of the Unit 1, Level 1 PRA model was completed in 1996. In addition to incorporating fault tree models for a few more balance-of-plant systems (for example, the non-safeguards station air system and the steam dump and circulating water systems), this update included modeling for a number of significant changes to the plant safeguards electrical systems that were not yet installed at the time of the IPE submittal. Examples include elimination of sub-fed 480V MCCs, division of the two Unit 1 safeguards 480 V AC buses into four buses and relocation of those buses within the plant, and significant reliability upgrades for the DC power system. Component failure and unavailability data for six key systems were updated for the period 1986 through 1995, as were the initiating event frequencies. LOCA frequencies were reanalyzed to make them more plant-specific, using a pipe failure study technique developed by EPRI.

Revision 1.0 of the Unit 1, Level 2 PRA model was completed in 1999, and was built upon the Level 1 Revision 1.0 model. In addition to the changes incorporated in the revision to the Level 1 model, this update reflected credit for the potential for hot leg creep rupture phenomenon to facilitate vessel failure at low pressure for early core damage sequences, and credit for a change to the emergency procedures that greatly reduced the risk from induced steam generator tube creep rupture events (these events were not modeled in the 1.0 analysis). Also, credit for containment spray recirculation was removed from the model, because of a change in emergency operating procedures.

Level 1 and Level 2, Revision 1.1

Revision 1.1 of the Unit 1, Level 1 model was completed in 1999. This was essentially the same model as Rev. 1.0; however, a single top fault tree approach to the quantification of overall CDF was used, as was a standard truncation level of 1E-10. Previously, the PRA models were quantified using SETS, which allowed different truncation levels for each individual core damage sequence.

No Level 2 or LERF model was updated with this designation (no revision to the Level 2 models or to LERF was performed which used the Level 1, Revision 1.1 model as input). The basis for this was the nearly identical nature of the Revision 1.0 and Revision 1.1 Level 1 models – no significant difference in the Level 2 results could exist based solely on the move to the Revision 1.1 model.

Level 1 and Level 2, Revision 1.2

Revision 1.2 of the Unit 1, Level 1 model was completed in 2001. Significant changes were incorporated during this revision. Many of these changes were based on comments received by the Westinghouse Owners Group (WOG) PRA Certification Team Review that took place in September 2000. Changes include:

- New LOCA break size groupings for small, medium and large LOCAs (SLOCA, MLOCA, LLOCA)
- New LOCA break size frequencies based on generic data from NUREG/CR-5750
- Update to several initiating event frequencies (LOOP, LODC)
- Inclusion of Offsite Power recovery actions for non-SBO events
- Creation of initiating event trees for the cooling water, component cooling, and instrument air systems
- PORV LOCA events have been added
- Changes to SBO success criteria (removal of diesel generator recovery)
- Random reactor coolant pump Seal Failure initiating event was added
- Updates to several system fault trees
- Credit for the Pressurizer PORV accumulator
- Upgrade to the Human Reliability Analysis (key operator actions)
- The mission time for the emergency diesel generators and cooling water pumps were changed from 6 hours to 24 hours since offsite power recovery is credited.

A full Level 2 revision to correspond with the Level 1, Revision 1.2 model has not been completed at this time. However, an update to the LERF results based on the Level 1, Revision 1.2 model has been performed.

Other than the changes to the underlying Level 1 model, the following changes were made to the LERF calculation itself:

- Failure of containment isolation was modeled using a fault tree model for each unscreened containment penetration from the previous analysis. The previous LERF analysis used a point value estimate for the failure of containment isolation.

- Core damage sequences involving early containment failure but without containment bypass (from the full Level 2 analysis) were excluded from the LERF result. These sequences had been conservatively added to the LERF calculation in the absence of certainty about whether they met an industry standard definition of large, early release that was still in development. The IPE source term analysis showed only the containment bypass events (induced-SGTR, ISLOCA) to result in the highest releases of volatile (non-

noble gas) radionuclides. SGTR events also involved large releases of volatiles, but were considered to be a late release. Containment isolation failure sequences involved early releases but the magnitude of the volatiles was categorized as medium. Also, the majority of these sequences were assumed to lead to early containment failure due to very conservative treatment of the hydrogen combustion phenomenon. In this revision only SGTR and containment isolation failures (including intersystem LOCAs), are considered in the evaluation of the LERF results.

Level 1, Revision 2.0

Subsequent to the RI-ISI assessment being performed, a Level 1, Revision 2.0 PRA model update was performed in order to obtain a working PRA model for Unit 2. Previously, all probabilistic risk analysis for Unit 2 involved application of the Unit 1 model results, with modifications that attempted to consider the impact of asymmetries between the units. The update was also performed to correct some errors and make some enhancements to the existing Revision 1.2 PRA model. The model update was completed in 2002 and was built upon the Level 1 Revision 1.2 model.

An evaluation was performed to determine the impact of the updated PRA model on the RI-ISI application, which was performed using Revision 1.2 of the PRA model. Based upon a review of the PRA model update, it was determined that there were no resulting changes to the RI-ISI consequence evaluation, and therefore no impact on the RI-ISI risk ranking and element selection activities. Additionally, the highest CCDP and CLERP used in the risk impact analysis did not change. This review is documented in Engineering Information Record No. 51-5053193-00.

MODEL REVIEWS:

NRC Review of IPE: The IPE was submitted to the NRC on March 1, 1994. The NRC review of the Prairie Island IPE was issued in May 1997. The Staff Evaluation Report (SER) concluded the following regarding the Prairie Island IPE:

1) The IPE is complete with regards to the information requested in Generic Letter 88-20 and associated guidance in NUREG-1335;
2) The IPE results are reasonable given the Prairie Island design, operation, and history; and
3) The IPE process is capable of identifying most likely severe accidents and severe accident vulnerabilities.

WOG Peer Review: In September 2000, the Westinghouse Owners Group PRA Peer Certification Review was performed on the 1999 update PRA model (Revision 1.1). In general, the review team concluded that the PINGP PRA could effectively be used to support applications involving risk significance determinations supported by deterministic analysis once the items noted in the report are addressed. A majority of the "Facts and Observation" (including all of the Level A findings) have been addressed in the Revision 1.2 model. While addressing the remaining "Facts and Observations" would take time to resolve completely, they are not expected to result in model changes that could significantly affect the overall results or conclusions of the Risk-Informed ISI consequence evaluation.

2. PROPOSED ALTERNATIVE TO CURRENT ISI PROGRAM REQUIREMENTS

2.1 ASME Section XI

ASME Section XI Examination Categories B-F, B-J, C-F-1 and C-F-2 currently contain the requirements for the nondestructive examination (NDE) of Class 1 and 2 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 and 2 piping (Examination Categories B-F, B-J, C-F-1 and C-F-2) 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 augmented inspection program was considered during the RI-ISI application:

- The augmented inspection program for flow accelerated corrosion (FAC) per Generic Letter 89-08 is relied upon to manage this damage mechanism but is not otherwise 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 PINGP. 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 1" nominal pipe size (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 ΔT assumed equal to the greatest potential ΔT for the transient, will identify all 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 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 sometimes 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 in assessing the potential for TASCS effects. The above criteria have previously been submitted by EPRI for generic approval (Letters dated February 28, 2001 and March 28, 2001, P.J. O'Regan (EPRI) to Dr. B. Sheron (USNRC), "Extension of Risk-Informed Inservice Inspection Methodology").

3.1 Scope of Program

The scope of this RI-ISI program includes all nonexempt Class 1 and 2 piping welds (IWB-1220 and IWC-1220). The systems included in the RI-ISI program are provided in Tables 3.1-1 and 3.1-2 for Units 1 and 2, respectively.

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.

Tables 3.3-1 and 3.3-2 summarize the failure potential assessment by system for each degradation mechanism that was identified as potentially operative in Units 1 and 2, respectively.

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 Tables 3.4-1 and 3.4-2 for Units 1 and 2, respectively.

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 PINGP Unit 1, the percentage of Class 1 welds selected for examination per the RI-ISI process was 10.0% (69 of 690 welds).

For PINGP Unit 2, the percentage of Class 1 welds selected for examination per the RI-ISI process was 10.4% (73 of 705 welds).

As stated in TR-112657, the existing FAC augmented inspection program provides the means to effectively manage this mechanism. No additional credit was taken for any FAC augmented inspection program locations beyond those selected by the RI-ISI process to meet the sampling percentage requirements.

A brief summary is provided in the following table, and the results of the selections are presented in Tables 3.5-1 and 3.5-2. Section 4 of EPRI TR-112657 was used as guidance in determining the examination requirements for these locations.

Unit	Class 1 Piping Welds(1)		Class 2 Piping Welds(2)		All Piping Welds(3)	
	Total	Selected	Total	Selected	Total	Selected
1	690	69	933	41	1623	110
2	705	73	951	40	1656	113

Notes

1. Includes all Category B-F and B-J locations.
2. Includes all Category C-F-1 and C-F-2 locations.
3. 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. 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.

None of the existing PINGP relief requests are being withdrawn due to the RI-ISI application.

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.

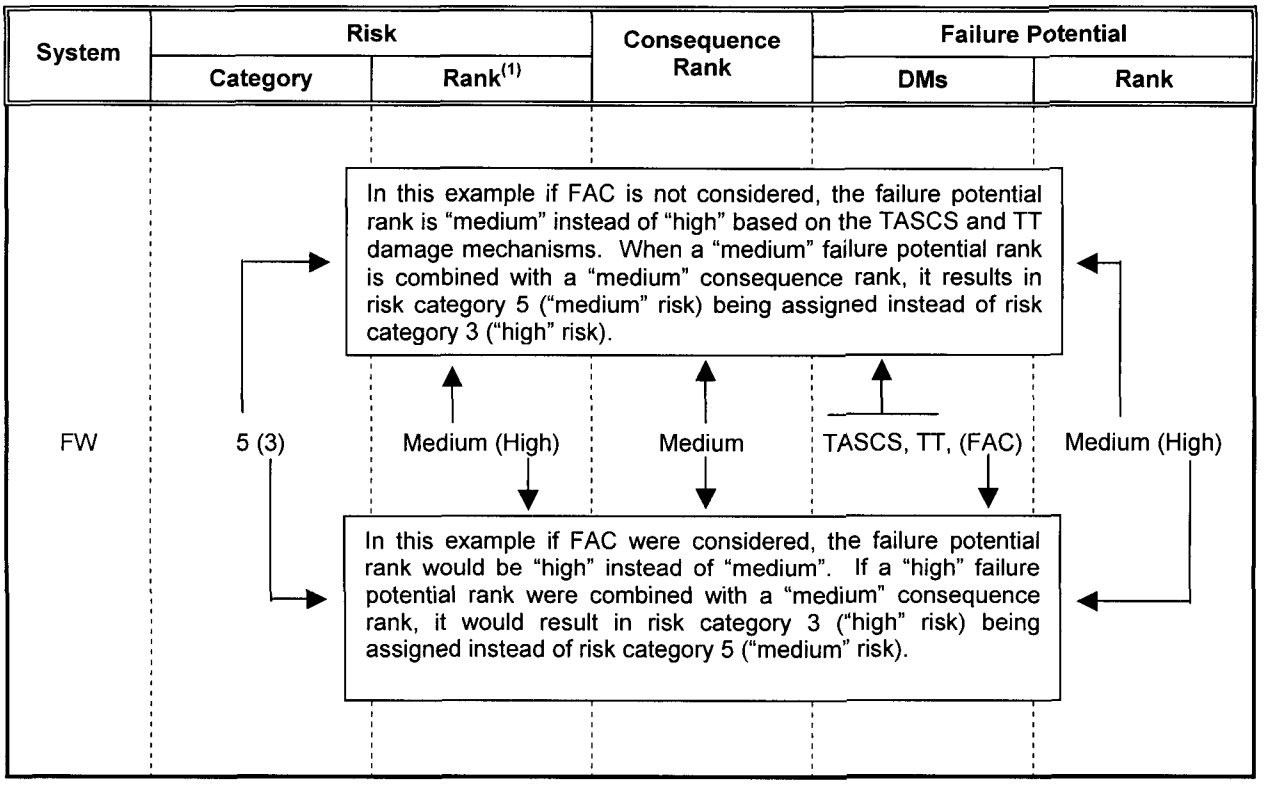
This evaluation identified the allocation of segments into High, Medium, and Low risk regions of the EPRI TR-112657 and ASME Code Case N-578 risk ranking matrix, and then determined for each of these risk classes what inspection changes are proposed for each of the locations in each segment. The changes 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.

Prairie Island 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 (6.8E-02) and CLERP (6.8E-03), 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_0 and is expected to have a value less than 1E-08. Piping locations identified as medium failure potential have a likelihood of $20x_0$. 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.

Tables 3.6-1 and 3.6-2 present summaries of the RI-ISI Program versus the applicable ASME Section XI Code Edition program requirements and identify on a per system basis each applicable risk category for Units 1 and 2, respectively. The presence of FAC was adjusted for in the performance of the quantitative analysis by excluding its impact on the risk ranking. The exclusion of the impact of FAC on the risk ranking and therefore in the determination of the change in risk is performed, because FAC is a damage mechanism managed by a separate, independent plant augmented inspection program. The RI-ISI Program credits and relies upon this augmented plant inspection program to manage this damage mechanism. The plant FAC Program will continue to determine where and when examinations shall be performed. Hence, since the number of FAC examination locations remains the same "before" and "after" and no delta exists, there is no need to include the impact of FAC in the performance of the risk impact analysis. However, in an effort to be as informative as possible, for those systems where FAC is present, Tables 3.6-1 and 3.6-2 present the information in such a manner as to depict what the resultant risk categorization is both with and without consideration of FAC. This is accomplished by enclosing the FAC damage mechanism, as well as all other resultant corresponding changes (failure potential rank, risk category and risk rank), in parenthesis. Again, this has only been done for information purposes, and has no impact on the assessment itself. The use of this approach to depict the impact of degradation mechanisms managed by augmented inspection programs on the risk categorization is consistent with that used in the delta risk assessment for the Arkansas Nuclear One, Unit 2 (ANO-2) pilot application. An example is provided below.



Note

1. The risk rank is not included in Tables 3.6-1 or 3.6-2 but it is included in Tables 5-2-1 and 5-2-2.

As indicated in the tables below, 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.

Unit 1 Risk Impact Results

System(1)	$\Delta Risk_{CDF}$		$\Delta Risk_{LERF}$	
	w/ POD	w/o POD	w/ POD	w/o POD
RC	-2.00E-07	-6.16E-08	-2.00E-08	-6.16E-09
RHR	-1.60E-08	-1.06E-08	-1.60E-09	-1.06E-09
SI	-1.37E-09	-1.37E-09	-1.37E-10	-1.37E-10
CVC	-1.41E-09	-1.39E-09	-1.41E-10	-1.39E-10
MS	negligible	negligible	negligible	negligible
FW	-2.40E-11	negligible	-2.40E-12	negligible
CS	-3.40E-10	-3.40E-10	-3.40E-11	-3.40E-11
AFW	-6.00E-12	1.00E-11	-6.00E-13	1.00E-12
Total	**-2.19E-07**	**-7.52E-08**	**-2.19E-08**	**-7.52E-09**

Note

1. Systems are described in Table 3.1-1.

Unit 2 Risk Impact Results

System[1]	$\Delta Risk_{CDF}$		$\Delta Risk_{LERF}$	
	w/ POD	w/o POD	w/ POD	w/o POD
RC	-1.91E-07	-4.69E-08	-1.91E-08	-4.69E-09
RHR	-1.60E-08	-1.06E-08	-1.60E-09	-1.06E-09
SI	-7.15E-09	-1.71E-09	-7.15E-10	-1.71E-10
CVC	-1.07E-09	-1.05E-09	-1.07E-10	-1.05E-10
MS	negligible	negligible	negligible	negligible
FW	-6.00E-12	3.00E-11	-6.00E-13	3.00E-12
CS	-3.40E-10	-3.40E-10	-3.40E-11	-3.40E-11
AFW	-6.00E-12	1.00E-11	-6.00E-13	1.00E-12
Total	**-2.16E-07**	**-6.05E-08**	**-2.16E-08**	**-6.05E-09**

Note

1. Systems are described in Table 3.1-2.

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 and 2 pressure boundaries 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 fourth inservice inspection interval. No changes to the Technical Specifications or Updated 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 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 Code 1989 Edition program requirements for in-scope piping is provided in Tables 5-1-1 and 5-2-1 for Unit 1 and Tables 5-1-2 and 5-2-2 for Unit 2. Tables 5-1-1 and 5-1-2 provide summary comparisons by risk region. Tables 5-2-1 and 5-2-2 provide the same comparison information, but in a more detailed manner by risk category, similar to the format used in Table 3.6-1 and 3.6-2.

PINGP is implementing the RI-ISI program during the first period of its fourth inspection interval. As such, 100% of the required RI-ISI program inspections will be completed in the fourth interval. Examinations shall be performed during the interval such that the period examination percentage requirements of ASME Section XI, paragraphs IWB-2412 and IWC-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 Inservice Inspection of Piping"

Supporting Onsite Documentation

Structural Integrity Calculation/File No. NMC-01-320, "Degradation Mechanism Evaluation for the Prairie Island Nuclear Generating Plant (PINGP) – Units 1/2", Revision 5 (NMC Document No. ENG-ME-594)

Structural Integrity Calculation/File No. NMC-01-321, "Risk-Informed Inservice Inspection Consequence Evaluation of Class 1 & 2 Piping – Prairie Island Nuclear Generating Plant Units 1 & 2 (Sept 04 Addendum)", Revision 0 (NMC Document No. ENG-ME-595)

NMC Document No. ENG-ME-583, "Service History Review for the Prairies Island Nuclear Generating Station" , Revision 0

Structural Integrity Calculation/File No. NMC-01-324, "Risk Ranking Summary, Matrix and Report for the Prairie Island Nuclear Generating Plant", Revision 2 (NMC Document No. ENG-ME-596)

Structural Integrity Calculation/File No. NMC-01-325, "Risk Impact Analysis for the Prairie Island Nuclear Generating Plant", Revision 2 (NMC Document No. ENG-ME-596)

Structural Integrity File No. NMC-01-103-3, Record of Conversation No. ROC-005, "Minutes of the Element Selection Meeting for the Risk-Informed ISI Project at the Prairie Island Nuclear Generating Plant", Revision 2, dated April 25-26, 2002, as updated October 22, 2004 and December 5, 2004 (NMC Document No. ENG-ME-597)

Areva Engineering Information Record No. 51-505193-00, "Prairie Island PRA Revision 2.0 Impact Review", dated November 1, 2004

Table 3.1-1

Unit 1 - System Selection and Segment / Element Definition

System Description	Number of Segments	Number of Elements
RC – Reactor Coolant	58	427
RHR – Residual Heat Removal	37	297
SI – Safety Injection	74	510
CVC – Chemical and Volume Control	13	161
MS – Main Steam	15	78
FW – Feedwater	15	52
CS – Containment Spray	8	55
AFW – Auxiliary Feedwater	6	43
Totals	**226**	**1623**

Table 3.1-2

Unit 2 - System Selection and Segment / Element Definition

System Description	Number of Segments	Number of Elements
RC – Reactor Coolant	60	418
RHR – Residual Heat Removal	38	302
SI – Safety Injection	73	536
CVC – Chemical and Volume Control	13	168
MS – Main Steam	20	92
FW – Feedwater	12	48
CS – Containment Spray	8	49
AFW – Auxiliary Feedwater	6	43
Totals	**230**	**1656**

Table 3.3-1

Unit 1 - 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								
RHR	X									X	
SI		X	X								
CVC		X									
MS											
FW	X										X
CS											
AFW	X	X									

Note

1. Systems are described in Table 3.1-1.

Table 3.3-2

Unit 2 - 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			X					
RHR	X									X	
SI	X	X	X								
CVC		X									
MS											
FW	X										X
CS											
AFW	X	X									

Note

1. Systems are described in Table 3.1-2.

Table 3.4-1

Unit 1 - Number of Segments by Risk Category With and Without Impact of FAC

System[1]	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			12	12			39	39	2	2	5	5		
RHR			1	1			17	17	1	1	18	18		
SI			1	1			24	24	2	2	43	43	4	4
CVC							4	4	3	3	6	6		
MS											15	15		
FW					7[2]	0			3	7	5	8		
CS							2	2			6	6		
AFW									2	2	4	4		
Total			14	14	7	0	86	86	13	17	102	105	4	4

Notes

1. Systems are described in Table 3.1-1.
2. Of these seven segments, four segments become Category 5 after FAC is removed from consideration due to the presence of another "medium" failure potential damage mechanism, and three segments become Category 6 after FAC is removed from consideration due to no other damage mechanism being present.

Table 3.4-2

Unit 2 - Number of Segments by Risk Category With and Without Impact of FAC

System[1]	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			14	14			39	39	2	2	5	5		
RHR			2	2			17	17	1	1	18	18		
SI			3	3			22	22	2	2	42	42	4	4
CVC							4	4	3	3	6	6		
MS											20	20		
FW					2[2]	0			5	6	5	6		
CS							2	2			6	6		
AFW									2	2	4	4		
Total			19	19	2	0	84	84	15	16	106	107	4	4

Notes

1. Systems are described in Table 3.1-2.
2. Of these two segments, one segment becomes Category 5 after FAC is removed from consideration due to the presence of another "medium" failure potential damage mechanism, and one segment becomes Category 6 after FAC is removed from consideration due to no other damage mechanism being present.

Table 3.5-1

Unit 1 - Number of Elements Selected for Inspection by Risk Category Excluding Impact of FAC

System[1]	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			79	20			322	33	2	1	24	0	0	0
RHR			4	1			174	18	4	1	115	0	0	0
SI			2	1			230	23	2	1	248	0	28	0
CVC			0	0			40	4	24	3	97	0	0	0
MS			0	0			0	0	0	0	78	0	0	0
FW			0	0			0	0	14	2[2]	38	0	0	0
CS			0	0			8	1	0	0	47	0	0	0
AFW			0	0			0	0	10	1	33	0	0	0
Total			85	22			774	79	56	9	680	0	28	0

Notes

1. Systems are described in Table 3.1-1.
2. One of the two welds was selected for examination by both the FAC and RI-ISI Programs. Since a damage mechanism other than FAC was identified, this weld will be subject to both FAC and RI-ISI examinations.

Table 3.5-2

Unit 2 - Number of Elements Selected for Inspection by Risk Category Excluding Impact of FAC

System[1]	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			81	21			319	32	2	1	16	0	0	0
RHR			2	1			178	18	4	1	118	0	0	0
SI			7	2			254	26	2	1	240	0	33	0
CVC			0	0			25	3	24	3	119	0	0	0
MS			0	0			0	0	0	0	92	0	0	0
FW			0	0			0	0	14	2[2]	34	0	0	0
CS			0	0			6	1	0	0	43	0	0	0
AFW			0	0			0	0	10	1	33	0	0	0
Total			90	24			782	80	56	9	695	0	33	0

Notes

1. Systems are described in Table 3.1-2.
2. One of the two welds was selected for examination by both the FAC and RI-ISI Programs. Since a damage mechanism other than FAC was identified, this weld will be subject to both FAC and RI-ISI examinations.

Table 3.6-1

Unit 1 - Risk Impact Analysis Results

System[1]	Category	Consequence Rank	Failure Potential		Inspections			CDF Impact[4]		LERF Impact[4]	
			DMs	Rank	SXI[2 and 3]	RI-ISI	Delta	w/ POD	w/o POD	w/ POD	w/o POD
RC	2	High	TASCS, TT	Medium	2	5	3	-5.30E-08	-2.04E-08	-5.30E-09	-2.04E-09
RC	2	High	TASCS	Medium	1	3	2	-3.26E-08	-1.36E-08	-3.26E-09	-1.36E-09
RC	2	High	TT	Medium	8	12	4	-1.14E-07	-2.72E-08	-1.14E-08	-2.72E-09
RC	4	High	None	Low	32	33	1	-3.40E-10	-3.40E-10	-3.40E-11	-3.40E-11
RC	5	Medium	TT, IGSCC	Medium	0	1	1	-1.00E-11	-1.00E-11	-1.00E-12	-1.00E-12
RC	6	Medium	None	Low	0	0	0	no change	no change	no change	no change
RC Total								**-2.00E-07**	**-6.16E-08**	**-2.00E-08**	**-6.16E-09**
RHR	2	High	TASCS	Medium	0	1	1	-1.22E-08	-6.80E-09	-1.22E-09	-6.80E-10
RHR	4	High	None	Low	7	18	11	-3.74E-09	-3.74E-09	-3.74E-10	-3.74E-10
RHR	5	Medium	E-C	Medium	0	1	1	-1.00E-11	-1.00E-11	-1.00E-12	-1.00E-12
RHR	6	Medium	None	Low	7	0	-7	negligible	negligible	negligible	negligible
RHR Total								**-1.60E-08**	**-1.06E-08**	**-1.60E-09**	**-1.06E-09**
SI	2	High	IGSCC	Medium	0	1	1	-6.80E-09	-6.80E-09	-6.80E-10	-6.80E-10
SI	4	High	None	Low	39	23	-16	5.44E-09	5.44E-09	5.44E-10	5.44E-10
SI	5	Medium	TT, IGSCC	Medium	0	1	1	-1.00E-11	-1.00E-11	-1.00E-12	-1.00E-12
SI	6	Medium	None	Low	8	0	-8	negligible	negligible	negligible	negligible
SI	7	Low	None	Low	2	0	-2	negligible	negligible	negligible	negligible
SIS Total								**-1.37E-09**	**-1.37E-09**	**-1.37E-10**	**-1.37E-10**
CVC	4	High	None	Low	0	4	4	-1.36E-09	-1.36E-09	-1.36E-10	-1.36E-10
CVC	5	Medium	TT	Medium	0	3	3	-5.40E-11	-3.00E-11	-5.40E-12	-3.00E-12
CVC	6	Medium	None	Low	0	0	0	no change	no change	no change	no change
CVC Total								**-1.41E-09**	**-1.39E-09**	**-1.41E-10**	**-1.39E-10**

Table 3.6-1

Unit 1 - Risk Impact Analysis Results

System[1]	Category	Consequence Rank	Failure Potential		Inspections			CDF Impact[4]		LERF Impact[4]	
			DMs	Rank	SXI[2 and 3]	RI-ISI	Delta	w/ POD	w/o POD	w/ POD	w/o POD
MS	6	Medium	None	Low	11	0	-11	negligible	negligible	negligible	negligible
MS Total								**negligible**	**negligible**	**negligible**	**negligible**
FW	5 (3)	Medium	TASCS, (FAC)	Medium (High)	1	1	0	-1.20E-11	no change	-1.20E-12	no change
FW	5	Medium	TASCS	Medium	1	1	0	-1.20E-11	no change	-1.20E-12	no change
FW	6 (3)	Medium	None (FAC)	Low (High)	2	0	-2	negligible	negligible	negligible	negligible
FW	6	Medium	None	Low	4	0	-4	negligible	negligible	negligible	negligible
FW Total								**-2.40E-11**	**negligible**	**-2.40E-12**	**negligible**
CS	4	High	None	Low	0	1	1	-3.40E-10	-3.40E-10	-3.40E-11	-3.40E-11
CS	6	Medium	None	Low	0	0	0	no change	no change	no change	no change
CS Total								**-3.40E-10**	**-3.40E-10**	**-3.40E-11**	**-3.40E-11**
AFW	5	Medium	TASCS, TT	Medium	2	1	-1	-6.00E-12	1.00E-11	-6.00E-13	1.00E-12
AFW	6	Medium	None	Low	5	0	-5	negligible	negligible	negligible	negligible
AFW Total								**-6.00E-12**	**1.00E-11**	**-6.00E-13**	**1.00E-12**
Grand Total								**-2.19E-07**	**-7.52E-08**	**-2.19E-08**	**-7.52E-09**

Notes

1. Systems are described in Table 3.1-1.
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. Piping weld examinations performed during the third interval per the 1989 Edition of ASME Code Section XI were used for comparison purposes for both units. In addition, for comparison purposes, the criteria of the 1998 Edition through 2000 Addenda of ASME Code Section XI were used to project the number of AFW piping welds that would have required examination since these requirements will be imposed in the upcoming fourth interval.
4. 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 3.6-2

Unit 2 - Risk Impact Analysis Results

System[1]	Category	Consequence Rank	Failure Potential		Inspections			CDF Impact[4]		LERF Impact[4]	
			DMs	Rank	SXI[2 and 3]	RI-ISI	Delta	w/ POD	w/o POD	w/ POD	w/o POD
RC	2	High	TASCS, TT	Medium	3	7	4	-7.34E-08	-2.72E-08	-7.34E-09	-2.72E-09
RC	2	High	TT, PWSCC	Medium	1	1	0	no change	no change	no change	no change
RC	2	High	TASCS	Medium	5	3	-2	-1.63E-08	1.36E-08	-1.63E-09	1.36E-09
RC	2	High	TT	Medium	5	10	5	-1.02E-07	-3.40E-08	-1.02E-08	-3.40E-09
RC	4	High	None	Low	34	32	-2	6.80E-10	6.80E-10	6.80E-11	6.80E-11
RC	5	Medium	TT, IGSCC	Medium	0	1	1	-1.00E-11	-1.00E-11	-1.00E-12	-1.00E-12
RC	6	Medium	None	Low	0	0	0	no change	no change	no change	no change
RC Total								**-1.91E-07**	**-4.69E-08**	**-1.91E-08**	**-4.69E-09**
RHR	2	High	TASCS	Medium	0	1	1	-1.22E-08	-6.80E-09	-1.22E-09	-6.80E-10
RHR	4	High	None	Low	7	18	11	-3.74E-09	-3.74E-09	-3.74E-10	-3.74E-10
RHR	5	Medium	E-C	Medium	0	1	1	-1.00E-11	-1.00E-11	-1.00E-12	-1.00E-12
RHR	6	Medium	None	Low	3	0	-3	negligible	negligible	negligible	negligible
RHR Total								**-1.60E-08**	**-1.06E-08**	**-1.60E-09**	**-1.06E-09**
SI	2	High	TT, IGSCC	Medium	1	1	0	no change	no change	no change	no change
SI	2	High	TASCS	Medium	0	1	1	-1.22E-08	-6.80E-09	-1.22E-09	-6.80E-10
SI	2	High	IGSCC	Medium	1	0	-1	6.80E-09	6.80E-09	6.80E-10	6.80E-10
SI	4	High	None	Low	21	26	5	-1.70E-09	-1.70E-09	-1.70E-10	-1.70E-10
SI	5	Medium	TT, IGSCC	Medium	0	1	1	-1.00E-11	-1.00E-11	-1.00E-12	-1.00E-12
SI	6	Medium	None	Low	8	0	-8	negligible	negligible	negligible	negligible
SI	7	Low	None	Low	2	0	-2	negligible	negligible	negligible	negligible
SI Total								**-7.15E-09**	**-1.71E-09**	**-7.15E-10**	**-1.71E-10**
CVC	4	High	None	Low	0	3	3	-1.02E-09	-1.02E-09	-1.02E-10	-1.02E-10
CVC	5	Medium	TT	Medium	0	3	3	-5.40E-11	-3.00E-11	-5.40E-12	-3.00E-12
CVC	6	Medium	None	Low	0	0	0	no change	no change	no change	no change
CVC Total								**-1.07E-09**	**-1.05E-09**	**-1.07E-10**	**-1.05E-10**

Table 3.6-2

Unit 2 - Risk Impact Analysis Results

System[1]	Category	Consequence Rank	Failure Potential		Inspections			CDF Impact[4]		LERF Impact[4]	
			DMs	Rank	SXI[2 and 3]	RI-ISI	Delta	w/ POD	w/o POD	w/ POD	w/o POD
MS	6	Medium	None	Low	16	0	-16	negligible	negligible	negligible	negligible
MS Total								**negligible**	**negligible**	**negligible**	**negligible**
FW	5 (3)	Medium	TASCS, (FAC)	Medium (High)	2	1	-1	-6.00E-12	1.00E-11	-6.00E-13	1.00E-12
FW	5	Medium	TASCS	Medium	3	1	-2	no change	2.00E-11	no change	2.00E-12
FW	6 (3)	Medium	None (FAC)	Low (High)	2	0	-2	negligible	negligible	negligible	negligible
FW	6	Medium	None	Low	2	0	-2	negligible	negligible	negligible	negligible
FW Total								**-6.00E-12**	**3.00E-11**	**-6.00E-13**	**3.00E-12**
CS	4	High	None	Low	0	1	1	-3.40E-10	-3.40E-10	-3.40E-11	-3.40E-11
CS	6	Medium	None	Low	0	0	0	no change	no change	no change	no change
CS Total								**-3.40E-10**	**-3.40E-10**	**-3.40E-11**	**-3.40E-11**
AFW	5	Medium	TASCS, TT	Medium	2	1	-1	-6.00E-12	1.00E-11	-6.00E-13	1.00E-12
AFW	6	Medium	None	Low	5	0	-5	negligible	negligible	negligible	negligible
AFW Total								**-6.00E-12**	**1.00E-11**	**-6.00E-13**	**1.00E-12**
Grand Total								**-2.16E-07**	**-6.05E-08**	**-2.16E-08**	**-6.05E-09**

Notes

1. Systems are described in Table 3.1-2.
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. Piping weld examinations performed during the third interval per the 1989 Edition of ASME Code Section XI were used for comparison purposes for both units. In addition, for comparison purposes, the criteria of the 1998 Edition through 2000 Addenda of ASME Code Section XI were used to project the number of AFW piping welds that would have required examination since these requirements will be imposed in the upcoming fourth interval.
4. 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 5-1-1

Unit 1 - Inspection Location Selection Comparison Between 1989 ASME Section XI Code and EPRI TR-112657 by Risk Region

System[1]	Code Category	High Risk Region					Medium Risk Region					Low Risk Region				
		Weld Count	1989 Section XI[2]		EPRI TR-112657		Weld Count	1989 Section XI[2]		EPRI TR-112657		Weld Count	1989 Section XI[2]		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	1		13	13	0	0		0	0	0	0	
	B-J	77	9	13	19		311	19	57	34		24	0	7	0	
RHR	B-J	0	0	0	0		28	7	0	4		0	0	0	0	
	C-F-1	4	0	0	1		150	0	0	15		115	7	0	0	
SI	B-J	2	0	0	1		21	5	1	3		51	1	13	0	
	C-F-1	0	0	0	0		209	33	3	21		223	9	8	0	
	C-F-2	0	0	0	0		2	1	0	0		2	0	0	0	
CVC	B-J	0	0	0	0		64	0	16	7		97	0	23	0	
MS	C-F-2	0	0	0	0		0	0	0	0		78	11	2	0	
FW	C-F-2	0	0	0	0		14	2	1	2[4]		38	6	0	0	
CS	C-F-1	0	0	0	0		8	0	0	1		47	0	0	0	
AFW[5]	C-F-2	0	0	0	0		10	2	0	1		33	5	0	0	
Total	B-F	2	2	0	1		13	13	0	0		0	0	0	0	
	B-J	79	9	13	20		424	31	74	48		172	1	43	0	
	C-F-1	4	0	0	1		367	33	3	37		385	16	8	0	
	C-F-2	0	0	0	0		26	5	1	3		151	22	2	0	

Notes

1. Systems are described in Table 3.1-1.
2. Since no examination selections had been made for the fourth interval ISI Program prior to the development of the RI-ISI Program, the third interval selections were used for comparison purposes. The Code of record for the third interval was the 1989 Edition of ASME Section XI. The Code Categories listed in the table are therefore in accordance with the 1989 Edition of ASME Section XI.
3. The column labeled "Other" is generally used to identify augmented inspection program locations credited per Section 3.6.5 of EPRI TR-112657. The EPRI methodology allows 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. As stated in Section 3.5 of this template, PINGP Unit 1 achieved a 10.0% sampling without relying on augmented inspection program locations beyond those selected by the RI-ISI process. The "Other" column has been retained in this table solely for uniformity purposes with the other RI-ISI application template submittals.

Notes for Table 5-1-1 (cont'd)

4. One of the two welds was selected for examination by both the FAC and RI-ISI Programs. Since a damage mechanism other than FAC was identified, this weld will be subject to both FAC and RI-ISI examinations.
5. The 1989 Edition of ASME Section XI did not require examinations on the AFW piping welds listed in this table. However, in accordance with the 1998 Edition through 2000 Addenda of ASME Section XI, AFW piping is being added to the ISI Program for the fourth ISI interval. The criteria of the 1998 Edition through 2000 Addenda of ASME Section XI were used to project the number of AFW welds that would have required examination during the fourth ISI interval per a standard ISI Program application.

Table 5-1-2

Unit 2 - Inspection Location Selection Comparison Between 1989 Section XI Code and EPRI TR-112657 by Risk Region

System[1]	Code Category	High Risk Region					Medium Risk Region					Low Risk Region				
		Weld Count	1989 Section XI[2]		EPRI TR-112657		Weld Count	1989 Section XI[2]		EPRI TR-112657		Weld Count	1989 Section XI[2]		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	1		13	13	0	0		0	0	0	0	
	B-J	79	12	9	20		308	21	60	33		16	0	7	0	
RHR	B-J	0	0	0	0		39	7	0	6		0	0	0	0	
	C-F-1	2	0	0	1		143	0	0	13		118	3	0	0	
SI	B-J	4	2	0	1		30	6	1	6		46	2	8	0	
	C-F-1	3	0	0	1		224	15	3	21		225	7	3	0	
	C-F-2	0	0	0	0		2	0	0	0		2	1	0	0	
CVC	B-J	0	0	0	0		49	0	14	6		119	0	28	0	
MS	C-F-2	0	0	0	0		0	0	0	0		92	16	4	0	
FW	C-F-2	0	0	0	0		14	5	1	2[4]		34	4	0	0	
CS	C-F-1	0	0	0	0		6	0	0	1		43	0	0	0	
AFW[5]	C-F-2	0	0	0	0		10	2	0	1		33	5	0	0	
Total	B-F	2	2	0	1		13	13	0	0		0	0	0	0	
	B-J	83	14	9	21		426	34	75	51		181	2	43	0	
	C-F-1	5	0	0	2		373	15	3	35		386	10	3	0	
	C-F-2	0	0	0	0		26	7	1	3		161	26	4	0	

Notes

1. Systems are described in Table 3.1-2.
2. Since no examination selections had been made for the fourth interval ISI Program prior to the development of the RI-ISI Program, the third interval selections were used for comparison purposes. The Code of record for the third interval was the 1989 Edition of ASME Section XI. The Code Categories listed in the table are therefore in accordance with the 1989 Edition of ASME Section XI.

Notes for Table 5-1-2 (cont'd)

3. The column labeled "Other" is generally used to identify augmented inspection program locations credited per Section 3.6.5 of EPRI TR-112657. The EPRI methodology allows 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. As stated in Section 3.5 of this template, PINGP Unit 2 achieved a 10% sampling without relying on augmented inspection program locations beyond those selected by the RI-ISI process. The "Other" column has been retained in this table solely for uniformity purposes with the other RI-ISI application template submittals.
4. One of the two welds was selected for examination by both the FAC and RI-ISI Programs. Since a damage mechanism other than FAC was identified, this weld will be subject to both FAC and RI-ISI examinations.
5. The 1989 Edition of ASME Section XI did not require examinations on the AFW piping welds listed in this table. However, in accordance with the 1998 Edition through 2000 Addenda of ASME Section XI, AFW piping is being added to the ISI Program for the fourth ISI interval. The criteria of the 1998 Edition through 2000 Addenda of ASME Section XI were used to project the number of AFW welds that would have required examination during the fourth ISI interval per a standard ISI Program application.

Table 5-2-1

Unit 1 - Inspection Location Selection Comparison Between 1989 Section XI Code and EPRI TR-112657 by Risk Category

System[1]	Risk		Consequence Rank	Failure Potential		Code Category	Weld Count	1989 Section XI[2]		EPRI TR-112657	
	Category	Rank		DMs	Rank			Vol/Sur	Sur Only	RI-ISI	Other[3]
RC	2	High	High	TASCS, TT	Medium	B-J	14	2	1	5	
RC	2	High	High	TASCS	Medium	B-J	12	1	1	3	
RC	2	High	High	TT	Medium	B-F	2	2	0	1	
						B-J	51	6	11	11	
RC	4	Medium	High	None	Low	B-F	13	13	0	0	
						B-J	309	19	57	33	
RC	5	Medium	Medium	TT, IGSCC	Medium	B-J	2	0	0	1	
RC	6	Low	Medium	None	Low	B-J	24	0	7	0	
RHR	2	High	High	TASCS	Medium	C-F-1	4	0	0	1	
RHR	4	Medium	High	None	Low	B-J	28	7	0	4	
						C-F-1	146	0	0	14	
RHR	5	Medium	Medium	E-C	Medium	C-F-1	4	0	0	1	
RHR	6	Low	Medium	None	Low	C-F-1	115	7	0	0	
SI	2	High	High	IGSCC	Medium	B-J	2	0	0	1	
SI	4	Medium	High	None	Low	B-J	19	5	0	2	
						C-F-1	209	33	3	21	
						C-F-2	2	1	0	0	
SI	5	Medium	Medium	TT, IGSCC	Medium	B-J	2	0	1	1	
SI	6	Low	Medium	None	Low	B-J	51	1	13	0	
						C-F-1	195	7	7	0	
						C-F-2	2	0	0	0	
SI	7	Low	Low	None	Low	C-F-1	28	2	1	0	

Table 5-2-1 (cont'd)

Unit 1 - Inspection Location Selection Comparison Between 1989 Section XI Code and EPRI TR-112657 by Risk Category

System[1]	Risk		Consequence Rank	Failure Potential		Code Category	Weld Count	1989 Section XI[2]		EPRI TR-112657	
	Category	Rank		DMs	Rank			Vol/Sur	Sur Only	RI-ISI	Other[3]
CVC	4	Medium	High	None	Low	B-J	40	0	11	4	
CVC	5	Medium	Medium	TT	Medium	B-J	24	0	5	3	
CVC	6	Low	Medium	None	Low	B-J	97	0	23	0	
MS	6	Low	Medium	None	Low	C-F-2	78	11	2	0	
FW	5 (3)	Medium (High)	Medium	TASCS, (FAC)	Medium (High)	C-F-2	9	1	1	1[4]	
FW	5	Medium	Medium	TASCS	Medium	C-F-2	5	1	0	1	
FW	6 (3)	Low (High)	Medium	None (FAC)	Low (High)	C-F-2	8	2	0	0	
FW	6	Low	Medium	None	Low	C-F-2	30	4	0	0	
CS	4	Medium	High	None	Low	C-F-1	8	0	0	1	
CS	6	Low	Medium	None	Low	C-F-1	47	0	0	0	
AFW[5]	5	Medium	Medium	TASCS, TT	Medium	C-F-2	10	2	0	1	
AFW[5]	6	Low	Medium	None	Low	C-F-2	33	5	0	0	

Notes

1. Systems are described in Table 3.1-1.
2. Since no examination selections had been made for the fourth interval ISI Program prior to the development of the RI-ISI Program, the third interval selections were used for comparison purposes. The Code of record for the third interval was the 1989 Edition of ASME Section XI. The Code Categories listed in the table are therefore in accordance with the 1989 Edition of ASME Section XI.
3. The column labeled "Other" is generally used to identify augmented inspection program locations credited per Section 3.6.5 of EPRI TR-112657. The EPRI methodology allows 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. As stated in Section 3.5 of this template, PINGP Unit 1 achieved a 10.0% sampling without relying on augmented inspection program locations beyond those selected by the RI-ISI process. The "Other" column has been retained in this table solely for uniformity purposes with the other RI-ISI application template submittals.
4. This weld was selected for examination by both the FAC and RI-ISI Programs. Since a damage mechanism other than FAC was identified, this weld will be subject to both FAC and RI-ISI examinations.
5. The 1989 Edition of ASME Section XI did not require examinations on the AFW piping welds listed in this table. However, in accordance with the 1998 Edition through 2000 Addenda of ASME Section XI, AFW piping is being added to the ISI Program for the fourth ISI interval. The criteria of the 1998 Edition through 2000 Addenda of ASME Section XI were used to project the number of AFW welds that would have required examination during the fourth ISI interval per a standard ISI Program application.

Table 5-2-2

Unit 2 - Inspection Location Selection Comparison Between 1989 Section XI Code and EPRI TR-112657 by Risk Category

System[1]	Risk		Consequence Rank	Failure Potential		Code Category	Weld Count	1989 Section XI[2]		EPRI TR-112657	
	Category	Rank		DMs	Rank			Vol/Sur	Sur Only	RI-ISI	Other[3]
RC	2	High	High	TASCS, TT	Medium	B-J	21	3	2	7	
RC	2	High	High	PWSCC, TT	Medium	B-F	1	1	0	1	
RC	2	High	High	TASCS	Medium	B-J	13	5	0	3	
RC	2	High	High	TT	Medium	B-F	1	1	0	0	
						B-J	45	4	7	10	
RC	4	Medium	High	None	Low	B-F	13	13	0	0	
						B-J	306	21	60	32	
RC	5	Medium	Medium	TT, IGSCC	Medium	B-J	2	0	0	1	
RC	6	Low	Medium	None	Low	B-J	16	0	7	0	
RHR	2	High	High	TASCS	Medium	C-F-1	2	0	0	1	
RHR	4	Medium	High	None	Low	B-J	39	7	0	6	
						C-F-1	139	0	0	12	
RHR	5	Medium	Medium	E-C	Medium	C-F-1	4	0	0	1	
RHR	6	Low	Medium	None	Low	C-F-1	118	3	0	0	
SI	2	High	High	TT, IGSCC	Medium	B-J	2	1	0	1	
SI	2	High	High	TASCS	Medium	C-F-1	3	0	0	1	
SI	2	High	High	IGSCC	Medium	B-J	2	1	0	0	
SI	4	Medium	High	None	Low	B-J	28	6	0	5	
						C-F-1	224	15	3	21	
						C-F-2	2	0	0	0	
SI	5	Medium	Medium	TT, IGSCC	Medium	B-J	2	0	1	1	
SI	6	Low	Medium	None	Low	B-J	46	2	8	0	
						C-F-1	192	5	2	0	
						C-F-2	2	1	0	0	
SI	7	Low	Low	None	Low	C-F-1	33	2	1	0	

Table 5-2-2 (cont'd)

Unit 2 - Inspection Location Selection Comparison Between 1989 Section XI Code and EPRI TR-112657 by Risk Category

System[1]	Risk		Consequence Rank	Failure Potential		Code Category	Weld Count	1989 Section XI[2]		EPRI TR-112657	
	Category	Rank		DMs	Rank			Vol/Sur	Sur Only	RI-ISI	Other[3]
CVC	4	Medium	High	None	Low	B-J	25	0	8	3	
CVC	5	Medium	Medium	TT	Medium	B-J	24	0	6	3	
CVC	6	Low	Medium	None	Low	B-J	119	0	28	0	
MS	6	Low	Medium	None	Low	C-F-2	92	16	4	0	
FW	5 (3)	Medium (High)	Medium	TASCS, (FAC)	Medium (High)	C-F-2	4	2	0	1[4]	
FW	5	Medium	Medium	TASCS	Medium	C-F-2	10	3	1	1	
FW	6 (3)	Low (High)	Medium	None (FAC)	Low (High)	C-F-2	2	2	0	0	
FW	6	Low	Medium	None	Low	C-F-2	32	2	0	0	
CS	4	Medium	High	None	Low	C-F-1	6	0	0	1	
CS	6	Low	Medium	None	Low	C-F-1	43	0	0	0	
AFW[5]	5	Medium	Medium	TASCS, TT	Medium	C-F-2	10	2	0	1	
AFW[5]	6	Low	Medium	None	Low	C-F-2	33	5	0	0	

Notes

1. Systems are described in Table 3.1-2.
2. Since no examination selections had been made for the fourth interval ISI Program prior to the development of the RI-ISI Program, the third interval selections were used for comparison purposes. The Code of record for the third interval was the 1989 Edition of ASME Section XI. The Code Categories listed in the table are therefore in accordance with the 1989 Edition of ASME Section XI.
3. The column labeled "Other" is generally used to identify augmented inspection program locations credited per Section 3.6.5 of EPRI TR-112657. The EPRI methodology allows 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. As stated in Section 3.5 of this template, PINGP Unit 2 achieved greater than a 10% sampling without relying on augmented inspection program locations beyond those selected by the RI-ISI process. The "Other" column has been retained in this table solely for uniformity purposes with the other RI-ISI application template submittals.
4. This weld was selected for examination by both the FAC and RI-ISI Programs. Since a damage mechanism other than FAC was identified, this weld will be subject to both FAC and RI-ISI examinations.
5. The 1989 Edition of ASME Section XI did not require examinations on the AFW piping welds listed in this table. However, in accordance with the 1998 Edition through 2000 Addenda of ASME Section XI, AFW piping is being added to the ISI Program for the fourth ISI interval. The criteria of the 1998 Edition through 2000 Addenda of ASME Section XI were used to project the number of AFW welds that would have required examination during the fourth ISI interval per a standard ISI Program application.