WOLF CREEK
NUCLEAR OPERATING CORPORATION

Richard A. Muench
Vice President Engineering & Information Services

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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station: P1-137
Washington, D. C. 20555

Subject: Docket No. 50-482: Relief Request for Application of an Alternative to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI Examination Requirements for Class 1 and 2 Piping Welds

Gentlemen:

In accordance with the provisions of 10 CFR 50.55a(a)(3)(i), Wolf Creek Nuclear Operating Corporation (WCNOC) requests relief from the ASME Section XI code examination requirements for inservice inspection of Class 1 and 2 piping welds (Categories B-F, B-J, C-F-1, and C-F-2). The proposed alternative, as described in Attachment 1,"Risk-Informed Inservice Inspection Program Plan – Wolf Creek Generating Station," provides an acceptable level of quality and safety as required by 10 CFR 50.55a(a)(3)(i).

The Wolf Creek Generating Station (WCGS) risk-informed inservice inspection (RI-ISI) program plan has been developed in accordance with the methodology provided in Electric Power Research Institute (EPRI) Topical Report TR-112657, "Revised Risk-Informed Inservice Inspection Evaluation Procedure," Revision B-A. EPRI TR-112657, Revision B, has been reviewed and accepted by the Nuclear Regulatory Commission (NRC). The NRC Staff has found TR-112657, Revision B, acceptable for referencing in licensing applications to the extent specified and under the limitations delineated in the report and the NRC Safety Evaluation Report, dated October 28, 1999.

The format of the WCGS RI-ISI program plan is consistent with the Nuclear Energy Institute (NEI)/industry template developed for applications of the EPRI RI-ISI methodology. Additional supporting documentation is available at the WCGS site for your review.

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The WCGS RI-ISI program plan was developed in conjunction with RI-ISI program plans for the plants operated by Pacific Gas and Electric Company, AmerenUE, TXU Electric, and STP Nuclear Operating Company. WCGS and these other plants make up an industry consortium of five plants as a result of a mutual agreement known as Strategic Teaming and Resource Sharing (STARS). The other members of the STARS group can also be expected to submit similar plant-specific relief requests. These additional relief requests will be submitted in parallel with this application, in order to reduce the amount of NRC resources required to review and approve the STARS applications. Attachment 2 describes the methodology for identifying differences in the STARS RI-ISI applications to assist in the review of the applications.

The recent event at the V.C. Summer facility in which through-wall cracking was identified in a 34-inch main loop hot leg reactor pressure vessel nozzle has led to an extensive industry effort to determine generic implications and appropriate corrective actions. As discussed in the Nuclear Energy Institute (NEI) letter from David Modeen to Dr. Brian Sheron dated December 14, 2000, the EPRI Materials Reliability Project will lead the industry effort to address the generic implications of the V.C. Summer event. WCNOC will closely monitor the progress of and will assess the recommendations for applicability.

Attachment 3 provides a summary of regulatory commitments made in this submittal.

WCNOC requests NRC approval of this relief request by August 2001 to support the WCGS refueling outage (RF-12), which is currently scheduled to begin in March 2002. WCGS intends to incorporate this risk-informed approach for Class 1 and 2 piping weld inspection into the second interval WCGS Inservice Inspection Program Plan which began in September 1995, and is in effect until September 2005.

Very truly yours,

Richard A. Muench

RAM/rlr
Attachments

cc: J. N. Donohew (NRC), w/a
W. D. Johnson (NRC), w/a
E. W. Merschoff (NRC), w/a
Senior Resident Inspector (NRC), w/a

RISK-INFORMED INSERVICE INSPECTION PROGRAM PLAN

WOLF CREEK GENERATING STATION (REVISION 0)

Table of Contents

1. INTRODUCTION

[The Wolf Creek Generating Station (WCGS) is currently in the second inservice inspection (ISI) interval as defined by the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Section XI Code for Program B. Pursuant to 10 CFR 50.55a(g)(4)(ii), the applicable ASME Section XI Code for the WCGS is the 1989 Edition, no Addenda.]

The objective of this submittal is to request a change to the ISI Program for Class [1 and 2] piping through the use of a risk-informed inservice inspection (RI-ISI) program. 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 Wolf Creek probabilistic risk assessment (PRA) model used to evaluate the consequences of pipe rupture for the RI-ISI assessment was the most current PRA model update. The Wolf Creek PRA was originally developed to satisfy the requirement of NRC Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities," that each licensee perform an Individual Plant Examination (IPE) to search for plant specific severe accident vulnerabilities. The results of the Wolf Creek PRA were submitted to the NRC, pursuant to this requirement, on September 28, 1992. The NRC issued a Safety Evaluation Report (SER) on the Wolf Creek IPE Submittal on November 18, 1996. The SER on the IPE concluded that the Wolf Creek PRA has met the intent of Generic Letter 88-20. Since completion of the Wolf Creek IPE, the PRA model has been used to support various plant programs. These include the Maintenance Rule program and Safety Monitor™ development.]

[The following Wolf Creek PRA results for at power plant operation were obtained from the updated model:

- Core Damage Frequency (CDF) = 5.5E-05 per year (excludes internal floods).

- Large Early Release Frequency (LERF) = 8.3E-07 per year.]

[This LERF value is dominated by Steam Generator Tube Rupture and Interfacing Systems Loss of Coolant Accident (LOCA) initiating events.]

[This LERF value is dominated by Steam Generator Tube Rupture and Interfacing Systems Loss of Coolant Accident (LOCA) initiating events.]

[In August 2000, the Wolf Creek PRA went through the Westinghouse Owner's Group Peer Review process. The overall preliminary assessment concluded the following:

- Risk significance determinations made by PRA are adequate to support regulatory applications when combined with deterministic insights.

- PRA results can support physical plant changes when it is used in conjunction with other deterministic approach.

- PRA results can be used in licensing submittals to the NRC to support position regarding absolute level of safety significance if supported by deterministic evaluations.]

[In order to continue to use the PRA model as a tool to support plant programs, periodic update of the model is necessary. The most recent update of the Wolf Creek PRA was completed in August of 1999, with a freeze date of January 1998. This update included numerous changes to the PRA model to reflect plant modifications, changes to plant specific and generic initiating event frequencies, import of initiating event frequencies for special initiators as fault tree solution files, changes to plant specific component failure rates and test and maintenance unavailability data, and expansion of the modeling scope for a number of systems previously modeled as single failure events.]

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 currently approved 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 programs were considered during the RI-ISI application:

- The augmented inspection program for flow accelerated corrosion (FAC) per Generic Letter 89-08, "Erosion/Corrosion - Induced Pipe Wall Thinning," is relied upon to manage this damage mechanism but is not otherwise affected or changed by the RI-ISI program.

[• Examinations on Main Steam and Feedwater system piping, defined as "No Break Zone" piping in Section 3.6.2 of the Updated Safety Analysis Report (USAR), shall be performed in accordance with NUREG-0800, Standard Review Plan (SRP) 3.6.2, "Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping," and SRP 6.6, "Inservice Inspection of Class 2 and 3 Components." The augmented inspection program for high energy "No Break Zone" piping is not affected by this 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 WCGS. 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 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 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 facing lines, significant top-to-bottomΔTs can develop in horizontal sections within about 25 pipe diameters, and the conditions can potentially be cyclic. For an upward or horizontal facing branch line connected to the hot fluid source, natural convective effects will fill the line with hot water. In the absence of in-leakage towards the hot fluid source, this will result in a well-mixed fluid condition where significant top-to-bottom ΔTs will not occur. 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. 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 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 can occur outward past a valve into a line with a significant temperature difference. However, since this is a 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.

These additional considerations for determining the potential for thermal fatigue as a result of the effects of TASCS were applied in the failure potential assessment for WCGS. This constitutes a deviation to the requirements of EPRI TR-112657 since the methodology does not presently provide any allowance for the consideration of cycle severity in assessing the potential for TASCS effects. For the reasons discussed above, this approach is considered technically justifiable. Furthermore, EPRI concurs with this position and intends to address this issue in a future revision to the methodology.

3.1 Scope of Program

The systems included in the RI-ISI program are provided in Table[ ] 3.1-1 [ ]. The piping and instrumentation diagrams and additional plant information including the existing plant ISI program were used to define the Class [1 and 2] 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 (isolation, bypass and large, early release). The impact on these measures due to both direct and indirect effects was considered using the guidance provided in EPRI TR-112657. Internal events, internal flooding, containment performance, other modes of operation (e.g., shutdown operation), and external events are evaluated in the analysis.

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.

Table[ ] 3.3-1 [ ] 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 (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[ ] 3.4-1 [ ].

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 WCGS, the percentage of Class 1 welds selected for examination per the RI-ISI process is 8.8%, which is not a significant departure from 10%.]

[One additional factor that was considered during the evaluation was that the overall percentage of Class 1 selections was 8.8% when both socket and non-socket piping welds were considered. Due to the fact that WCGS only has four socket welds in Class 1 piping, there is no appreciable change in the selection percentage when only non-socket welds are considered. However, it should be noted that non-socket welds are subject to volumetric examination, so this percentage does not rely upon welds that are solely subject to a VT-2 visual examination.]

A brief summary is provided below, and the results of the selection process are presented in Table[ ] 3.5-1 [ ]. It should be noted that no credit was taken for any FAC or existing high energy "No Break Zone" piping augmented inspection program locations in meeting the sampling percentage requirements. 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	705	62	1384	58	2089	120

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 segments are subject to the same root cause conditions. Additional examinations will be performed on these elements up to a number equivalent to the number of elements required to be inspected on the segment or segments initially. 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.

At this time, all the RI-ISI examination locations that have been selected provide >90% coverage. In instances where locations may be 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 WCGS 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 CDF and LERF be less than 1E-07 and 1E-08 per year per system, respectively.

Wolf Creek Nuclear Operating Corporation (WCNOC) 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 [(1.03E-02)] and CLERP [(6.07E-04)], 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. The PBFlikelihoods and POD values used in the analysis are consistent with those used in the approved RI-ISI pilot applications at Arkansas Nuclear One, Unit 2, and Vermont Yankee, as documented in References 9 an 14 of EPRI TR-112657.

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 [ ]. The presence of FAC was adjusted for in the performance of the quantitative analysis by excluding its impact on the risk ranking. However, in an effort to be as informative as possible, for those systems where FAC is present, the information in Table 3.6-1 is presented 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.

System	**Risk**		**Consequence Rank**	**Failure Potential**	
	Category	**Rank(1)**		**DMs**	**Rank**
AE	5 (3)	Medium (High)	Medium	TASCS, TT, (FAC)	Medium (High)

In this example if FAC is not considered, the failure potential rank is "medium" instead of "high" based on the TASCS and TT damage mechanisms. When a "medium" failure potential rank is combined with a "medium" consequence rank, it results in risk category 5 ("medium" risk) being assigned instead of risk category 3 ("high" risk).

In this example if FAC were considered, the failure potential rank would be "high" instead of "medium". If a "high" failure potential rank were combined with a "medium" consequence rank, it would result in risk category 3 ("high" risk) being assigned instead of risk category 5 ("medium" risk).

Note

1. The risk rank is not included in Table 3.6-1 but it is included in Table 5-2.

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
AB	negligible	negligible	negligible	negligible
AE	-3.00E-11	-1.00E-11	-3.00E-12	-1.00E-12
BB	-1.36E-08	-8.24E-10	-8.01E-10	-4.86E-11
BG	-9.56E-09	-5.43E-09	-5.65E-10	-3.21E-10
BN	negligible	negligible	negligible	negligible
EF	negligible	negligible	negligible	negligible
EJ	-7.67E-09	1.55E-10	-4.52E-10	9.10E-12
EM	4.15E-11	4.15E-11	2.04E-12	2.04E-12
EN	negligible	negligible	negligible	negligible
EP	3.19E-10	3.19E-10	1.92E-11	1.92E-11
GS	no change	no change	no change	no change
Total	**-3.05E-08**	**-5.75E-09**	**-1.80E-09**	**-3.40E-10**

Note

1. Systems are described in Table 3.1-1.

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

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 second inservice inspection interval. No changes to the [USAR] 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
D. Evaluate, develop a corrective action plan or plans
E. Decide
F. Implement
G. Monitor
H. 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 program requirements for in-scope piping is provided in Tables [5-1 and 5-2]. Table[5-1] provide[s] a summary comparison by risk region. Table[5-2] provide[s] the same comparison information, but in a more detailed manner by risk category, similar to the format used in Table[ ] 3.6-1 [ ].

[WCGS is currently in the middle of the second period of its second inspection interval. Up until this point, 33% of the examinations required by ASME Section XI have been completed for Examination Categories B-F, B-J, C-F-1, and C-F-2 piping welds. The final outage scheduled for the second period is Refuel Outage 12 (RF-12), which will occur in Spring 2002. In RF-12, the examinations determined by the RI-ISI process will replace those formerly selected per ASME Section XI criteria. Since 33% of the examinations have been completed thus far in the second interval, 67% of the RI-ISI examinations will be performed during RF-12 and the remaining refueling outages in the third period so that 100% of the selected examinations are performed during the course of the interval.]

Subsequent ISI intervals will implement 100% of the examination locations selected per the RI-ISI program. These examinations will be distributed between periods such that the period 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

[Calculation No. AN-00-35, "STARS Risk-Informed ISI Project– Consequence Evaluation," Wolf Creek Generating Station, Rev. 0.]

[Calculation No. WCRE-12, "Risk-Informed ISI Basis Document," Wolf Creek Generating Station, Rev. 0.]

["Wolf Creek Risk Ranking Summary, Matrix and Report," Rev. 0, dated October 10, 2000.]

[Record of Conversation No. ROC-002, "Minutes of the Element Selection Meeting for the Risk-Informed ISI Project at the Callaway Plant and Wolf Creek Generating Station," dated August 24th and 25th, 2000.]

["Risk Impact Analysis for the Wolf Creek Generating Station," Rev. 0.]

Table 3.1-1

System Selection and Segment / Element Definition

System Description	ASME Code Class	Number of Segments	Number of Elements
AB – Main Steam System	Class 2	16	154
AE – Main Feedwater System	Class 2	13	124
BB – Reactor Coolant System	Class 1	81	337
BG – Chemical and Volume Control System	Class 1 and 2	50	192
BN – Borated Refueling Water Storage System	Class 2	29	125
EF – Essential Service Water System	Class 2	8	26
EJ – Residual Heat Removal System	Class 1 and 2	49	524
EM – High Pressure Coolant Injection System	Class 1 and 2	67	397
EN – Containment Spray System	Class 2	14	93
EP – Accumulator Safety Injection System	Class 1	20	115
GS – Containment Hydrogen Control System	Class 2	1	2
Totals		**348**	**2089**

Table 3.3-1

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
AB											
AE	X										X
BB	X	X									
BG	X	X									
BN											
EF											
EJ	X	X									
EM		X	X								
EN											
EP			X								
GS											

Note

1. Systems are described in Table 3.1-1.

Table 3.4-1

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
AB											16	16		
AE					13[2]	0			0	4	0	9		
BB			22	22			53	53			2	2	4	4
BG			9	9			21	21	5	5	10	10	5	5
BN							3	3			26	26		
EF											8	8		
EJ			12	12			31	31			3	3	3	3
EM			8	8			9	9	8	8	37	37	5	5
EN											14	14		
EP							4	4	4	4	12	12		
GS													1	1
Total			51	51	13	0	121	121	17	21	128	137	18	18

Notes

1. Systems are described in Table 3.1-1.
2. Of these 13 segments, 4 segments become Category 5 after FAC is removed from consideration due to the presence of another "medium" failure potential damage mechanism, and 9 segments become Category 6 after FAC is removed from consideration due to no other damage mechanisms being present.

Table 3.5-1

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
AB											154	0		
AE									16	2	108	0		
BB			38	11			281	26			6	0	12	0
BG			18	5			98	12	7	2	51	0	18	0
BN							3	0			122	0		
EF											26			
EJ			18	6			456	47			3	0	47	0
EM			8	0			22	3	20	2	309	0	38	0
EN											93	0		
EP							20	2	12	2	83	0		
GS													2	0
Total			82	22			880	90	55	8	955	0	117	0

Notes

1. Systems are described in Table 3.1-1.

Table 3.6-1

Risk Impact Analysis Results

System(1)	Category	Consequence Rank	Failure Potential		Inspections			CDF Impact(3)		LERF Impact(3)	
			DMs	Rank	Section XI(2)	RI-ISI	Delta	w/ POD	w/o POD	w/ POD	w/o POD
AB	6	Medium	None	Low	14	0	-14	negligible	negligible	negligible	negligible
AB Total								**negligible**	**negligible**	**negligible**	**negligible**
AE	5 (3)	Medium	TASCS, (FAC)	Medium (High)	1	2	1	-3.00E-11	-1.00E-11	-3.00E-12	-1.00E-12
AE	6 (3)	Medium	None (FAC)	Low (High)	11	0	-11	negligible	negligible	negligible	negligible
AE Total								**-3.00E-11**	**-1.00E-11**	**-3.00E-12**	**-1.00E-12**
BB	2	High	TASCS, TT	Medium	2	3	1	-4.33E-09	-1.03E-09	-2.55E-10	-6.07E-11
BB	2	High	TASCS	Medium	3	2	-1	-1.85E-09	1.03E-09	-1.09E-10	6.07E-11
BB	2	High	TT	Medium	4	6	2	-8.65E-09	-2.06E-09	-5.10E-10	-1.21E-10
BB	4	High	None	Low	50	26	-24	1.24E-09	1.24E-09	7.28E-11	7.28E-11
BB	6	Medium	None	Low	0	0	0	no change	no change	no change	no change
BB	7	Low	None	Low	0	0	0	no change	no change	no change	no change
BB Total								**-1.36E-08**	**-8.24E-10**	**-8.01E-10**	**-4.86E-11**
BG	2	High	TASCS	Medium	0	2	2	-3.71E-09	-2.06E-09	-2.19E-10	-1.21E-10
BG	2	High	TT	Medium	0	3	3	-5.56E-09	-3.09E-09	-3.28E-10	-1.82E-10
BG	4	High	None	Low	7	12	5	-2.58E-10	-2.58E-10	-1.52E-11	-1.52E-11
BG	5	Medium	TT	Medium	0	2	2	-3.60E-11	-2.00E-11	-3.60E-12	-2.00E-12
BG	6	Medium	None	Low	0	0	0	no change	no change	no change	no change
BG	7	Low	None	Low	0	0	0	no change	no change	no change	no change
BG Total								**-9.56E-09**	**-5.43E-09**	**-5.65E-10**	**-3.21E-10**
BN	4	High	None	Low	0	0	0	no change	no change	no change	no change
BN	6	Medium	None	Low	9	0	-9	negligible	negligible	negligible	negligible

Table 3.6-1

Risk Impact Analysis Results

System(1)	Category	Consequence Rank	Failure Potential		Inspections			CDF Impact(3)		LERF Impact(3)	
			DMs	Rank	Section XI(2)	RI-ISI	Delta	w/ POD	w/o POD	w/ POD	w/o POD
BN Total								**negligible**	**negligible**	**negligible**	**negligible**
EF	6	Medium	None	Low	2	0	-2	negligible	negligible	negligible	negligible
EF Total								**negligible**	**negligible**	**negligible**	**negligible**
EJ	2	High	TASCS, TT	Medium	1	0	-1	6.18E-10	1.03E-09	3.64E-11	6.07E-11
EJ	2	High	TASCS	Medium	1	2	1	-3.09E-09	-1.03E-09	-1.82E-10	-6.07E-11
EJ	2	High	TT	Medium	5	4	-1	-4.33E-09	1.03E-09	-2.55E-10	6.07E-11
EJ	4	High	None	Low	30	47	17	-8.76E-10	-8.76E-10	-5.16E-11	-5.16E-11
EJ	6	Medium	None	Low	1	0	-1	negligible	negligible	negligible	negligible
EJ	7	Low	None	Low	9	0	-9	negligible	negligible	negligible	negligible
EJ Total								**-7.67E-09**	**1.55E-10**	**-4.52E-10**	**9.10E-12**
EM	2	High	TT	Medium	0	0	0	no change	no change	no change	no change
EM	4	High	None	Low	4	3	-1	5.15E-11	5.15E-11	3.04E-12	3.04E-12
EM	5	Medium	IGSCC	Medium	1	2	1	-1.00E-11	-1.00E-11	-1.00E-12	-1.00E-12
EM	6	Medium	None	Low	19	0	-19	negligible	negligible	negligible	negligible
EM	7	Low	None	Low	2	0	-2	negligible	negligible	negligible	negligible
EM Total								**4.15E-11**	**4.15E-11**	**2.04E-12**	**2.04E-12**
EN	6	Medium	None	Low	8	0	-8	negligible	negligible	negligible	negligible
EN Total								**negligible**	**negligible**	**negligible**	**negligible**
EP	4	High	None	Low	8	2	-6	3.09E-10	3.09E-10	1.82E-11	1.82E-11
EP	5	Medium	IGSCC	Medium	3	2	-1	1.00E-11	1.00E-11	1.00E-12	1.00E-12
EP	6	Medium	None	Low	6	0	-6	negligible	negligible	negligible	negligible
EP Total								**3.19E-10**	**3.19E-10**	**1.92E-11**	**1.92E-11**

Table 3.6-1

Risk Impact Analysis Results

System(1)	Category	Consequence Rank	Failure Potential		Inspections			CDF Impact(3)		LERF Impact(3)	
			DMs	Rank	Section XI(2)	RI-ISI	Delta	w/ POD	w/o POD	w/ POD	w/o POD
GS	7	Low	None	Low	0	0	0	no change	no change	no change	no change
GS Total								**no change**	**no change**	**no change**	**no change**
Grand Total								**-3.05E-08**	**-5.75E-09**	**-1.80E-09**	**-3.40E-10**

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 this count. Inspection locations previously subjected to a surface examination only are 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. Hence, the word "negligible" is given in these cases in lieu of values for CDF and LERF Impact. In those cases where no inspections were being performed previously via Section XI, and none are planned for RI-ISI purposes, "no change" is listed instead of "negligible".

Table 5-1

Inspection Location Selection Comparison Between 1989 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	1989 Section XI		EPRI TR-112657		Weld Count	1989 Section XI		EPRI TR-112657		Weld Count	1989 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]
AB	C-F-2											154	14	0	0	
AE	C-F-2						16	1	0	2		108	11	0	0	
BB	B-F	1	1	0	0		13	13	0	5						
	B-J	37	8	7	11		268	37	14	21		18	0	2	0	
BG	B-J	18	0	10	5		33	0	12	5		12	0	4	0	
	C-F-1						72	7	3	9		57	0	0	0	
BN	C-F-1						3	0	0	0		122	9	1	0	
EF	C-F-2											26	2	0	0	
EJ	B-J	14	6	0	3		22	3	0	3		2	1	0	0	
	C-F-1	4	1	0	3		434	27	0	44		48	9	0	0	
EM	B-J	8	0	3	0		40	5	4	5		104	2	24	0	
	C-F-1						2	0	0	0		243	19	0	0	

Table 5-1

Inspection Location Selection Comparison Between 1989 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	1989 Section XI		EPRI TR-112657		Weld Count	1989 Section XI		EPRI TR-112657		Weld Count	1989 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)
EN	C-F-1											93	8	0	0	
EP	B-J						32	11	0	4		83	6	11	0	
GS	C-F-2											2	0	0	0	
Total	B-F	1	1	0	0		13	13	0	5						
	B-J	77	14	20	19		395	56	30	38		219	9	41	0	
	C-F-1	4	1	0	3		511	34	3	53		563	45	1	0	
	C-F-2						16	1	0	2		290	27	0	0	

Notes

1. Systems are described in Table 3.1-1.
2. The ASME Code Category is based on the 1989 Edition of the ASME Section XI Code.
3. The column labeled "Other" is generally used to identify augmented inspection program locations that are credited beyond those locations selected per the RI-ISI process, as addressed in Section 3.6.5 of EPRI TR-112657. This option was not applicable for the Wolf Creek Generating Station RI-ISI application. The "Other" column has been retained in this table solely for uniformity purposes with other RI-ISI application template submittals.

Table 5-2

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

System(1)	Risk		Consequence Rank	Failure Potential		Code Category	Weld Count	1989 Section XI		EPRI TR-112657	
	Category	Rank		DMs	Rank			Vol/Sur	Sur Only	RI-ISI	Other(2)
AB	6	Low	Medium	None	Low	C-F-2	154	14	0	0	
AE	5 (3)	Medium (High)	Medium	TASCS, (FAC)	Medium (High)	C-F-2	16	1	0	2	
AE	6 (3)	Low (High)	Medium	None (FAC)	Low (High)	C-F-2	108	11	0	0	
BB	2	High	High	TASCS, TT	Medium	B-J	11	2	0	3	
BB	2	High	High	TASCS	Medium	B-J	6	3	0	2	
BB	2	High	High	TT	Medium	B-F	1	1	0	0	
						B-J	20	3	7	6	
BB	4	Medium	High	None	Low	B-F	13	13	0	5	
						B-J	268	37	14	21	
BB	6	Low	Medium	None	Low	B-J	6	0	2	0	
BB	7	Low	Low	None	Low	B-J	12	0	0	0	
BG	2	High	High	TASCS	Medium	B-J	9	0	2	2	
BG	2	High	High	TT	Medium	B-J	9	0	8	3	
BG	4	Medium	High	None	Low	B-J	26	0	6	3	
						C-F-1	72	7	3	9	
BG	5	Medium	Medium	TT	Medium	B-J	7	0	6	2	
BG	6	Low	Medium	None	Low	B-J	12	0	4	0	
						C-F-1	39	0	0	0	
BG	7	Low	Low	None	Low	C-F-1	18	0	0	0	
BN	4	Medium	High	None	Low	C-F-1	3	0	0	0	
BN	6	Low	Medium	None	Low	C-F-1	122	9	1	0	
EF	6	Low	Medium	None	Low	C-F-2	26	2	0	0	

Table 5-2

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

System	Risk		Consequence Rank	Failure Potential		Code Category	Weld Count	1989 Section XI		EPRI TR-112657	
	Category	Rank		DMs	Rank			Vol/Sur	Sur Only	RI-ISI	Other(2)
EJ	2	High	High	TASCS, TT	Medium	B-J	2	1	0	0	
EJ	2	High	High	TASCS	Medium	B-J	8	1	0	2	
EJ	2	High	High	TT	Medium	B-J	4	4	0	1	
						C-F-1	4	1	0	3	
EJ	4	Medium	High	None	Low	B-J	22	3	0	3	
						C-F-1	434	27	0	44	
EJ	6	Low	Medium	None	Low	B-J	2	1	0	0	
						C-F-1	1	0	0	0	
EJ	7	Low	Low	None	Low	C-F-1	47	9	0	0	
EM	2	High	High	TT	Medium	B-J	8	0	3	0	
EM	4	Medium	High	None	Low	B-J	20	4	0	3	
						C-F-1	2	0	0	0	
EM	5	Medium	Medium	IGSCC	Medium	B-J	20	1	4	2	
EM	6	Low	Medium	None	Low	B-J	104	2	24	0	
						C-F-1	205	17	0	0	
EM	7	Low	Low	None	Low	C-F-1	38	2	0	0	
EN	6	Low	Medium	None	Low	C-F-1	93	8	0	0	
EP	4	Medium	High	None	Low	B-J	20	8	0	2	
EP	5	Medium	Medium	IGSCC	Medium	B-J	12	3	0	2	
EP	6	Low	Medium	None	Low	B-J	83	6	11	0	
GS	7	Low	Low	None	Low	C-F-2	2	0	0	0	

Notes

1. Systems are described in Table 3.1-1.
2. The column labeled "Other" is generally used to identify augmented inspection program locations that are credited beyond those locations selected per the RI-ISI process, as addressed in Section 3.6.5 of EPRI TR-112657. This option was not applicable for the WCGS RI-ISI application. The "Other" column has been retained in this table solely for uniformity purposes with other RI-ISI application template submittals.

Description of Difference Methodology

1. As discussed in the cover letter, the Strategic Teaming and Resource Sharing (STARS) group developed their respective risk-informed inservice inspection (RI-ISI) program plans (referred to as templates from here on) collaboratively (see Note 6).

2. The templates are similar; where there are differences, the difference will be bracketed [ ]. Plant/Licensee names will not be bracketed to ease readability of the template.

3. Information contained in tables and notes is plant specific and will not be bracketed.

4. To allow for comparison of the templates, below is a table correlating plant specific system nomenclature.

	CPSES	STP	Callaway	WCGS	DCPP
Reactor Coolant System	RCS	RCS	BB	BB	RCS
Chemical and Volume Control System	CVCS	CVCS	BG, BN	BG, BN	CVCS
Safety Injection System	SIS	SIS	EM, EP	EM, EP	SIS
Residual Heat Removal System	RHRS	RHRS	EJ	EJ	RHRS
Feedwater System	FWS	FW & AFW	AE	AE	FWS
Main Steam System	MSSS	MSS	AB	AB	MSSS
Containment Spray System	CSS	CSS	EN	EN	CSS
Sludge Lancing System	--	SLS	--	--	--
Essential Service Water System	--	--	EF	EF	--
Containment Hydrogen Control System	--	--	--	GS	--

CPSES - Comanche Peak Steam Electric Station
STP - South Texas Project
Callaway - Callaway Plant
WCGS - Wolf Creek Generating Station
DCPP - Diablo Canyon Power Plant

5. STP Nuclear Operating Company has an approved American Society of Mechanical Engineers (ASME) Code Class 1 RI-ISI program plan. The STP Nuclear Operating Company application is for ASME Code Class 1 piping socket welds and Class 2 piping welds.

6. The following is a discussion on the process used to develop the template.

   The STARS group contracted with Structural Integrity Associates (SIA) to support the development of the RI-ISI templates. SIA was selected based on their previous work in developing the STP Nuclear Operating Company ASME Code Class 1 template and their team of subcontractors. SIA had teamed with Inservice Engineering and Duke Engineering Services Incorporated (DESI). Both subcontractors have experience in developing RI-ISI program plans.

In order to facilitate technology transfer, the STARS group developed the Degradation Mechanism Evaluation and the Consequence Evaluation. The contractor team provided training, oversight, and technical support in the development of the evaluations.

In order to maximize the synergies of these common plants, technical representatives from each of the plants met for 3 weeks at CPSES to develop these evaluations. The Inservice Inspection engineers from each plant met together and developed the plant specific Degradation Mechanism Evaluation. This effort was lead by SIA. Each plants drawings, history, and other applicable data were reviewed by the entire team. Commonalities and differences were discussed; technical issues were resolved and each pipe segment for each plant was subsequently evaluated for potential degradation mechanisms.

Likewise, probablistic risk assessment engineers from each plant met together and developed their plant specific Consequence Evaluation. This effort was lead by DESI. Again, engineers had their plant specific information, which was reviewed before by the entire team. Commonalities and differences were discussed; technical issues were resolved and each event was evaluated for potential consequences.

Inservice Engineering then combined the work of the two groups to develop the template and perform the delta risk calculation.

LIST OF COMMITMENTS

The following table identifies those actions committed to by Wolf Creek Nuclear Operating Corporation (WCNOC) in this document. Any other statements in this submittal are provided for information purposes and are not considered to be commitments. Please direct questions regarding these commitments to Mr. Tony Harris, Manager Regulatory Affairs at Wolf Creek Generating Station, (316) 364-4038.

COMMITMENT	Due Date/Event
WCNOC will closely monitor the progress of and will assess the industry recommendations resulting from the EPRI- Materials Reliability Project evaluation of the V.C. Summer event.	Ongoing
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 second inservice inspection interval.	Prior to RF-12