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NLS2004023
March 11, 2004

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555-0001

Subject: Risk-Informed Inservice Inspection Program (Relief Request RI-34)
Cooper Nuclear Station, NRC Docket No. 50-298, DPR-46

Reference:
1. Electric Power Research Institute Topical Report TR-112657 Revision B-A, "Revised Risk-Informed Inservice Inspection Evaluation Procedure," December 1999.

2. Letter to G. Vine (Electric Power Research Institute) from W. Bateman (U.S. Nuclear Regulatory Commission) dated October 28, 1999, "Safety Evaluation Report Related to EPRI Risk-Informed Inservice Inspection Evaluation Procedure (EPRI TR-112657, Revision B, July 1999)."

The purpose of this letter is to propose a Risk-Informed Inservice Inspection (RI-ISI) Program for Cooper Nuclear Station (CNS). This RI-ISI Program is being submitted as an alternative to existing ASME Boiler and Pressure Vessel Code, Section XI requirements for the selection and examination of Class 1 and 2 piping welds. Relief from the specified Code requirements is requested under the provisions of 10CFR50.55a(a)(3)(i). The implementation of the RI-ISI program will result in a reduction in piping weld examinations, and an associated reduction in occupational radiation exposure, but with little or no change in risk to the public due to piping failure.

Attachment 1 documents Relief Request Number RI-34, which summarizes the CNS RI-ISI Program provided in Enclosure 1 to that attachment. This Program was developed in accordance with Electric Power Research Institute (EPRI) Topical Report TR-112657, Revision B-A (Reference 1). Nuclear Regulatory Commission (NRC) acceptance of the EPRI TR-112657 report as a basis for developing an RI-ISI Program is documented in Reference 2.

CNS is currently in the third inspection period of the third ISI interval. The Nebraska Public Power District (NPPD) plans to implement the RI-ISI Program during the third period to support inspection activities during the next refueling outage (RFO-22). In order to support planning activities associated with RFO-22, NPPD requests NRC approval of the proposed alternative by August 1, 2004.

A047

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Should you have any questions concerning this matter, please contact Mr. Paul Fleming at (402) 825-2774.

Sincerely,

Stewart B. Minahan
Acting Site Vice President
/wrv

cc: Regional Administrator w/attachment
USNRC - Region IV

Senior Project Manager w/attachment
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NPG Distribution w/o attachment

Records w/attachment

ATTACHMENT 1
RELIEF REQUEST NUMBER: RI-34

COMPONENT IDENTIFICATION

Code Classes:	1 and 2
References:	IWB-2500, IWC-2500, Table IWB-2500-1, Table IWC-2500-1
Examination Categories:	B-F, B-J, C-F-1, and C-F-2
Item Numbers:	B5.10, B5.20, B5.130, B5.140, B9.1 0, B9.20, B9.30, B9.40, C5.50, and C5.80.
Description:	Risk-Informed Inservice Inspection (RI-ISI).
Component Numbers:	All Class 1 and 2 pressure retaining piping welds

APPLICABLE CODE EDITION AND ADDENDA

1989 Edition, No Addenda

CODE REQUIREMENT

ASME Section XI (1989 Edition), 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.

ASME Section XI (1989 Edition), 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 require 7.5%, but not less than 28 welds to be selected for examination. Note- Cooper Nuclear Station (CNS) does not have any Category C-F-1 welds.

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

BASIS FOR RELIEF

The scope for ASME Section XI Inservice Inspection (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. Service experience has shown that 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 application of risk-informed selection criteria in order to determine the scope of ISI programs at nuclear power plants. Electric Power Research Institute (EPRI) studies indicate that the application of these techniques will allow operating nuclear plants to reduce the examination scope of current ISI programs by as much as 60% to 80%, significantly reduce costs, and continue to maintain high nuclear plant safety standards.

NPPD has applied the methodology of EPRI Topical Report TR-112657 in the development of the proposed CNS RI-ISI Program (see Enclosure 1 to this Attachment). 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." The use of this methodology for the selection and subsequent examination of Class 1 and 2 piping welds will provide an acceptable level of quality and safety.

Relief is requested in accordance with 10CFR50.55a(a)(3)(i). The Nuclear Regulatory Commission has previously approved several RI-ISI Programs based on methodology contained in EPRI Topical Report TR-112657, Revision B-A. A similar RI-ISI submittal has been recently approved for Salem, Units 1 and 2.[1]

PROPOSED ALTERNATE PROVISIONS

As an alternative to existing ASME Section XI requirements for piping weld selection and examination volumes, NPPD will implement the alternative RI-ISI program described in Enclosure 1.

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1. Letter from J. Clifford (NRC) to R. Anderson (PSEG Nuclear), dated October 1, 2003, TAC NOS. MB7537 and MB7538).

APPLICABLE TIME PERIOD

Approval of this alternative is requested for the remainder of the third ten-year interval of the ISI Program for CNS, beginning with the last outage (RFO 22) of the third period, and for the fourth ten-year ISI interval, which will begin on March 1, 2006.

ENCLOSURE 1

RISK-INFORMED INSERVICE INSPECTION PLAN

COOPER NUCLEAR STATION, REVISION 0

RISK-INFORMED INSERVICE INSPECTION PROGRAM PLAN

COOPER NUCLEAR STATION, REVISION 0

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1. INTRODUCTION

The Cooper Nuclear Station (CNS) is currently in the third inservice inspection (ISI) interval as defined by the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Section XI Code for Inspection Program B. The CNS plans to start implementing a risk-informed inservice inspection (RI-ISI) program during the third inspection period. Initial RI-ISI Program implementation is planned for the plant's twenty second refueling outage (RE22) scheduled for Spring 2005. The ASME Section XI Code of Record for the third ISI interval at the CNS is the 1989 Edition.

The objective of this submittal is to request the use of a risk-informed process for the inservice inspection of Class 1 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 Cooper Nuclear Station Individual Plant Evaluation (CNS IPE) was submitted to the NRC in March 1993. On October 21, 1994 the NRC sent a request for additional information. The questions in the request were addressed in a letter dated February 20, 1995. The NRC responded in a letter dated May 18, 1995 and approved the CNS IPE results. The letter concluded that the CNS IPE met the intent of the GL88-20, identifying plant specific vulnerabilities using the guidance in NUREG-1335.

The CNS IPE consisted of the Level 1 PRA and back-end analysis consistent with GL88-20 requirements. In the NPPD response to GL88-20, it was noted that the PRA study would be considered a living study, in anticipation of model revisions from time to time to reflect changes to procedures, plant operating data, etc.

Several model updates have been completed since the IPE was submitted. The scope of the updates was based on review of results and plant input to the model. The scope of the updates included revisions to system models, refinement of assumptions, and re-quantification of the Level 1 model. These revisions and the final review comments, constituted the CNS PRA 1996b model.

After completing the 1996 update of the Level I PRA, a detailed plant-specific Level 2 model was developed that incorporated the large early release frequency based on the revised results of the Level 1 PRA. The results of the 1998 Level 2 model and 1996b Level 2 are integrated into the updated CNS PRA (1998).

An initial industry peer review of the Cooper Nuclear Station PRA was conducted in July 1997 (published September 1997) with a second industry peer review performed November 2001 (published April 2002). The CNS PRA model is currently being revised to address the comments received from these detailed reviews. This major revision to the PRA will result in a new revision to quantified results and will be reviewed and approved internally prior to release. Although this on-going work is not used in preparation of this submittal, certain conclusions regarding internal flooding were considered qualitatively and reviewed against the most current plant information for potential insights.

The Risk-Informed Inservice Inspection (RI-ISI) consequence evaluation is based on the Cooper Nuclear Station PRA 96b model. The base case Core Damage Frequency (CDF) is 1.3E-05/year, and the LERF is 5.6E-07/year.

The Results Summary of the 2001 BWROG CNS PRA Certification published in April 2002 contains the following statements:

- "All of the PRA elements identified as part of the NEI 00-02 PRA Peer Review process were included in the Cooper PRA. In terms of the overall assessment of each element, all were consistently graded as sufficient to support risk-informed decision-making when combined with deterministic insights (i.e. a blended approach). All elements are judged fully capable of supporting absolute risk determination to support Grade 3 applications when the footnoted items are performed."

- "The average grade level of each of the PRA elements is quite consistent indicating that most PRA elements have been addressed in a manner that could allow supporting applications up to Grade 3 with the incorporation of recommended enhancements or additional deterministic analysis. In terms of the average element scores, areas that stand out as particularly strong are the following:
  - Quantification
  - System Analysis"

- "The areas that provide the greatest opportunities for improvement on a relative basis are the following:
  - Initiating Event Analysis
  - Data Analysis
  - Human Reliability Analysis"

The main comments in the above review were connected with the treatment of the human action dependencies using more recent methods, use of most recent CNS operating data where available and finalizing the most recent draft initiating event analysis document along with development of plant specific support system trip models. It is not expected that these issues would impact the consequence rankings established in the RI-ISI analysis, mainly because the risk importance of the systems in the RI-ISI process is dominated by the LOCA events.

Based on the above, it is judged that the current PRA model, used in the RI-ISI evaluation, has an acceptable quality to support this application.

2. PROPOSED ALTERNATIVE TO CURRENT ISI PROGRAMS

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

- The plant 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.

- The CNS is incorporating the guidance contained in BWR Vessel and Internals Project Report No. BWRVIP-75. BWRVIP-75 provides alternative criteria to NRC Generic Letter 88-01 for the examination of welds susceptible to intergranular stress corrosion cracking (IGSCC). Both Generic Letter 88-01 and BWRVIP-75 specify examination extent and frequency requirements for austenitic stainless steel welds that are classified as Categories A through G, dependent upon their susceptibility to IGSCC. In accordance with EPRI TR-112657, piping welds identified as Category A are considered resistant to IGSCC and are assigned a low failure potential provided no other damage mechanisms are present. As such, the examination of welds identified as Category A inspection locations is subsumed by the RI-ISI Program. The existing plant augmented inspection program for the other piping welds susceptible to IGSCC at the CNS (the CRD return line nozzle cap weld is classified as Category D) remains unaffected by the RI-ISI Program submittal.

- The plant augmented inspection program for feedwater nozzle cracking per NUREG 0619 is implemented per the provisions provided in GE-NE-523-A71-0594 and the associated NRC Safety Evaluation. The feedwater nozzle-to-safe end weld locations are included in the scope of both the NUREG 0619 Program and the RI-ISI Program. The plant augmented inspection program requirements for these locations are not affected or changed by the RI-ISI Program.

3. RISK-INFORMED ISI PROCESS

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

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

A deviation to the EPRI RI-ISI methodology has been implemented in the failure potential assessment for the CNS. 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 locations where stratification is likely to occur, but allows for no assessment of severity. As such, many locations will be identified as subject to TASCS where no significant potential for thermal fatigue exists. The critical attribute missing from the existing methodology that would allow consideration of fatigue severity is a criterion that addresses the potential for fluid cycling. The impact of this additional consideration on the existing TASCS susceptibility criteria is presented below.

- **Turbulent penetration TASCS**

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

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

- **Low flow TASCS**

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

- **Valve leakage TASCS**

  Sometimes a very small leakage flow of hot water can occur outward past a valve into a line that is relatively colder, creating a significant temperature difference. However, since this is 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.

In summary, these additional considerations for determining the potential for thermal fatigue as a result of the effects of TASCS provide an allowance for the consideration of cycle severity. The above criteria have previously been submitted by EPRI for generic approval (Letters dated February 28, 2001 and March 28, 2001, from P.J. O'Regan (EPRI) to Dr. B. Sheron (USNRC), "Extension of Risk-Informed Inservice Inspection Methodology"). The methodology used in the CNS RI-ISI application for assessing TASCS potential conforms to these updated criteria. Final materials reliability program (MRP) guidance on the subject of TASCS will be incorporated into

the CNS RI-ISI application if different than the criteria used. It should be noted that the NRC has granted approval for RI-ISI relief requests incorporating these TASCS criteria at several facilities, including Comanche Peak (SER dated September 28, 2001) and South Texas Project (SER dated March 5, 2002).

3.1 Scope of Program

The systems included in the RI-ISI Program are provided in Table 3.1. The piping and instrumentation diagrams and additional plant information including the existing plant ISI Program were used to define the Class 1 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 (i.e., isolation, bypass and large early release). The consequence evaluation included an assessment of shutdown and external events. The impact on these measures due to both direct and indirect effects was considered using the guidance provided in EPRI TR-112657.

3.3 Failure Potential Assessment

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

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

3.4 Risk Characterization

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

The results of these calculations are presented in Table 3.4.

3.5 Element and NDE Selection

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

For the CNS, the percentage of Class 1 piping welds selected strictly for RI-ISI purposes was 8.8%. It should be noted that this sampling percentage for Class 1 piping locations includes both socket and non-socket welds. If only non-socket welded locations are considered, the percentage of Class 1 piping welds selected for examination increases to 11.3%.

The above sampling percentage does not take credit for any inspection locations selected for examination per the plant's augmented inspection program for FAC beyond those selected per the RI-ISI process. It should be noted that no FAC examinations are being credited to satisfy RI-ISI selection requirements. Inspection locations selected for RI-ISI purposes that are in the FAC Program will be subjected to an independent examination to satisfy the RI-ISI Program requirements.

The only non Category A inspection location selected for examination per the plant's augmented inspection program for IGSCC (Category D) was also selected for RI-ISI purposes to satisfy Risk Category 4 selection requirements.

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

Unit	Class 1 Piping Welds(1)		Class 2 Piping Welds(2)		All Piping Welds(3)	
	Total	Selected	Total	Selected	Total	Selected
1	650	57	930	4	1580	61

Notes

1. Includes all Category B-F and B-J locations.
2. Includes all Category C-F-2 locations. There are no Category C-F-1 piping welds at the CNS.
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 during the current outage. No additional examinations will be performed if there are no additional elements identified as being susceptible to the same root cause conditions.

3.5.2 Program Relief Requests

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

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

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

Relief Request	Relief Request Description
RI-08, Rev. 0[1]	Pertains to the usage of sample expansion criteria per the requirements of Generic Letter 88-01 in lieu of the requirements of IWB-2430.
RI-20, Rev. 1[2]	Pertains to partial surface examination coverage of weld RVD-BF-14.
RI-22, Rev. 0[3]	Pertains to partial volumetric examination coverage of welds FWA-BJ-81, RAS-BJ-10 and RBS-BJ-6A.

Notes

1. *Section 3.5.1 of this template provides the requirements for additional examinations.* These requirements ensure that additional examinations are focused on inspection locations subject to the same root cause conditions or degradation mechanisms. Relief Request RI-08 can be withdrawn.
2. *This weld was scheduled for examination per Section XI in the 3rd period. This weld also is a RI-ISI* selection (risk category 4) and remains scheduled for examination in RE22. However, for RI-ISI purposes, this inspection location will be volumetrically examined. Relief Request RI-20 can be modified or withdrawn dependent upon the results of the upcoming examination.
3. These welds are addressed as follows:
   i. FWA-BJ-81 – This weld was scheduled for examination per Section XI in the 3rd period. However, this weld is not a RI-ISI selection (risk category 6) and the examination will not be performed. Relief Request RI-22 can be modified to remove this weld from consideration.
   ii. RAS-BJ-10 – This weld was examined per Section XI in the 2nd period. This weld is not a RI-ISI selection (risk category 4) and an examination will not be required in future intervals. However, the examination of this weld has been credited for the current 3rd interval. Therefore, this portion of Relief Request RI-22 remains unchanged.
   iii. RBS-BJ-6A – This weld was scheduled for examination per Section XI in the 3rd period. However, this weld is not a RI-ISI selection (risk category 4) and the examination will not be performed. Relief Request RI-22 can be modified to remove this weld from consideration.

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.

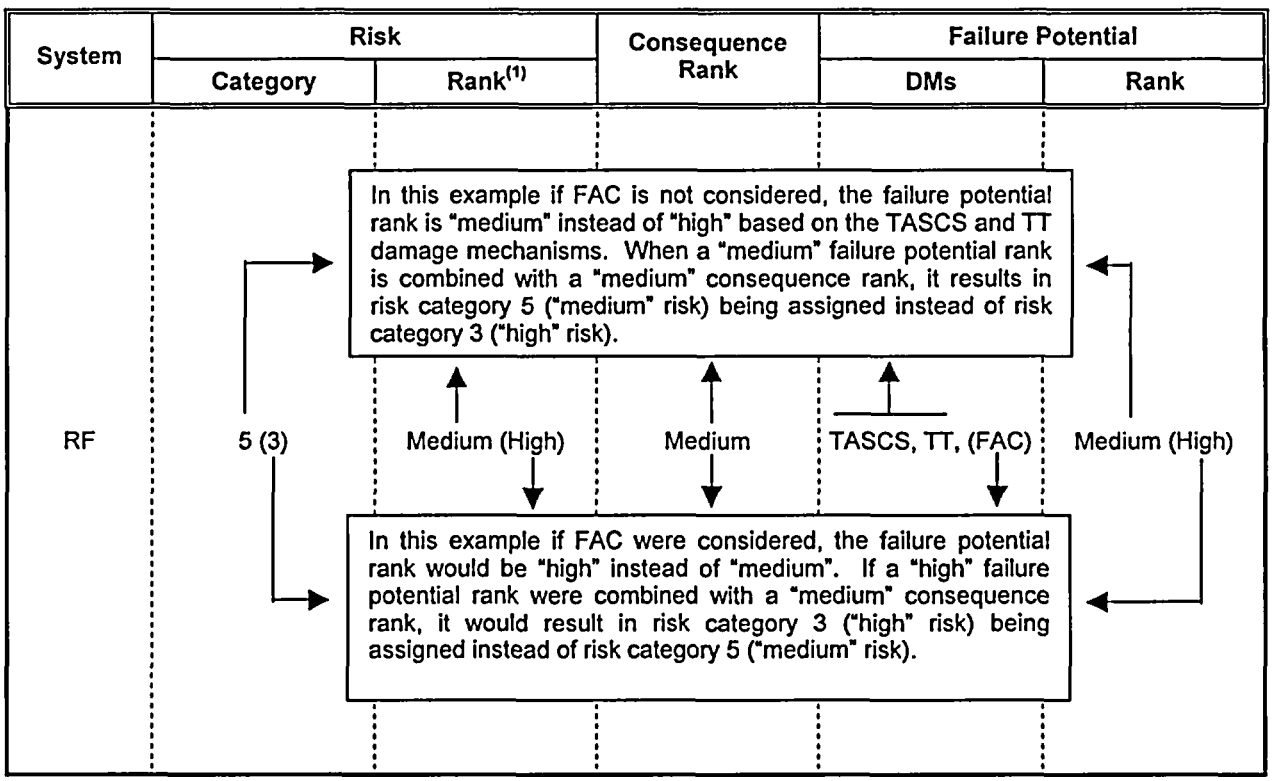
3.6.1 Quantitative Analysis

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.

The CNS 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 (1E-03) and CLERP (1E-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.

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 and IGSCC was adjusted for in the performance of the quantitative analysis by excluding their impact on the risk ranking. The exclusion of the impact of FAC and IGSCC on the risk ranking and therefore in the determination of the change in risk is

performed, because FAC and IGSCC are damage mechanisms managed by separate, independent plant augmented inspection programs. The RI-ISI Program credits and relies upon these plant augmented inspection programs to manage these damage mechanisms. The plant FAC and IGSCC Programs will continue to determine where and when examinations shall be performed. Hence, since the number of FAC and IGSCC examination locations remains the same "before" and "after" and no delta exist, there is no need to include the impact of FAC and IGSCC in the performance of the risk impact analysis. However, in an effort to be as informative as possible, for those systems where FAC or IGSCC is present, Table 3.6-1 presents the information in such a manner as to depict what the resultant risk categorization is both with and without consideration of FAC or IGSCC. This is accomplished by enclosing the FAC or IGSCC damage mechanism, as well as all other resultant corresponding changes (failure potential rank, risk category and risk rank), in parentheses. 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 plant 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 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
NB	5.00E-12	5.00E-12	5.00E-13	5.00E-13
NBDR	-1.50E-11	-1.50E-11	-1.50E-12	-1.50E-12
NBI	-1.00E-11	-1.00E-11	-1.00E-12	-1.00E-12
RR	7.85E-10	7.85E-10	7.85E-11	7.85E-11
RWCU	5.00E-12	5.00E-12	5.00E-13	5.00E-13
RCIC	no change	no change	no change	no change
RHR	3.50E-11	3.50E-11	3.50E-12	3.50E-12
CS	1.20E-10	1.20E-10	1.20E-11	1.20E-11
HPCI	negligible	negligible	negligible	negligible
MS	2.00E-11	6.00E-11	2.00E-12	6.00E-12
MSDR	no change	no change	no change	no change
RF	3.29E-10	3.45E-10	3.29E-11	3.45E-11
SDV	negligible	negligible	negligible	negligible
SLC	-1.50E-11	-1.50E-11	-1.50E-12	-1.50E-12
PNC	negligible	negligible	negligible	negligible
REC	no change	no change	no change	no change
Total	**1.26E-09**	**1.32E-09**	**1.26E-10**	**1.32E-10**

Note

1. Systems are described in Table 3.1.

3.6.2 Defense-in-Depth

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

This process has two key independent ingredients, that is, a determination of each location's susceptibility to degradation and secondly, an independent assessment of the consequence of the piping failure. These two ingredients assure defense in depth is maintained. First, by evaluating a location's

susceptibility to degradation, the likelihood of finding flaws or indications that may be precursors to leak or ruptures is increased. Secondly, the consequence assessment effort has a single failure criterion. As such, no matter how unlikely a failure scenario is, it is ranked High in the consequence assessment, and at worst Medium in the risk assessment (i.e., Risk Category 4), if as a result of the failure there is no mitigative equipment available to respond to the event. In addition, the consequence assessment takes into account equipment reliability, and less credit is given to less reliable equipment.

All locations within the Class 1 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 third 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 1989 Code Edition program requirements for in-scope piping is provided in Tables 5-1 and 5-2. Table 5-1 provides a summary comparison by risk region. Table 5-2 provides the same comparison information, but in a more detailed manner by risk category, similar to the format used in Table 3.6-1.

The CNS intends to start implementing the RI-ISI Program during the plant's twenty second refueling outage (RE22) scheduled for Spring 2005. Beginning with RE22, inspection locations selected per the RI-ISI process will replace those formerly selected per ASME Section XI criteria. By the end of the second period, 55% of the piping weld examinations required by ASME Section XI have been completed thus far in the third ISI interval for Examination Categories B-F, B-J and C-F-2. To ensure the performance of 100% of the required examinations during the current ten-year ISI interval, 45% of the inspection locations selected for examination per the RI-ISI process will be examined in the third period.

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

COOP-08Q-302, "Consequence Evaluation for Cooper Nuclear Station (CNS)", Revision 2, January 8, 2004

COOP-08Q-301, "Degradation Mechanism Evaluation for the Class 1 and Class 2 Piping Welds at Cooper Nuclear Station (CNS)", Revision 2, January 9, 2004

COOP-08Q-303, "Service History Review for Cooper Nuclear Station (CNS)", Revision 1, December 2, 2003

COOP-08Q-304, "Risk Ranking for Cooper Nuclear Station (CNS)", Revision 1, March 8, 2004

COOP-08Q-305, "Minutes of the Element Selection Meeting for the RI-ISI Project at Cooper Nuclear Station (CNS)", Revision 1, December 15, 2003

COOP-08Q-306, "Risk Impact Analysis for Cooper Nuclear Station (CNS)", Revision 1, March 8, 2004

Table 3.1

System Selection and Segment / Element Definition

System Description	Number of Segments	Number of Elements
NB – Nuclear Boiler System	6	6
NBDR – Nuclear Boiler Drain System	3	25
NBI – Nuclear Boiler Instrumentation System	4	22
RR – Reactor Recirculation System	19	114
RWCU – Reactor Water Cleanup System	3	30
RCIC – Reactor Core Isolation Cooling System	4	55
RHR – Residual Heat Removal System	66	588
CS – Core Spray System	28	177
HPCI – High Pressure Coolant Injection System	12	96
MS – Main Steam System	24	260
MSDR – Main Steam Drain System	3	7
RF – Reactor Feedwater System	42	89
SDV – Scram Discharge Volume System	2	40
SLC – Standby Liquid Control System	4	55
PNC – Primary Containment Cooling and Nitrogen Inerting System	6	12
REC – Reactor Equipment Cooling System	2	4
Totals	**228**	**1580**

Table 3.3

Failure Potential Assessment Summary

System[1]	Thermal Fatigue		Stress Corrosion Cracking				Localized Corrosion			Flow Sensitive	
	TASCS	TT	IGSCC	TGSCC	ECSCC	PWSCC	MIC	PIT	CC	E-C	FAC
NB			X								
NBDR											X
NBI											
RR									X		
RWCU											
RCIC											
RHR											
CS									X		
HPCI											
MS		X									
MSDR											
RF		X							X		X
SDV											
SLC											
PNC											
REC											

Note

1. Systems are described in Table 3.1.

Table 3.4

Number of Segments by Risk Category With and Without Impact of FAC and IGSCC

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
NB			1[2]	0			4	5			1	1		
NBDR	1[3]	0					1	2					1	1
NBI							2	2			2	2		
RR			10	10			7	7					2	2
RWCU							2	2					1	1
RCIC											4	4		
RHR							14	14			38	38	14	14
CS			2	2			6	6			6	6	14	14
HPCI											11	11	1	1
MS							4	4	1	1	19	19		
MSDR											2	2	1	1
RF	9[4]	0	4	4	9[5]	0	9	18	1	3	10	17		
SDV													2	2
SLC							1	1			3	3		
PNC													6	6
REC													2	2
Total	10	0	17	16	9	0	50	61	2	4	96	103	44	44

Notes

1. Systems are described in Table 3.1.
2. This segment becomes Category 4 after IGSCC is removed from consideration due to no other damage mechanisms being present.
3. This segment becomes Category 4 after FAC is removed from consideration due to no other damage mechanisms being present.
4. These nine segments become Category 4 after FAC is removed from consideration due to no other damage mechanisms being present.

Notes for Table 3.4 (Cont'd)

5. Of these nine segments, two become Category 5 after FAC is removed from consideration due to the presence of other "medium" failure potential damage mechanisms, and seven become Category 6 after FAC is removed from consideration due to no other damage mechanisms being present.

Table 3.5

Number of Elements Selected for Inspection by Risk Category Excluding Impact of FAC and IGSCC

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
NB							5	2[2]			1	0		
NBDR							23	3					2	0
NBI							16	2			6	0		
RR			10	3			88	9					16	0
RWCU							26	3					4	0
RCIC											55	0		
RHR							69	7			469	0	50	0
CS			2	1			32	4			60	0	83	0
HPCI											93	0	3	0
MS							104	11	35	4	121	0		
MSDR											6	0	1	0
RF			4	1			54	6	3	2	28	0		
SDV													40	0
SLC							22	3			33	0		
PNC													12	0
REC													4	0
Total	0	0	16	5	0	0	439	50	38	6	872	0	215	0

Notes

1. Systems are described in Table 3.1.
2. One of these two piping welds has been selected for examination per Cooper's augmented inspection program for IGSCC (Category D) and is being credited for RI-ISI purposes.

Table 3.6-1

Risk Impact Analysis Results

System[1]	Category	Consequence Rank	Failure Potential		Inspections			CDF Impact[4]		LERF Impact[4]	
			DMs	Rank	SXI[2]	RI-ISI[3]	Delta	w/ POD	w/o POD	w/ POD	w/o POD
NB	4 (2)	High	None (IGSCC)	Low (Medium)	1	1	0	no change	no change	no change	no change
NB	4	High	None	Low	2	1	-1	5.00E-12	5.00E-12	5.00E-13	5.00E-13
NB	6a	Medium	None	Low	1	0	-1	negligible	negligible	negligible	negligible
NB Total								**5.00E-12**	**5.00E-12**	**5.00E-13**	**5.00E-13**
NBDR	4 (1)	High	None (FAC)	Low (High)	0	1	1	-5.00E-12	-5.00E-12	-5.00E-13	-5.00E-13
NBDR	4	High	None	Low	0	2	2	-1.00E-11	-1.00E-11	-1.00E-12	-1.00E-12
NBDR	7a	Low	None	Low	0	0	0	no change	no change	no change	no change
NBDR Total								**-1.50E-11**	**-1.50E-11**	**-1.50E-12**	**-1.50E-12**
NBI	4	High	None	Low	0	2	2	-1.00E-11	-1.00E-11	-1.00E-12	-1.00E-12
NBI	6a	Medium	None	Low	0	0	0	no change	no change	no change	no change
NBI Total								**-1.00E-11**	**-1.00E-11**	**-1.00E-12**	**-1.00E-12**
RR	2	High	CC	Medium	10	3	-7	7.00E-10	7.00E-10	7.00E-11	7.00E-11
RR	4	High	None	Low	26	9	-17	8.50E-11	8.50E-11	8.50E-12	8.50E-12
RR	7a	Low	None	Low	0	0	0	no change	no change	no change	no change
RR Total								**7.85E-10**	**7.85E-10**	**7.85E-11**	**7.85E-11**
RWCU	4	High	None	Low	4	3	-1	5.00E-12	5.00E-12	5.00E-13	5.00E-13
RWCU	7a	Low	None	Low	1	0	-1	negligible	negligible	negligible	negligible
RWCU Total								**5.00E-12**	**5.00E-12**	**5.00E-13**	**5.00E-13**
RCIC	6a	Medium	None	Low	0	0	0	no change	no change	no change	no change
RCIC Total								**no change**	**no change**	**no change**	**no change**
RHR	4	High	None	Low	14	7	-7	3.50E-11	3.50E-11	3.50E-12	3.50E-12
RHR	6a	Medium	None	Low	38	0	-38	negligible	negligible	negligible	negligible
RHR	7a	Low	None	Low	0	0	0	no change	no change	no change	no change
RHR Total								**3.50E-11**	**3.50E-11**	**3.50E-12**	**3.50E-12**

Table 3.6-1 (Cont'd)

Risk Impact Analysis Results

System[1]	Category	Consequence Rank	Failure Potential		Inspections			CDF Impact[4]		LERF Impact[4]	
			DMs	Rank	SXI[2]	RI-ISI[3]	Delta	w/ POD	w/o POD	w/ POD	w/o POD
CS	2	High	CC	Medium	2	1	-1	1.00E-10	1.00E-10	1.00E-11	1.00E-11
CS	4	High	None	Low	8	4	-4	2.00E-11	2.00E-11	2.00E-12	2.00E-12
CS	6a	Medium	None	Low	7	0	-7	negligible	negligible	negligible	negligible
CS	7a	Low	None	Low	4	0	-4	negligible	negligible	negligible	negligible
CS Total								**1.20E-10**	**1.20E-10**	**1.20E-11**	**1.20E-11**
HPCI	6a	Medium	None	Low	8	0	-8	negligible	negligible	negligible	negligible
HPCI	7a	Low	None	Low	0	0	0	no change	no change	no change	no change
HPCI Total								**negligible**	**negligible**	**negligible**	**negligible**
MS	4	High	None	Low	27	11	-16	8.00E-11	8.00E-11	8.00E-12	8.00E-12
MS	5a	Medium	TT	Medium	2	4	2	-6.00E-11	-2.00E-11	-6.00E-12	-2.00E-12
MS	6a	Medium	None	Low	9	0	-9	negligible	negligible	negligible	negligible
MS Total								**2.00E-11**	**6.00E-11**	**2.00E-12**	**6.00E-12**
MSDR	6a	Medium	None	Low	0	0	0	no change	no change	no change	no change
MSDR	7a	Low	None	Low	0	0	0	no change	no change	no change	no change
MSDR Total								**no change**	**no change**	**no change**	**no change**
RF	2	High	CC	Medium	4	1	-3	3.00E-10	3.00E-10	3.00E-11	3.00E-11
RF	4 (1)	High	None (FAC)	Low (High)	8	3	-5	2.50E-11	2.50E-11	2.50E-12	2.50E-12
RF	4	High	None	Low	11	3	-8	4.00E-11	4.00E-11	4.00E-12	4.00E-12
RF	5a (3)	Medium	TT, (FAC)	Medium (High)	0	1	1	-1.80E-11	-1.00E-11	-1.80E-12	-1.00E-12
RF	5a	Medium	TT	Medium	0	1	1	-1.80E-11	-1.00E-11	-1.80E-12	-1.00E-12
RF	6a (3)	Medium	None (FAC)	Low (High)	2	0	-2	negligible	negligible	negligible	negligible
RF	6a	Medium	None	Low	4	0	-4	negligible	negligible	negligible	negligible
RF Total								**3.29E-10**	**3.45E-10**	**3.29E-11**	**3.45E-11**

Table 3.6-1 (Cont'd)

Risk Impact Analysis Results

System[1]	Category	Consequence Rank	Failure Potential		Inspections			CDF Impact[4]		LERF Impact[4]	
			DMs	Rank	SXI[2]	RI-ISI[3]	Delta	w/ POD	w/o POD	w/ POD	w/o POD
SDV	7a	Low	None	Low	3	0	-3	negligible	negligible	negligible	negligible
SDV Total								**negligible**	**negligible**	**negligible**	**negligible**
SLC	4	High	None	Low	0	3	3	-1.50E-11	-1.50E-11	-1.50E-12	-1.50E-12
SLC	6a	Medium	None	Low	0	0	0	no change	no change	no change	no change
SLC Total								**-1.50E-11**	**-1.50E-11**	**-1.50E-12**	**-1.50E-12**
PNC	7a	Low	None	Low	1	0	-1	negligible	negligible	negligible	negligible
PNC Total								**negligible**	**negligible**	**negligible**	**negligible**
REC	7a	Low	None	Low	0	0	0	no change	no change	no change	no change
REC Total								**no change**	**no change**	**no change**	**no change**
Grand Total								**1.26E-09**	**1.32E-09**	**1.26E-10**	**1.32E-10**

Notes

1. Systems are described in Table 3.1.
2. Only those ASME Section XI Code inspection locations that received a volumetric examination in addition to a surface examination are included in the count. Inspection locations previously subjected to a surface examination only were not considered in accordance with Section 3.7.1 of EPRI TR-112657.
3. Inspection locations selected for RI-ISI purposes that are in the plant's augmented inspection programs for flow accelerated corrosion (FAC) and intergranular stress corrosion cracking (IGSCC) are subject to the following requirements dependent upon risk categorization:
   i. Risk Categories 2 (1) and 5 (3) – these inspection locations are susceptible to medium failure potential damage mechanisms in addition to FAC. In these cases, inspection locations selected for examination by both the FAC and RI-ISI Programs may be included in the RI-ISI count, provided the ultrasonic thickness measurement performed for FAC is judged inadequate to have detected the other damage mechanisms subsequently identified by the RI-ISI Program. For the CNS RI-ISI application, the risk category 5 (3) inspection location [risk category 2 (1) does not exists] selected for examination per the plant's augmented inspection program for FAC that was selected for RI-ISI purposes was not credited in detecting the presence of other damage mechanisms (e.g., thermal fatigue).
   ii. Risk Categories 2 (2) and 5 (5) – these inspection locations are susceptible to other medium failure potential damage mechanisms in addition to IGSCC. In these cases, inspection locations selected for examination by both the IGSCC and RI-ISI Programs should be included in both counts, but only those locations that were previously being credited in the Section XI Program and are now being credited in the RI-ISI Program. The examination performed for IGSCC is judged adequate to have detected the other damage mechanisms subsequently identified by the RI-ISI Program. For the CNS RI-ISI application, these risk category combinations do not exist, and this requirement is therefore not applicable.
   iii. Risk Category 4 (1) – these inspection locations are susceptible to FAC only. In these cases, inspection locations selected for examination by both the FAC and RI-ISI Programs should not be included in the RI-ISI count since they do not represent additional examinations. For the CNS RI-ISI application, no inspection locations selected for examination per the plant's augmented inspection program for FAC are being credited for RI-ISI purposes.

Notes for Table 3.6-1 (Cont'd)

iv. Risk Category 4 (2) – these inspection locations are susceptible to IGSCC only. In these cases, inspection locations selected for examination by both the IGSCC and RI-ISI Programs should be included in both counts, but only those locations that were previously credited in the Section XI Program and are now being credited in the RI-ISI Program. For the CNS RI-ISI application, one risk category 4 (2) inspection location was selected for examination per the plant's augmented inspection program for IGSCC and is being credited for RI-ISI purposes. This inspection location was previously credited in the Section XI Program.

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

Inspection Location Selection Comparison Between 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	Section XI		EPRI TR-112657		Weld Count	Section XI		EPRI TR-112657		Weld Count	Section XI		EPRI TR-112657	
			Vol/Sur	Sur Only	RI-ISI	Other[2]		Vol/Sur	Sur Only	RI-ISI	Other[2]		Vol/Sur	Sur Only	RI-ISI	Other[2]
NB	B-F						3	3	0	2[3]						
	B-J						2	0	0	0		1	1	0	0	
NBDR	B-F						1	0	1	0						
	B-J						22	0	6	3		2	0	1	0	
NBI	B-F						2	0	2	2		2	0	2	0	
	B-J						14	0	2	0		4	0	1	0	
RR	B-F	10	10	0	3		5	5	0	3						
	B-J						83	21	3	6		16	0	1	0	
RWCU	B-J						26	4	6	3		4	1	0	0	
RCIC	C-F-2											55	0	0	0	
RHR	B-J						68	14	1	7		35	0	2	0	
	C-F-2						1	0	0	0		484	38	1	0	
CS	B-F	2	2	0	1		2	2	0	1						
	B-J						30	6	3	3		14	0	3	0	
	C-F-2											129	11	0	0	
HPCI	B-J											2	0	0	0	
	C-F-2											94	8	0	0	
MS	B-J						104	27	1	11		45	4	6	0	
	C-F-2						35	2	1	4		76	5	0	0	
MSDR	B-J											7	0	2	0	
RF	B-J	4	4	0	1		57	19	0	8		28	6	0	0	
SDV	C-F-2											40	3	0	0	

Table 5-1 (Cont'd)

Inspection Location Selection Comparison Between 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	Section XI		EPRI TR-112657		Weld Count	Section XI		EPRI TR-112657		Weld Count	Section XI		EPRI TR-112657	
			Vol/Sur	Sur Only	RI-ISI	Other[2]		Vol/Sur	Sur Only	RI-ISI	Other[2]		Vol/Sur	Sur Only	RI-ISI	Other[2]
SLC	B-F						1	0	1	1						
	B-J						21	0	5	2		33	0	9	0	
PNC	C-F-2											12	1	0	0	
REC	C-F-2											4	0	0	0	
Total	B-F	12	12	0	4		14	10	4	9		2	0	2	0	
	B-J	4	4	0	1		427	91	27	43		191	12	25	0	
	C-F-2						36	2	1	4		894	66	1	0	

Notes

1. Systems are described in Table 3.1.
2. The column labeled "Other" is generally used to identify plant augmented inspection program locations credited per Section 3.6.5 of EPRI TR-112657. The EPRI methodology allows plant augmented inspection program locations to be credited if the inspection locations selected strictly for RI-ISI purposes produce less than a 10% sampling of the overall Class 1 weld population. As stated in Section 3.5 of this template, the CNS achieved an 8.8% sampling without relying on plant augmented inspection program locations beyond those selected for RI-ISI purposes either due to the presence of other damage mechanisms, or to satisfy Risk Category 4 selection requirements. The "Other" column has been retained in this table solely for uniformity purposes with the other RI-ISI application template submittals.
3. One of these two piping welds has been selected for examination per Cooper's augmented inspection program for IGSCC (Category D) and is being credited for RI-ISI purposes.

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

System[1]	Risk		Consequence Rank	Failure Potential		Code Category	Weld Count	Section XI		EPRI TR-112657	
	Category	Rank		DMs	Rank			Vol/Sur	Sur Only	RI-ISI	Other[2]
NB	4 (2)	Medium (High)	High	None (IGSCC)	Low (Medium)	B-F	1	1	0	1[3]	
NB	4	Medium	High	None	Low	B-F	2	2	0	1	
						B-J	2	0	0	0	
NB	6a	Low	Medium	None	Low	B-J	1	1	0	0	
NBDR	4 (1)	Medium (High)	High	None (FAC)	Low (High)	B-J	6	0	0	1	
NBDR	4	Medium	High	None	Low	B-F	1	0	1	0	
						B-J	16	0	6	2	
NBDR	7a	Low	Low	None	Low	B-J	2	0	1	0	
NBI	4	Medium	High	None	Low	B-F	2	0	2	2	
						B-J	14	0	2	0	
NBI	6a	Low	Medium	None	Low	B-F	2	0	2	0	
						B-J	4	0	1	0	
RR	2	High	High	CC	Medium	B-F	10	10	0	3	
RR	4	Medium	High	None	Low	B-F	5	5	0	3	
						B-J	83	21	3	6	
RR	7a	Low	Low	None	Low	B-J	16	0	1	0	
RWCU	4	Medium	High	None	Low	B-J	26	4	6	3	
RWCU	7a	Low	Low	None	Low	B-J	4	1	0	0	
RCIC	6a	Low	Medium	None	Low	C-F-2	55	0	0	0	
RHR	4	Medium	High	None	Low	B-J	68	14	1	7	
						C-F-2	1	0	0	0	
RHR	6a	Low	Medium	None	Low	B-J	15	0	1	0	
						C-F-2	454	38	1	0	
RHR	7a	Low	Low	None	Low	B-J	20	0	1	0	
						C-F-2	30	0	0	0	

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

System[1]	Risk		Consequence Rank	Failure Potential		Code Category	Weld Count	Section XI		EPRI TR-112657	
	Category	Rank		DMs	Rank			Vol/Sur	Sur Only	RI-ISI	Other[2]
CS	2	High	High	CC	Medium	B-F	2	2	0	1	
CS	4	Medium	High	None	Low	B-F	2	2	0	1	
						B-J	30	6	3	3	
CS	6a	Low	Medium	None	Low	C-F-2	60	7	0	0	
CS	7a	Low	Low	None	Low	B-J	14	0	3	0	
						C-F-2	69	4	0	0	
HPCI	6a	Low	Medium	None	Low	B-J	2	0	0	0	
						C-F-2	91	8	0	0	
HPCI	7a	Low	Low	None	Low	C-F-2	3	0	0	0	
MS	4	Medium	High	None	Low	B-J	104	27	1	11	
MS	5a	Medium	Medium	TT	Medium	C-F-2	35	2	1	4	
MS	6a	Low	Medium	None	Low	B-J	45	4	6	0	
						C-F-2	76	5	0	0	
MSDR	6a	Low	Medium	None	Low	B-J	6	0	2	0	
MSDR	7a	Low	Low	None	Low	B-J	1	0	0	0	
RF	2	High	High	CC	Medium	B-J	4	4	0	1	
RF	4 (1)	Medium (High)	High	None (FAC)	Low (High)	B-J	30	8	0	3	
RF	4	Medium	High	None	Low	B-J	24	11	0	3	
RF	5a (3)	Medium (High)	Medium	TT, (FAC)	Medium (High)	B-J	2	0	0	1	
RF	5a	Medium	Medium	TT	Medium	B-J	1	0	0	1	
RF	6a (3)	Low (High)	Medium	None (FAC)	Low (High)	B-J	13	2	0	0	
RF	6a	Low	Medium	None	Low	B-J	15	4	0	0	
SDV	7a	Low	Low	None	Low	C-F-2	40	3	0	0	

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

System[1]	Risk		Consequence Rank	Failure Potential		Code Category	Weld Count	Section XI		EPRI TR-112657	
	Category	Rank		DMs	Rank			Vol/Sur	Sur Only	RI-ISI	Other[2]
SLC	4	Medium	High	None	Low	B-F	1	0	1	1	
						B-J	21	0	5	2	
SLC	6a	Low	Medium	None	Low	B-J	33	0	9	0	
PNC	7a	Low	Low	None	Low	C-F-2	12	1	0	0	
REC	7a	Low	Low	None	Low	C-F-2	4	0	0	0	

Notes

1. Systems are described in Table 3.1.
2. The column labeled "Other" is generally used to identify plant augmented inspection program locations credited per Section 3.6.5 of EPRI TR-112657. The EPRI methodology allows plant augmented inspection program locations to be credited if the inspection locations selected strictly for RI-ISI purposes produce less than a 10% sampling of the overall Class 1 weld population. As stated in Section 3.5 of this template, the CNS achieved an 8.8% sampling without relying on plant augmented inspection program locations beyond those selected for RI-ISI purposes either due to the presence of other damage mechanisms, or to satisfy Risk Category 4 selection requirements. The "Other" column has been retained in this table solely for uniformity purposes with the other RI-ISI application template submittals.
3. This piping weld has been selected for examination per Cooper's augmented inspection program for IGSCC (Category D) and is being credited for RI-ISI purposes.

Correspondence Number: NLS2004023

The following table identifies those actions committed to by Nebraska Public Power District (NPPD) in this document. Any other actions discussed in the submittal represent intended or planned actions by NPPD. They are described for information only and are not regulatory commitments. Please notify the Licensing & Regulatory Affairs Manager at Cooper Nuclear Station of any questions regarding this document or any associated regulatory commitments.

COMMITMENT	COMMITTED DATE OR OUTAGE
None	