Nebraska Public Power District
Always there when you need us

10 CFR 50.55a

NLS2006015
February 23, 2006

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

Subject: 10 CFR 50.55a Requests for Fourth Ten-Year Inservice Inspection Interval
Cooper Nuclear Station, Docket No. 50-298, DPR-46

The purpose of this letter is to request that the Nuclear Regulatory Commission (NRC) grant the Nebraska Public Power District (NPPD) relief from, and authorize alternatives to, certain inservice inspection (ISI) code requirements for the Cooper Nuclear Station (CNS) pursuant to 10 CFR 50.55a. The 10 CFR 50.55a requests pertain to both inservice examination and system pressure test requirements in Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. The applicable ASME Code for the fourth ten-year interval is the 2001 Edition, 2003 Addenda. These requests are applicable to the fourth ten-year ISI interval, which commences on March 1, 2006. In order to support planning for Refueling Outage 23 (tentatively scheduled to commence October 21, 2006), NPPD requests approval of these requests by September 1, 2006.

Requests that were approved for the third ten-year interval are being resubmitted, as applicable (RI-02, RI-05, RI-13, RI-15, RI-34, PR-02, PR-04, and PR-06), for NRC review and approval for the fourth ten-year interval. Request Number PR-11 is a new request, and approval of RI-34 will include approval of Revision 1 of the enclosed CNS "Risk-Informed Inservice Inspection Program Plan." Attachment 1 contains a summary listing of the changes for the fourth ten-year interval. Attachment 2 contains the inservice examination requests, and Attachment 3 contains system pressure test requests.

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

Sincerely,

Randall K. Edington

Randall K. Edington
Vice President - Nuclear and
Chief Nuclear Officer

A047

COOPER NUCLEAR STATION
P.O. Box 98 / Brownville, NE 68321-0098
***Telephone:** (402) 825-3811 / **Fax:** (402) 825-5211*
www.nppd.com

/sl

Attachments
Enclosure

cc: U.S. Nuclear Regulatory Commission w/attachments and Enclosure
Regional Office - Region IV

Senior Project Manager w/attachments and Enclosure
USNRC - NRR Project Directorate IV-1

Senior Resident Inspector w/attachments and Enclosure
USNRC - CNS

NPG Distribution w/o attachments and Enclosure

CNS Records w/attachments and Enclosure

Cooper Nuclear Station Inservice Inspection Program
Summary of Changes for the Fourth Ten-Year Interval 10 CFR 50.55a Requests

Third Ten-Year Interval Requests	Fourth Ten-Year Interval Requests	Comments – Fourth Ten-Year Interval Requests	Page Numbers
Attachment 2, Inservice Examination Requests			
RI-02, Rev. 1 – Use of Existing Calibration Blocks for Ultrasonic Examination of Class 1 and Class 2 Components – Relief approved per NRC SER dated October 23, 1997 (TAC No. M94000).	**RI-02, Rev. 0** – Reformatted using the new Nuclear Energy Institute (NEI) guideline (NEI White Paper, Revision 1, "Standard Format for Requests from Commercial Reactor Licensees Pursuant to 10 CFR 50.55a," dated June 2004) and updated the request to include the new code references.	The circumstances and basis for the previous NRC approval of this request have not changed.	1 thru 4
RI-05, Rev. 0 – Inspection of Residual Heat Removal (RHR) Heat Exchanger Shell Welds – Relief approved per NRC SER dated October 23, 1997 (TAC No. M94000).	**RI-05, Rev. 0** – Reformatted using the new NEI guideline and updated the request to include the new code references.	This request was submitted in the third interval describing inservice inspection weld components RHR-CA-3A and -3B for RHR A&B Heat Exchangers as "tubesheet-to-shell" welds when the welds are actually "shell-to-shell" welds. Since the code requirements are the same whether these welds are defined as C1.10 or C1.30, it is believed that the basis for which the NRC previously granted relief would still be valid.	5 thru 8

Cooper Nuclear Station Inservice Inspection Program
Summary of Changes for the Fourth Ten-Year Interval 10 CFR 50.55a Requests
(Continued)

Third Ten-Year Interval Requests	Fourth Ten-Year Interval Requests	Comments – Fourth Ten-Year Interval Requests	Page Numbers
Attachment 2, Inservice Examination Requests			
RI-13, Rev. 2 – Examination and Testing of Class 1, 2, and 3 Snubbers - Relief approved per NRC SER dated March 11, 1999 (TAC No. MA2138).	**RI-13, Rev. 0** – Reformatted using the new NEI guideline and updated the request to include the new code references.	The circumstances and basis for the previous NRC approval of this request have not changed.	9 thru 15
RI-15, Rev. 0 – Examination of Peripheral Control Rod Drive Housing Welds - Relief approved per NRC SER dated October 23, 1997 (TAC No. M9400).	**RI-15, Rev. 0** – Reformatted using the new NEI guideline and updated the request to include the new code references.	The circumstances and basis for the previous NRC approval of this request have not changed.	16 thru 18
RI-34, Rev. 0 – Risk-Informed Inservice Inspection - Relief approved per NRC SER dated December 9, 2004 (TAC MC2351).	**RI-34, Rev. 0** – Reformatted using the new NEI guideline and updated the request to include the new code references.	The circumstances and basis for the previous NRC approval of this request have not changed.	19 thru 22
Attachment 3, System Pressure Test Requests			
PR-02, Rev. 0 – Definition of Pressure Retaining Boundary for System Leakage Test - Relief approved per NRC SER dated October 23, 1997 (TAC No. M94000).	**PR-02, Rev. 0** – Reformatted using the new NEI guideline and updated the request to include the new code references.	The circumstances and basis for the previous NRC approval of this request have not changed.	1 thru 4

Cooper Nuclear Station Inservice Inspection Program
Summary of Changes for the Fourth Ten-Year Interval 10 CFR 50.55a Requests
(Continued)

Third Ten-Year Interval Requests	Fourth Ten-Year Interval Requests	Comments – Fourth Ten-Year Interval Requests	Page Numbers
Attachment 3, System Pressure Test Requests			
PR-04, Rev. 1 – Exemption from Pressure Testing Reactor Vessel Head Flange Seal Leak Detection System - Relief approved per NRC SER dated February 24, 2000 (TAC No. MA5090).	**PR-04, Rev. 0** – Reformatted using the new NEI guideline and updated the request to include the new code references.	The circumstances and basis for the previous NRC approval of this request have not changed.	5 thru 8
PR-06, Rev. 0 – Alternate Pressure Testing for Buried Components - Relief approved per NRC SER dated October 23, 1997 (TAC No. M94000).	**PR-06, Rev. 0** – Reformatted using the new NEI guideline and updated the request to include the new code references.	The circumstances and basis for the previous NRC approval of this request have not changed.	9 thru 11
	New Request PR-11, Rev. 0 – System Pressure Test Boundary	A similar request has been approved by the NRC for the Fitzpatrick Nuclear Power Plant (TAC No. MC7207).	12 thru 13

10 CFR 50.55a Request Number RI-02
Use of Existing Calibration Blocks for Ultrasonic Examination of Class 1 and Class 2 Components

Proposed Alternative in Accordance with 10 CFR 50.55a(a)(3)(ii)
Hardship or Unusual Difficulty without Compensating Increase in Level of Quality or Safety

ASME Code Component(s) Affected

Code Classes:	1 and 2
Examination Categories:	B-A, B-F, B-G-1, B-J, C-A, C-B, C-F-1, C-F-2
Item Numbers:	B1.11, B1.12, B1.21, B1.22, B1.30, B1.40, B5.10, B6.40, B9.11, B9.31, C1.10, C1.20, C1.30, C2.21, C5.11, and C5.51
Component Numbers:	Various

Applicable Code Edition and Addenda

ASME Code Section XI, 2001 Edition, 2003 Addenda

Applicable Code Requirement

Appendix III, Paragraph III-3411 as Supplemented by Table I-2000-1

Article IWA-2000, "Examination and Inspection," Paragraph IWA-2232, "Ultrasonic Examination," states that *Ultrasonic examination shall be conducted in accordance with Appendix I.*

Mandatory Appendix I, "Ultrasonic Examinations," Article I-2000, "Examination Requirements," Subarticle I-2200, "Vessels Not Greater than 2 in. (51mm) in Thickness and all Piping Welds," states the following:

> Paragraph I-2210, "Vessels," states that *Ultrasonic examination of vessels not greater than 2 in. in thickness shall be conducted in accordance with Appendix III, as supplemented by Table I-2000-1.*
>
> Paragraph I-2220, "Welds in Piping," states that *Ultrasonic examination procedures, equipment, and personnel used to detect and size flaws in piping welds shall be qualified by performance demonstration in accordance with Appendix VIII and no other I-2000 requirements apply.*

10 CFR 50.55a Request Number RI-02 (Continued)
Use of Existing Calibration Blocks for Ultrasonic Examination of Class 1 and Class 2 Components

Appendix I, Supplement 1, "Calibration Block Material and Thickness," states that:

(a) The material from which the blocks are fabricated shall be one of the following:
(1) a nozzle dropout from the component;
(2) a component prolongation; or
(3) material of the same material specification, product form, and heat treatment condition as one of the materials being joined.
(b) Where two or more base material thicknesses are involved, the calibration block thickness shall be of a size sufficient to contain the entire examination path.

Appendix III, "Ultrasonic Examination of Vessels Not Greater than 2 Inches (51 mm) in Thickness," Article III-3000, "Calibration," Subarticle III-3400, "Basic Calibration Blocks," Paragraph III-3411, "Material Specification," states:

(a) The calibration blocks for similar metal welds shall be fabricated from one of the materials being joined by the weld.
(b) Calibration blocks for dissimilar welds shall be fabricated from the material specified for the side of the weld from which the examination will be conducted. If the examination will be conducted from both sides, calibration reflectors shall be provided in both materials.
(c) Where the examination is to be performed from only one side of the joint, the calibration block material shall be of the same specification as the material on that side of the joint.
(d) If material of the same specification is not available, material of similar chemical analysis, tensile properties, and metallurgical structure may be used.
(e) When the component material is clad, and the cladding is determined to be important to the examination, the block shall be clad by the same welding procedure as the production part. When the automatic method is impractical, a manual method shall be used.

Reason for Request

Pursuant to 10 CFR 50.55a, "Codes and Standards," Paragraph (a)(3), relief is requested from the requirements of ASME Code Section XI, Appendix III, Paragraph III-3411, as supplemented by Table I-2000-1, for several of the calibration blocks currently being used at the Cooper Nuclear Station (CNS) that lack the documentation necessary to demonstrate compliance with the material specification requirements of Appendix III, Paragraph III-3411, as supplemented by Table I-2000-1. This is because the documentation requirements existing at the time of the fabrication of the blocks did not require traceability to the material's chemical or physical certifications. Consequently, the only documentation available for these existing calibration blocks is verification of the appropriate P-number grouping.

10 CFR 50.55a Request Number RI-02 (Continued)
Use of Existing Calibration Blocks for Ultrasonic Examination of Class 1 and Class 2 Components

It would be a hardship or unusual difficulty without a compensating increase in the level of quality or safety to fabricate a new set of calibration blocks in order to satisfy the documentation requirements of the current code.

Proposed Alternative and Basis for Use

All future calibration blocks will meet the material specification requirements of ASME Section XI, Appendix III, Paragraph III-3411, as supplemented by Table I-2000-1, and will be provided with the documentation necessary to demonstrate compliance with these requirements. Additionally, when using existing calibration blocks that lack the appropriate documentation, acoustic similarity comparisons will be made between the attenuation of the calibration blocks and the material velocity of the materials being examined. This additional comparison will provide adequate assurance that the existing blocks will provide the proper ultrasonic calibration and sensitivity. Existing records which indicate the appropriate P-number grouping will provide reasonable assurance of structural integrity.

Using the provisions of this 10 CFR 50.55a request as an alternative to the specific requirements of Appendix III, Paragraph III-3411, as supplemented by Table I-2000-1, identified above, will continue to provide an acceptable level of quality and safety. Therefore, the Nebraska Public Power District (NPPD) requests authorization to use this alternative in lieu of the ASME Section XI, Appendix III, Paragraph III-3411, as supplemented by Table I-2000-1, requirements for calibration block material specifications in order to allow the continued use of the existing calibration blocks.

Duration of Proposed Alternative

This proposed alternative will be used for the entire fourth ten-year interval of the Inservice Inspection Program for Cooper Nuclear Station (CNS).

Precedents

RI-02, Revision 1, was previously approved by the NRC for the third ten-year interval for CNS on October 23, 1997 (TAC No. M94000).

10 CFR 50.55a Request Number RI-02 (Continued)
Use of Existing Calibration Blocks for Ultrasonic Examination of Class 1 and Class 2 Components

References

1. NPPD Letter NLS950157 to USNRC, "Third Ten-Year Interval Inservice Inspection Program," dated October 18, 1995.

2. USNRC letter to NPPD, "Cooper Nuclear Station - Request for Additional Information Regarding Third Ten-Year Interval Inservice Inspection Program (TAC No. M94000)," dated February 8, 1996.

3. NPPD Letter NLS960050 to USNRC, "Response to Request for Additional Information and Submittal of Revision 1 to the Third Ten-Year Interval Inservice Inspection Program," dated April 11, 1996.

10 CFR 50.55a Request Number RI-05
Inspection of RHR Heat Exchanger Shell Welds

Relief Request in Accordance with 10 CFR 50.55a(g)(5)(iii)
Inservice Inspection Impracticality

ASME Code Component(s) Affected

Code Class:	2
Examination Category:	C-A
Item Number:	C1.10
Component Numbers:	RHR Heat Exchanger 1A, Weld No. RHR-CA-3A RHR Heat Exchanger 1B, Weld No. RHR-CA-3B

Applicable Code Edition and Addenda

ASME Code Section XI, 2001 Edition, 2003 Addenda

Applicable Code Requirement

Table IWC-2500-1

ASME Section XI, 2001 Edition, 2003 Addenda, Article IWC-2000, "Examination and Inspection," Subarticle IWC-2500, "Examination and Pressure Test Requirements," states that (a) *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.*

ASME Section XI, 2001 Edition, 2003 Addenda, Table IWC-2500-1, "Examination Categories," Examination Category C-A, "Pressure Retaining Welds in Pressure Vessels," Item No. C1.10, requires a volumetric examination to be performed on heat exchanger shell circumferential welds each inspection interval by examination requirements/Figure No. IWC-2500-1, "Vessel Circumferential Welds."

10 CFR 50.55a(b)(2)(xix), "Substitution of Alternative Methods," states that *The provisions for the substitution of alternative examination methods, a combination of methods, or newly developed techniques in the 1997 Addenda of IWA-2240 must be applied. The provisions in IWA-2240, 1998 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section are not approved for use.*

10 CFR 50.55a Request Number RI-05 (Continued)
Inspection of RHR Heat Exchanger Shell Welds

Section XI, 1997 Addenda, Article IWA-2000, "Examination and Inspection," Subarticle IWA-2200, "Examination Methods," Paragraph IWA-2240, "Alternate Examinations," states that *Alternate examination methods, a combination of methods, or newly developed techniques may be substituted for the methods specified in this Division, provided the Inspector is satisfied that the results are demonstrated to be equivalent or superior to those of the specified method.*

Impracticality of Compliance

Relief is requested from the requirements of ASME Code Section XI, Table IWC-2500-1, because the Residual Heat Removal (RHR) heat exchanger shell welds as shown in Figure RI-05.1, "RHR Heat Exchanger Distributor Ring-to-Shell Weld Detail" (located at the end of this relief request), are designed with a geometry that provides a corner trap for ultrasonic signals and has limited accessibility. The geometric reflectors inherent in this design prevent a meaningful ultrasonic examination from being performed on these welds.

An investigation into the feasibility of performing ultrasonic examinations on the subject welds was conducted during the second ten-year interval of the Inservice Inspection (ISI) Program for CNS. Various ultrasonic examination techniques tried during the investigation concluded that a meaningful ultrasonic examination could not be performed on this joint configuration. The investigation determined that the distributor ring-to-shell weld configuration was not accessible for performing either a volumetric or surface examination. As a result, CNS applied for specific relief from the examination requirements of Table IWC-2500-1, which was granted by the NRC for the second and third inspection intervals.

Burden Caused by Compliance

To comply with the code-required examinations of the welds, the component would have to be redesigned or disassembled.

Proposed Alternative and Basis for Use

As an alternative to the code-required examination, CNS will perform a visual examination, VT-1, of the applicable welds each inspection interval. Additionally, a visual examination, VT-2, at the required frequency specified by Table IWC-2500-1, Category C-H, will be performed on the shell side of the heat exchanger.

Using the provisions of this relief request as an alternative to the specific requirements of Table IWC-2500-1, identified above, will continue to provide reasonable assurance of structural integrity of the welds. Therefore, pursuant to 10 CFR 50.55a(g)(5)(iii), NPPD requests relief from the ASME Section XI examination requirements for performing a volumetric examination of these RHR heat exchanger shell welds.

10 CFR 50.55a Request Number RI-05 (Continued)
Inspection of RHR Heat Exchanger Shell Welds

Duration of Proposed Alternative

The proposed alternative will be used for the entire fourth ten-year interval of the Inservice Inspection Program for CNS.

Precedents

RI-05, Revision 0, was previously approved by the NRC for the third ten-year interval for CNS on October 23, 1997 (TAC No. M94000).

References

1. NPPD Letter NLS950119 to USNRC, "Second Ten-Year Interval Inservice Inspection Relief Requests," dated June 21, 1995.

2. NPPD Letter NLS950157 to USNRC, "Third Ten-Year Interval Inservice Inspection Program," dated October 18, 1995.

3. USNRC letter to NPPD, "Cooper Nuclear Station - Request for Additional Information Regarding Third Ten-Year Interval Inservice Inspection Program," dated February 8, 1996 (TAC No. M94000).

4. NPPD Letter NLS960050 to USNRC, "Response to Request for Additional Information and Submittal of Revision 1 to the Third Ten-Year Interval Inservice Inspection Program," dated April 11, 1996.

10 CFR 50.55a Request Number RI-05 (Continued)
Inspection of RHR Heat Exchanger Shell Welds

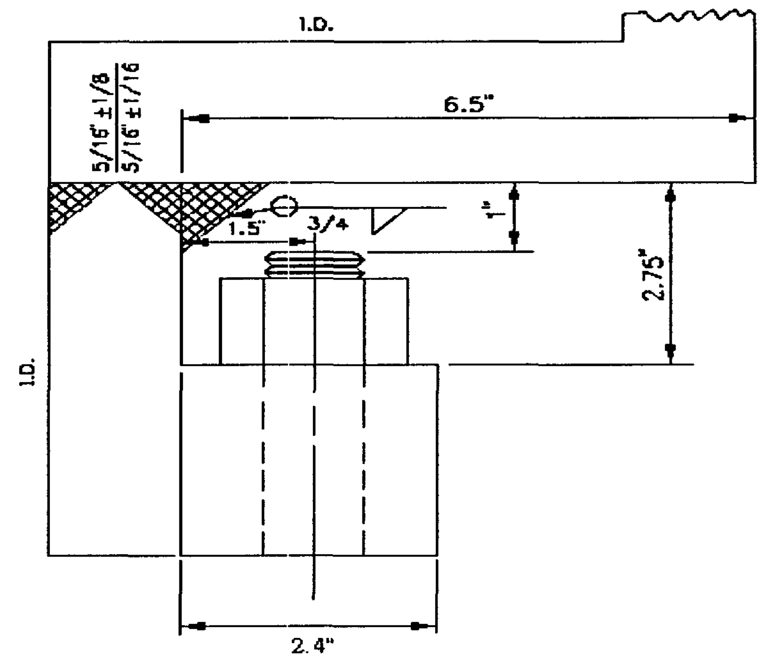

Figure RI-05.1

RHR Heat Exchanger Distributor Ring-to-Shell Weld Detail (Shell-to-Shell)

10 CFR 50.55a Request Number RI-13
Examination and Testing of Class 1, 2, and 3 Snubbers

Proposed Alternative in Accordance with 10 CFR 50.55a(a)(3)(i)
Alternative Provides Acceptable Level of Quality and Safety

ASME Code Component(s) Affected

Code Class:	1, 2, and 3
Examination Category:	F-A
Component Numbers:	Applicable Safety-Related Class 1, 2, and 3 Snubbers

Applicable Code Edition and Addenda

ASME Code Section XI, 2001 Edition, 2003 Addenda

Applicable Code Requirement

Paragraphs IWF-5200 and IWF-5300

Paragraph IWF-1220, "Snubber Inspection Requirements," states that *The inservice inspection requirements for snubbers shall be in accordance with the requirements of IWF-5000.*

Article IWF-5000, "Inservice Inspection Requirements for Snubbers":

Paragraph IWF-5200, "Preservice Examinations and Tests," states:
a) *Preservice examinations shall be performed in accordance with ASME/ANSI OM, Part 4, using the VT-3 visual examination method described in IWA-2213.*
b) *Preservice tests shall be performed in accordance with ASME/ANSI OM, Part 4.*
c) *Integral and nonintegral attachments for snubbers, including lugs, bolting, pins, and clamps, shall be examined in accordance with the requirements of this Subsection.*

Paragraph IWF-5300, "Inservice Examinations and Tests," states:
a) *Inservice examinations shall be performed in accordance with ASME/ANSI OM, Part 4, using the VT-3 visual examination method described in IWA-2213.*
b) *Inservice tests shall be performed in accordance with ASME/ANSI OM, Part 4.*
c) *Integral and nonintegral attachments for snubbers, including lugs, bolting, pins, and clamps, shall be examined in accordance with the requirements of this Subsection.*

10 CFR 50.55a Request Number RI-13 (Continued)
Examination and Testing of Class 1, 2, and 3 Snubbers

Paragraph IWA-2213, "VT-3 Examination," states:
VT-3 examinations are conducted to determine the general mechanical and structural condition of components and their supports by verifying parameters such as clearances, settings, and physical displacements; and to detect discontinuities and imperfections, such as loss of integrity at bolted or welded connections, loose or missing parts, debris, corrosion, wear, or erosion. VT-3 includes examinations for conditions that could affect operability or functional adequacy of snubbers and constant load and spring type supports.

Reason for Request

Pursuant to 10 CFR 50.55a, "Codes and Standards," Paragraph (a)(3), relief is requested from the requirements of ASME Code Section XI requirements. The proposed alternative provides assurance of snubber operability and component integrity and provides an acceptable level of quality and safety. Currently, the CNS Technical Requirements Manual (TRM) includes a comprehensive program for visual examination and functional testing of safety-related hydraulic and mechanical snubbers, including all ASME Code Class designated 1, 2, and 3 snubbers. The requirements have been reproduced in this 10 CFR 50.55a request.

A significant portion of the safety-related snubbers at CNS are also ASME Code Class 1, 2, or 3. The overlap of the visual examination and testing programs per ASME Section XI and the TRM for the Code Class snubbers presents an unnecessary redundancy.

The snubber visual examination program (currently in the TRM) and the program required by the IWF paragraphs of ASME Section XI are similar in content. Both programs include parallel criteria for operability, schedule, and sample size.

The CNS snubber testing program and the testing program required by the IWF paragraphs of ASME Section XI are also very similar in content. Both programs include parallel requirements for operability testing. Similar requirements for testing frequency, sample size, and additional sampling for failures are also included in both programs.

Regarding test frequency and sample size, the CNS Snubber Program calls for testing of 10% of the total snubber population every 18 months. Due to this sampling, 10% of the Code Class snubbers will be tested as required by OMa-1988, Part 4. Over a ten-outage cycle, 100% of the total safety-related snubber population are tested.

Regarding sample expansion for failures, the CNS Snubber Program is similar to the requirements of OMa-1988, Part 4, such that for each snubber that does not meet the functional test criteria, an additional 10% of that type of snubber shall be functionally tested.

10 CFR 50.55a Request Number RI-13 (Continued)
Examination and Testing of Class 1, 2, and 3 Snubbers

It is CNS's position that the snubber visual examination and testing program described below meets the intent of the ASME Section XI, IWF requirements. No commensurate increase in plant safety with cost benefit will be realized by imposing both inspection programs on the Code Class snubbers at CNS.

Proposed Alternative and Basis for Use

CNS will perform visual examinations of Code Class snubbers in accordance with the CNS Snubber Program described below in lieu of the IWF requirements. Personnel performing the visual examinations shall be qualified as VT-3 examiners in accordance with ASME Section XI requirements. CNS will perform functional testing of Code Class snubbers in accordance with the CNS Snubber Program in lieu of the requirements of IWF-5200 and IWF-5300. The examination of Code Class snubber integral attachments will be performed in accordance with IWB/IWC/IWD-2500.

CNS Snubber Program

The following surveillance requirements apply to all safety related snubbers

1. Visual Inspection Interval

   All snubbers shall be visually inspected in accordance with the schedule given in Table RI-13, "Snubber Visual Inspection Interval." Snubbers may be categorized in groups, "accessible" or "inaccessible" based on their accessibility for inspection during reactor operation and by type, hydraulic or mechanical. These groups may be inspected separately or jointly according to the schedule given in Table RI-13.

2. Visual Inspection Acceptance Criteria

   Visual inspections shall verify (1) that there are no visible indications of damage or impaired OPERABILITY; and (2) attachments to the foundation or supporting structure are secure. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, providing that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per paragraph 5 or 6 as applicable. However, when the fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be determined inoperable and cannot be determined OPERABLE via functional testing for the purpose of establishing the next visual inspection interval. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers.

10 CFR 50.55a Request Number RI-13 (Continued)
Examination and Testing of Class 1, 2, and 3 Snubbers

3. At least once per 18 months, a representative sample, 10% of the total of each type of snubber in use in the plant, shall be functionally tested either in place or in a bench test. For each snubber that does not meet the functional test acceptance criteria of paragraph 5 or 6, an additional 10% of that type of snubber shall be functionally tested.

4. The representative sample selected for functional testing shall include various configurations, operating environments, and the range of size and capacity of snubbers.

   a. In addition to the regular sample, snubbers that failed the previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber (if it is repaired and installed in another position) and the spare snubber shall be retested. Test results of these snubbers may not be included for the re-sampling.

   b. If any snubber selected for functional testing either fails to lockup or fails to move, i.e., frozen in place, the cause will be evaluated, and if caused by manufacturer or design deficiency all snubbers of the same design and subject to the same defect shall be tested or inspected to determine if the defect is present. This testing requirement shall be independent of the requirements stated above for snubbers not meeting the functional test acceptance criteria.

   c. For the snubber(s) found inoperable, an engineering evaluation shall be performed to determine the need for further action or testing on affected components.

5. Hydraulic Snubbers Functional Test Acceptance Criteria

   The hydraulic snubber functional test shall verify that:

   a. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

   b. Snubber bleed, or release rate, where required, is within the specified range in compression or tension.

6. Mechanical Snubbers Functional Test Acceptance Criteria

   The mechanical snubber functional test shall verify that:

   a. The force that initiates free movement of the snubber rod in either tension or compression is less than the specified maximum drag force.

10 CFR 50.55a Request Number RI-13 (Continued)
Examination and Testing of Class 1, 2, and 3 Snubbers

b. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

c. Snubber release rate, where required, is within the specified range in compression or tension.

7. Snubber Service Life Monitoring

A record of the service life of each snubber as noted in paragraph 1, the date at which the designated service life commences, and the installation and maintenance records on which the designated service life is based shall be maintained.

8. Surveillance Requirement

Concurrent with the first in-service visual inspection and at least once per 18 months thereafter, the installation and maintenance records of each snubber noted in paragraph 1 shall be reviewed to verify that the indicated service life has not been exceeded or will not be exceeded prior to the next scheduled snubber service life review. If the indicated service life will be exceeded prior to the next scheduled snubber service life review, the snubber service life shall be reevaluated or the snubber shall be replaced or reconditioned so as to extend its service life beyond the date of the next scheduled service life review. This reevaluation, replacement, or reconditioning shall be indicated in the records.

Table RI-13
Snubber Visual Inspection Interval

Number of Inoperable Snubbers			
Population or Category [1,2]	**Column A Extend Interval [3,6]**	**Column B Repeat Interval [4,6]**	**Column C Reduce Interval [5,6]**
1	0	0	1
80	0	0	2
100	0	1	4
150	0	3	8
200	2	5	13
300	5	12	25

10 CFR 50.55a Request Number RI-13 (Continued)
Examination and Testing of Class 1, 2, and 3 Snubbers

Note 1: The next visual inspection interval for a snubber population or category size shall be determined based upon the previous inspection interval and the number of inoperable snubbers found during that interval. Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible, and by type, hydraulic or mechanical. These categories may be examined separately or jointly. However, the licensee must make and document that decision before any inspection and shall use that decision as the basis upon which to determine the next inspection interval for that category.

Note 2: Interpolation between population or category sizes and the number of inoperable snubbers is permissible. Use next lower integer for the value of the limit for Columns A, B, or C if that integer includes a fractional value of inoperable snubbers as determined by interpolation.

Note 3: If the number of inoperable snubbers is equal to or less than the number in Column A, the next inspection interval may be twice the previous interval but not greater than 48 months.

Note 4: If the number of inoperable snubbers is equal to or less than the number in Column B but greater than the number in Column A, the next inspection interval shall be the same as the previous interval.

Note 5: If the number of inoperable snubbers is equal to or greater than the number in Column C, the next inspection interval shall be two-thirds of the previous interval. However, if the number of inoperable snubbers is less than the number in Column C but greater than the number in Column B, the next interval shall be reduced proportionally by interpolation, that is, the previous interval shall be reduced by a factor that is one-third of the ratio of the difference between the number of inoperable snubbers found during the previous interval and the number in Column B to the difference in the numbers in Columns B and C.

Note 6: Each surveillance requirement shall be performed within the specified time interval with a maximum allowable extension not to exceed 25 percent of the surveillance interval.

10 CFR 50.55a Request Number RI-13 (Continued)
Examination and Testing of Class 1, 2, and 3 Snubbers

Based on the above, NPPD requests authorization to use the proposed alternative in lieu of ASME Section XI, Paragraphs IWF-5200 and IWF-5300, requirements for visual examination and functional testing of Code Class snubbers.

Duration of Proposed Alternative

The proposed alternative will be used for the entire fourth ten-year interval of the Inservice Inspection Program for CNS.

Precedents

RI-13, Revision 2, was previously approved by the NRC for the third ten-year interval for CNS on March 11, 1999 (TAC No. MA2138).

References

1. NPPD Letter NLS950119 to USNRC, "Second Ten-Year Interval Inservice Inspection Relief Requests," dated June 21, 1995.

2. NPPD Letter NLS950157 to USNRC, "Third Ten-Year Interval Inservice Inspection Program," dated October 18, 1995.

3. USNRC letter to NPPD, "Cooper Nuclear Station – Inservice Inspection Relief Request RI-13," dated February 27, 1996 (TAC No. M94266).

4. USNRC letter to NPPD, "Evaluation of the Third Ten-Year Interval Inspection Program Plan and Associated Requests for Relief for Cooper Nuclear Station (TAC No. M94000)," dated October 23, 1997.

5. NPPD Letter NLS980020 to USNRC, "Inservice Inspection Relief Requests," dated April 23, 1998.

10 CFR 50.55a Request Number RI-15
Examination of Peripheral Control Rod Drive Housing Welds

Relief Request in Accordance with 10 CFR 50.55a(g)(5)(iii)
Inservice Inspection Impracticality

Component Identification

Code Class:	1
Examination Category:	B-O
Item Number:	B14.10
Component Numbers:	Applicable Control Rod Drive Housing Welds

Applicable Code Edition and Addenda

ASME Code Section XI, 2001 Edition, 2003 Addenda

Applicable Code Requirement

Table IWB-2500-1

Article IWB-2000, "Examination and Inspection," Subarticle IWB-2500, "Examination and Pressure Test Requirements," Part (a) states that *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.*

10 CFR 50.55a(b)(2)(xix), "Substitution of Alternative Methods," states that *The provisions for the substitution of alternative examination methods, a combination of methods, or newly developed techniques in the 1997 Addenda of IWA-2240 must be applied. The provisions in IWA-2240, 1998 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section are not approved for use.*

Section XI, 1997 Addenda, Paragraph IWA-2240, "Alternative Examinations," states that *Alternative examination methods, a combination of methods, or newly developed techniques may be substituted for the methods specified in this Division, provided the Inspector is satisfied that the results are demonstrated to be equivalent or superior to those of the specified method.*

Table IWB-2500-1, Examination Category B-O, requires a volumetric or surface examination to be performed on 10% of the peripheral Control Rod Drive (CRD) housing welds.

10 CFR 50.55a Request Number RI-15 (Continued)
Examination of Peripheral Control Rod Drive Housing Welds

Impracticality of Compliance

Relief is requested from the requirements of ASME Code Section XI, Table IWB-2500-1, because clearances between the support skirt and the CRDs restrict access for examination personnel, inside the support skirt, making the code-required surface examination impractical. There are thirty-six CRD housings on the periphery. Each housing has an upper and lower weld. A surface examination of 10% of these welds would require the welds in four housings to be examined. The upper CRD housing welds are located inside the reactor vessel skirt. The twelve-inch diameter hole in the reactor vessel support skirt is too small to permit access for a surface examination. The lower CRD housing welds are accessible.

Burden Caused by Compliance

To perform the code-required surface examination, the CRDs and reactor vessel support skirt would require design modification to allow access for examination.

Proposed Alternative and Basis for Use

In lieu of performing the Code-required examinations, CNS proposes to examine 100% of eight peripheral CRD lower housing welds during the inspection interval and visually examine (VT-2) the remaining CRD housing welds (upper and lower) in conjunction with the Class 1 system leakage test after each refueling outage. Using the provisions of this request as an alternative to the specific requirements of Table IWB-2500-1, identified above, will provide reasonable assurance of structural integrity of the welds. Therefore, pursuant to 10 CFR 50.55a(g)(5)(iii), NPPD requests relief from the specific Table IWB-2500-1 requirements identified in this request.

Duration of Proposed Alternative

The proposed alternative will be used for the entire fourth ten-year interval of the Inservice Inspection Program for CNS.

Precedents

RI-15, Revision 0, was previously approved by the NRC for the third ten-year interval for CNS on October 23, 1997 (TAC No. M94000).

10 CFR 50.55a Request Number RI-15 (Continued)
Examination of Peripheral Control Rod Drive Housing Welds

References

1. NPPD Letter NLS950119 to USNRC, "Second Ten-Year Interval Inservice Inspection Relief Requests," dated June 21, 1995.

2. NPPD Letter NLS950157 to USNRC, "Third Ten-Year Interval Inservice Inspection Program," dated October 18, 1995.

3. USNRC letter to NPPD, "Cooper Nuclear Station - Request for Additional Information Regarding Third Ten-Year Interval Inservice Inspection Program," dated February 8, 1996 (TAC No. M94000).

4. NPPD Letter NLS960050 to USNRC, "Response to Request for Additional Information and Submittal of Revision 1 to the Third Ten-Year Interval Inservice Inspection Program," dated April 11, 1996.

10 CFR 50.55a Request Number RI-34
Risk-Informed Inservice Inspection

Proposed Alternative in Accordance with 10 CFR 50.55a(a)(3)(i)
Alternative Provides Acceptable Level of Quality and Safety

ASME Code Component(s) Affected

Code Classes:	1 and 2
Examination Categories:	B-F, B-J, and C-F-2
Item Numbers:	B5.10, B5.20, B9.11, B9.21, B9.31, B9.32, B9.40, C5.51, and C5.81
Component Numbers:	All Class 1 and Class 2 Pressure Retaining Piping Welds

Applicable Code Edition and Addenda

ASME Code Section XI, 2001 Edition, 2003 Addenda

Applicable Code Requirement

Table IWB-2500-1 and Table IWC-2500-1

Article IWB-2000, "Examination and Inspection," Subarticle IWB-2500, "Examination and Pressure Test Requirements," states that (a) *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, require 100% and 25%, respectively, of the total number of non-exempt welds.

Section XI, Subarticle IWC-2500, "Examination and Pressure Test Requirements," states that (a) *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, Category C-F-1, does not apply to CNS. Category C-F-2 requires 7.5%, but not less than 28 welds to be selected for examination.

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.

10 CFR 50.55a Request Number RI-34 (Continued)
Risk-Informed Inservice Inspection

10 CFR 50.55a(b)(2)(xix), "Substitution of Alternative Methods," states that *The provisions for the substitution of alternative examination methods, a combination of methods, or newly developed techniques in the 1997 Addenda of IWA-2240 must be applied. The provisions in IWA-2240, 1998 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section are not approved for use.*

Section XI, 1997 Addenda, Subarticle IWA-2200, "Examination Methods," Paragraph IWA-2240, "Alternate Examinations," states that *Alternate examination methods, a combination of methods, or newly developed techniques may be substituted for the methods specified in this Division, provided the Inspector is satisfied that the results are demonstrated to be equivalent or superior to those of the specified method.*

Reason for Request

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 inservice inspection 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 as documented in the NRC-approved EPRI Topical Report TR-112657 in the development of Revision 1 of the CNS Risk-Informed Inservice Inspection (RI-ISI) Program. The use of this methodology for the selection and subsequent examination of Class 1 and Class 2 piping welds will provide an acceptable level of quality and safety.

10 CFR 50.55a Request Number RI-34 (Continued)
Risk-Informed Inservice Inspection

In accordance with the NEI Guidance for a Risk Informed Living Program, NEI 04-05, CNS reviewed the design changes, PRA changes, examination results, procedure changes, and applicable industry operating experience for the last period of the Third Inservice Inspection Interval. In summary, the Refueling Outage 22 plant improvements do not impact the consequence rankings established in the RI-ISI analysis. The design changes do not adversely affect the baseline reliability of equipment or systems included in the CNS probabilistic risk assessment model. Relevant industry events have not identified any new failure mechanisms. The success criteria of front line mitigation systems remain unchanged, and the existing model results adequately represent both the expected pre-accident and post-accident response of the plant.

Proposed Alternative and Basis for Use

As an alternative to existing Section XI requirements (Tables IWB-2500-1 and IWC-2500-1) for piping weld selection and examination volumes, NPPD will implement the alternative CNS RI-ISI Program Plan described in the enclosure ("Risked-Informed Inservice Inspection Program Plan, Cooper Nuclear Station"), which is consistent with the methodology as described in EPRI TR-112657B-A, with the exception of one deviation from this methodology: CNS assessed susceptibility of piping segments and elements at CNS to thermal stratification, cycling, and striping (TASCS) in accordance with the guidance in letters from P.J. O'Regan (EPRI), "Extension of Risk-Informed Inservice Inspection (RI-ISI) Methodology," dated February 28 and March 28, 2001 (ADAMS Accession Nos. ML010650169 and ML011070238, respectively). CNS will incorporate the applicable NRC-approved final guidance of Materials Reliability Program 24 (MRP-24) into the RI-ISI program for assessing TASCS. The proposed alternative to the piping ISI requirements with regard to the number of locations, the locations of inspections, and the methods of inspection will provide an acceptable level of quality and safety.

Duration of Proposed Alternative

This proposed alternative will be used for the entire fourth ten-year interval of the CNS Inservice Inspection Program.

Precedents

RI-34 was previously approved by the NRC for the third ten-year interval for CNS on December 9, 2004 (TAC No. MC2351).

10 CFR 50.55a Request Number RI-34 (Continued)
Risk-Informed Inservice Inspection

References

1. USNRC letter from W. Bateman to G. Vine (EPRI), "Safety Evaluation Report Related to EPRI Risk-Informed Inservice Inspection Evaluation Procedure (EPRI TR-112657, Revision B, July 1999)," dated October 28, 1999.

2. NPPD Letter NLS2004023 to USNRC, "Risk-Informed Inservice Inspection Program (Relief Request RI-34)," dated March 11, 2004, ADAMS Accession No. ML040760812.

3. USNRC letter to NPPD, "Request for Additional Information Regarding Risk-Informed Relief Request RI-34," dated May 20, 2004, TAC No. MC2351.

4. USNRC letter to NPPD, "Request for Additional Information Regarding Risk-Informed Relief Request RI-34," dated June 17, 2004, TAC No. MC2351.

5. NPPD Letter NLS2004081 to USNRC, "Response to Request for Additional Information Regarding Risk-Informed Relief Request RI-34," dated July 29, 2004, ADAMS Accession No. ML042160125.

6. NPPD Letter NLS2004091 to USNRC, "Response to Request for Additional Information Regarding Risk-Informed Relief Request RI-34," dated August 26, 2004.

10 CFR 50.55a Request Number PR-02
Definition of Pressure Retaining Boundary for System Leakage Test

Proposed Alternative in Accordance with 10 CFR 50.55a(a)(3)(ii)
Hardship or Unusual Difficulty without Compensating Increase
in Level of Quality or Safety

ASME Code Component(s) Affected

Code Class:	1
Examination Category:	B-P
Item Number:	B15.10
Component Numbers:	All Components Subject to Pressurization During a System Leakage Test

Applicable Code Edition and Addenda

ASME Code Section XI, 2001 Edition, 2003 Addenda

Applicable Code Requirement

Paragraph IWB-5222(a)

Article IWB-5000, "System Pressure Tests," Sub-subarticle IWB-5220, "System Leakage Test," Paragraph IWB-5222, "Boundaries," states that:

(a) *The pressure retaining boundary during the system leakage test shall correspond to the reactor coolant boundary, with all valves in the position required for normal reactor operation startup. The visual examination shall, however, extend to and include the second closed valve at the boundary extremity.*
(b) *The pressure retaining boundary during the system leakage test conducted at or near the end of each inspection interval shall extend to all Class 1 pressure retaining components within the system boundary.*

Table IWB-2500-1, Category B-P, Note 2, requires that the system leakage test be conducted prior to plant startup following a refueling outage.

Reason for Request

Pursuant to 10 CFR 50.55a, "Codes and Standards," Paragraph (a)(3), relief is requested from the requirements of ASME Code Section XI requirements for performing a system leakage test using the boundaries stated in Paragraph IWB-5222(a) because performing the pressure test with this boundary would result in a hardship without a compensating increase in quality and safety due to excessive radiation exposure and personnel safety concerns (temperature levels in the drywell).

10 CFR 50.55a Request Number PR-02 (Continued)
Definition of Pressure Retaining Boundary for System Leakage Test

Proposed Alternative and Basis for Use

In lieu of a system leakage test during reactor startup, as required by IWB-5222(a), a system pressure test is performed at the pressure associated with 100% rated reactor power.

a) Three of the four feedwater check valves will be closed for the system pressure test following a refueling outage. The inboard check valve on one feedwater line is kept open by reactor water cleanup (RWCU) flow. The RWCU system is kept in service during the pressure tests. The outboard check valves are the Class 1 boundary valves.

b) The four outboard main steam isolation valves (MSIV) will be closed for the system pressure test and the ten-year system pressure test [IWB-5222(b)]. The inboard MSIVs are opened to pressurize the system to the outboard valves. The outboard MSIVs are the Class 1 boundary valves.

c) Both high pressure coolant injection (HPCI) and both reactor core isolation cooling (RCIC) steam supply valves will be closed for the system pressure test following a refueling outage. These valves close automatically on low steam supply pressure. During the ten-year system pressure test [IWB-5222(b)], the system will be pressurized to the outboard valves. The outboard valves are the Class 1 boundary valves.

The position of the valves for the system leakage test as described above is consistent with the intent of IWB-5222(a). Abnormal lineups and installation of jumpers is not required for the system leakage test. The valves described above are normally open during a reactor startup. In order to pressurize the reactor coolant pressure boundary for testing, these valves must be closed. Except as described above, the Class 1 boundary is pressurized as required by the code. The VT-2 inspection includes the entire reactor coolant pressure boundary.

Since the portions of the piping between the valves described above are operated at or above reactor pressure during normal operation, any through-wall leakage would be detected by the drywell leakage collection system, or by operations personnel on normal rounds.

Performing a system pressure test at 100 percent reactor power would result in a hardship without a compensating increase in quality and safety since the proposed alternative provides reasonable assurance of operational readiness of the subject components.

10 CFR 50.55a Request Number PR-02 (Continued)
Definition of Pressure Retaining Boundary for System Leakage Test

In summary, three of the Feedwater Check valves, the outboard MSIVs, the Main Steam Line Drain valves, and the HPCI and RCIC steam supply valves will be closed during the system leakage test, but will be included in the VT-2 visual examination. A VT-2 examination will be performed during the system leakage test at a pressure not less than that associated with 100% rated reactor power and will provide reasonable assurance of the continued operational readiness of mechanical connections, extending to the Class 1 boundary. In addition, once at or near the end of the inspection interval the system leakage test shall extend to the Class 1 boundary as required by IWB-5222(b).

Based on the above, NPPD requests relief from the ASME Section XI requirements for performing a system leakage test using the boundaries stated in IWB-5222(a).

Duration of Proposed Alternative

This proposed alternative will be used for the entire fourth ten-year interval of the ISI Program for CNS.

Precedents

PR-02, Revision 2, was previously approved by the NRC for the third ten-year interval for CNS on October 23, 1997 (TAC No. M94000).

References

1. NPPD Letter NLS950157 to USNRC, "Third Ten-Year Interval Inservice Inspection Program," dated October 18, 1995.

2. USNRC letter to NPPD, "Cooper Nuclear Station - Request for Additional Information Regarding Third Ten-Year Interval Inservice Inspection Program," dated February 8, 1996 (TAC No. M94000).

3. NPPD Letter NLS960050 to USNRC, "Response to Request for Additional Information and Submittal of Revision 1 to the Third Ten-Year Interval Inservice Inspection Program," dated April 11, 1996.

4. USNRC letter to NPPD, "Cooper Nuclear Station - Request for Additional Information Regarding Third Ten-Year Interval Inservice Inspection Program (TAC No. M94000)," dated June 3, 1996.

10 CFR 50.55a Request Number PR-02 (Continued)
Definition of Pressure Retaining Boundary for System Leakage Test

5. NPPD Letter NLS960131 to USNRC, "Response to Request for Additional Information Regarding Revision 1 to the Third Ten-Year Interval Inservice Inspection Program (TAC No. M94000) and Withdrawal of Request for Relief RI-23," dated August 5, 1996.

10 CFR 50.55a Request Number PR-04
Exemption from Pressure Testing Reactor Vessel Head Flange Seal Leak Detection System

Relief Request in Accordance with 10 CFR 50.55a(g)(5)(iii)
Inservice Inspection Impracticality

ASME Code Component(s) Affected

Code Class:	1
Examination Category:	B-P
Item Numbers:	B15.10
Component Numbers:	Line No. 1-MS-152-1"

Applicable Code Edition and Addenda

ASME Code Section XI, 2001 Edition, 2003 Addenda

Applicable Code Requirement

Paragraphs IWB-5210(b) and IWB-5221(a)

Article IWB-5000, "System Pressure Tests," Sub-Article IWB-5200, "System Test Requirements":

Paragraph IWB-5210(b), "Test," states that *The system pressure tests and visual examinations shall be conducted in accordance with IWA-5000 and this Article. The contained fluid in the system shall serve as the pressurizing medium.*

Sub-subarticle IWB-5220, "System Leakage Test," Paragraph IWB-5221(a), "Pressure," states that *The system leakage test shall be conducted at a pressure not less than the pressure corresponding to 100% rated reactor power.*

Impracticality of Compliance

The Reactor Vessel Head Flange Leak Detection Line is separated from the reactor pressure boundary by one passive membrane, a silver plated O-ring located on the vessel flange. A second O-ring is located on the opposite side of the tap in the vessel flange (see Figure PR-04.1 at the end of this relief request). This line is required during plant operation in order to indicate failure of the inner flange seal O-ring. Failure of the O-ring would result in the annunciation of a High Level Alarm in the control room. Upon receipt of this alarm, control room operators would quantify the leakage rate from the O-ring and then isolate the leak detection line from the drywell sump. Failure of the inner O-ring is the only condition under which this line is pressurized.

10 CFR 50.55a Request Number PR-04 (Continued)
Exemption from Pressure Testing Reactor Vessel Head Flange Seal Leak Detection System

The configuration of this system precludes hydrostatic testing while the vessel head is removed because the odd configuration of the vessel tap coupled with the high test pressure requirement (1000 psig minimum), prevents the tap in the flange from being temporarily plugged. Adequate testing cannot be performed when the head is installed because the seal prevents complete filling of the line, which has no available vent. Operational testing of this line is precluded, because the line will only be pressurized in the event of a failure of the inner O-ring. It is impracticable to purposely fail the inner O-ring in order to perform a pressure test.

Burden Caused by Compliance

The system pressure test required by Section XI, IWB-5210(b) and IWB-5221(a), of the ASME Code for the Reactor Pressure Vessel (RPV) head flange leak detection line is impractical because of the possibility of damage to the RPV head flange O-ring seals.

Proposed Alternative and Basis for Use

Two equivalent testing methodologies are proposed. Option 2 is requested as an alternative only if scheduling or plant operations prevent Option 1 from being performed:

Option 1: A VT-2 visual examination will be performed on the line when the reactor cavity is flooded. The minimum hydrotest pressure while the reactor cavity is flooded is based on the flood depth of the cavity when the vessel head is removed. This flood depth is approximately 20 feet of water (8-10 psi). This option would require a four hour hold time prior to conducting the VT-2 inspection, and does not require insulation to be removed. Therefore, the time needed to be in the drywell is reduced which would reduce radiation dose to personnel.

Option 2: As an alternative to Option 1, a pneumatic test at 100 psig will be performed. During the performance of this test, the insulation will be removed. The line will be pressurized to 100 psig and "snooped."* A VT-2 visual inspection will also be performed. The pneumatic test meets or exceeds the ability of the approved test methodology to detect leakage. The piping insulation removal and reinstallation, and snooping the line adds significant time to the inspection.

Either of the testing alternatives will be performed in accordance with the frequency specified in Table IWB-2500-1 of Section XI of the ASME Code.

---

* Snooping is a recognized industry term for a soap bubble test. Once the line is pneumatically pressurized, a spray of liquid soap, specifically manufactured for this application, is applied to the connections along the line. Any leakage will be indicated by the formation of bubbles.

10 CFR 50.55a Request Number PR-04 (Continued)
Exemption from Pressure Testing Reactor Vessel Head Flange Seal Leak Detection System

The minimum hydrotest pressure while the reactor cavity is flooded, based on a flood depth of 20 feet of water, during conduct of the visual inspection is 8 to 10 psi. The pneumatic test is performed at 100 psig. Both the visual examination and the pneumatic test utilize sufficient pressure to provide reasonable assurance that any gross inservice flaws will be detected in the subject line. The system pressure test required by Section XI, IWB-5210(b) and IWB-5220(a), for the RPV head flange leak detection line is impractical, and the proposed alternatives provide reasonable assurance of the structural integrity of the subject line.

Based on the above, NPPD requests relief from the ASME Section XI, IWB-5210(b) and IWB-5220(a), requirements for static and operational pressure testing of the Reactor Vessel Head Flange Seal Leak Detection System.

Duration of Proposed Alternative

This proposed alternative will be used for the entire fourth ten-year interval of the ISI Program for CNS.

Precedents

PR-04, Revision 1, was previously approved by the NRC for the third ten-year interval for CNS on February 24, 2000 (TAC No. MA5090), ADAMS Accession No. ML003685789.

References

1. NPPD Letter NLS990028 to USNRC, "Inservice Inspection Relief Request PR-04, " Revision 1, dated March 19, 1999.

2. NPPD Letter NLS990112 to USNRC, "Inservice Inspection Relief Request PR-04, Revision 1 – Additional Information," dated November 12, 1999 (ADAMS Accession No. ML993270195).

10 CFR 50.55a Request Number PR-04 (Continued)
Exemption from Pressure Testing Reactor Vessel Head Flange Seal Leak Detection System

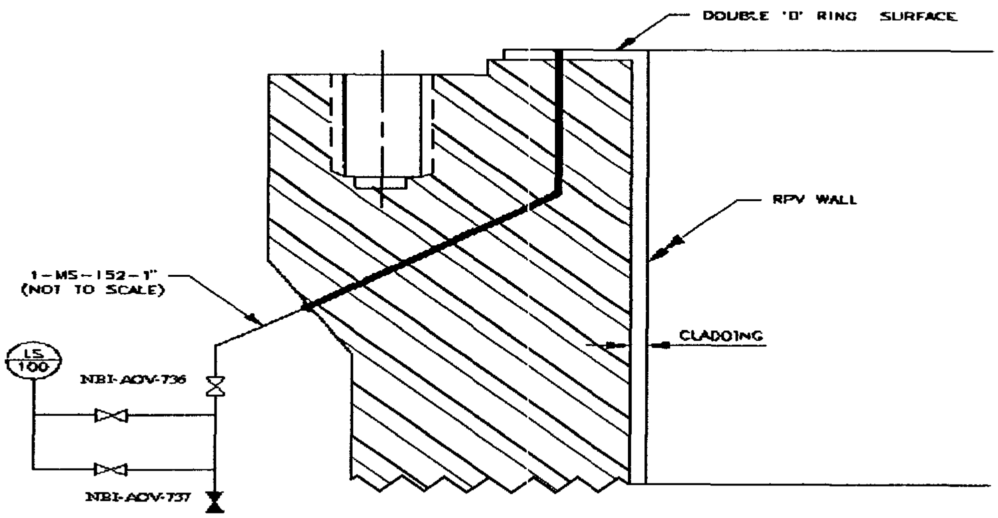

Figure PR-04.1
Head Flange Seal Leak Detection Schematic

10 CFR 50.55a Request Number PR-06
Alternate Pressure Testing for Buried Components

Proposed Alternative in Accordance with 10 CFR 50.55a(a)(3)(i)
Alternative Provides Acceptable Level of Quality and Safety

ASME Code Component(s) Affected

Code Classes:	3
Examination Categories:	D-B
Item Numbers:	D2.10
Component Numbers:	Buried Class 3 Pressure Retaining Components Subject to System Pressure Testing in the Service Water System

Applicable Code Edition and Addenda

ASME Code Section XI, 2001 Edition, 2003 Addenda

Applicable Code Requirement

IWA-5244(b)(1)

Article IWA-5000, "System Pressure Tests," Sub-subarticle 5240, "Visual Examination," Paragraph IWA-5244, "Buried Components," states:

(b) For buried components where a VT-2 visual examination cannot be performed, the examination requirement is satisfied by the following:
(1) The system pressure test for buried components that are isolable by means of valves shall consist of a test that determines the rate of pressure loss. Alternatively, the test may determine the change in flow between the ends of the buried components. The acceptable rate of pressure loss or flow shall be established by the Owner.

Reason for Request

Pursuant to 10 CFR 50.55a, "Codes and Standards," Paragraph (a)(3), relief is requested from the requirements of ASME Code Section XI, IWA-5244(b)(1), because the isolation valves are not suitable for performing a pressure isolation function. For the buried portion of the Service Water critical supply headers, isolation valves are installed in the system. The isolation valves located in the Service Water building and the control building that isolate the buried piping are large butterfly valves, which are not suitable for performing a pressure isolation function. Each critical header supplies two Residual Heat Removal Service Water booster pumps, one Reactor Equipment Cooling heat exchanger, and one diesel generator. A butterfly isolation valve is installed in the main header in the Service Water building and in each of these branch supply lines in the control building.

10 CFR 50.55a Request Number PR-06 (Continued)
Alternate Pressure Testing for Buried Components

However, since these valves are not designed to be leak tight, these five butterfly valves would provide multiple leakage paths. Leakage testing of this buried piping and determining the rate of pressure loss would require extensive valve seat maintenance and would not provide conclusive test results.

IWA-5244(b)(1) also allows determining a change in flow between the ends of the buried components. Flow instrumentation is installed in the Service Water lines in the control building. However, no flow instrumentation is installed in the system upstream of the buried piping. Accurate flow measurements using temporary flow instrumentation (e.g., ultrasonic flow meters) are not possible due to insufficient runs of straight pipe between the pump discharge and the buried piping.

The installation of permanent flow instruments would require significant system modifications. The cost of these modifications, when weighed against the benefits, is not justifiable. The following proposed alternative would provide reasonable assurance that any significant leakage from the buried piping will be detected.

Proposed Alternative and Basis for Use

In lieu of performing a system pressure test in accordance with the requirements specified in IWA-5244(b)(1), CNS shall use the provisions of IWA-5244(b)(2) to confirm that flow during operation is not impaired. IWA-5244(b)(2) states that *The system pressure test for nonisolable buried components shall consist of a test to confirm that flow during operation is not impaired.* The proposed alternative provides an acceptable level of quality and safety.

The integrity of the buried piping will be verified during quarterly pump testing. Using the downstream flow instruments, flow rate is set at the fixed test reference value and documented in the test record. The pump discharge pressure is then measured and used to determine the head produced by the pump. Head and flow rate are interdependent variables, which, together, define pump hydraulic performance. As the pump degrades, the developed head will decrease at the reference flow rate. However, due to the location of the flow rate instruments (downstream of the buried piping) a decrease in pump head during testing may also indicate side-stream leakage into the isolated non-critical header or through-wall leakage in the buried portion of the Service Water system piping. This is because the head developed by the pump decreases as flow rate increases. Significant through-wall leakage would be evident because the total flow rate would increase even though the downstream indicated flow rate is set at the reference value. Therefore, a satisfactory quarterly service water pump test also verifies the integrity of the buried system supply piping.

10 CFR 50.55a Request Number PR-06 (Continued)
Alternate Pressure Testing for Buried Components

Should the pump test results fall in the required action range of the code, then additional testing and evaluations will be performed to determine whether the unsatisfactory test results are due to side-stream leakage past butterfly isolation valves, degraded pump performance, or through-wall leakage.

Duration of Proposed Alternative

This proposed alternative will be used for the entire fourth ten-year interval of the ISI Program for CNS.

Precedents

PR-06, Revision 0, was previously approved by the NRC for the third ten-year interval for CNS on October 23, 1997 (TAC No. M94000).

References

1. NPPD Letter NLS950157 to USNRC, "Third Ten-Year Interval Inservice Inspection Program," dated October 18, 1995.

2. USNRC letter to NPPD, "Cooper Nuclear Station - Request for Additional Information Regarding Third Ten-Year Interval Inservice Inspection Program," dated February 8, 1996 (TAC No. M94000).

3. NPPD Letter NLS960050 to USNRC, "Response to Request for Additional Information and Submittal of Revision 1 to the Third Ten-Year Interval Inservice Inspection Program," dated April 11, 1996.

10 CFR 50.55a Request Number PR-11
System Pressure Test Boundary

Proposed Alternative in Accordance with 10 CFR 50.55a(a)(3)(ii)
Hardship or Unusual Difficulty without Compensating Increase
in Level of Quality or Safety

ASME Code Component(s) Affected

Code Classes:	1
Examination Categories:	B-P
Item Numbers:	B15.10
Component Numbers:	Reactor Coolant Pressure Boundary

Applicable Code Edition and Addenda

ASME Code Section XI, 2001 Edition, 2003 Addenda

Applicable Code Requirement

Paragraph IWB-5222(b)

Article IWB-5000, "System Pressure Tests," Sub-Article IWB-5200, "System Test Requirements," Sub-subarticle IWB-5220, "System Leakage Test," Paragraph IWB-5222, "Boundaries," states that *(b) The pressure retaining boundary during the system leakage test conducted at or near the end of each inspection interval shall extend to all Class 1 pressure retaining components within the system boundary.*

Reason for Request

The many vent, drain, and branch (VTDB) connections 1-inch nominal pipe size (NPS) and smaller off the reactor coolant pressure boundary have double manual isolation valves. The requirement to extend the system leakage test boundary for the leakage test conducted at or near the end of each inspection interval to the outboard valve on these VTDB connections results in a hardship without a compensating increase in the level of quality and safety. Repositioning the inboard manual valves before and after the test will take considerable time and will result in an unnecessary increase in dose to plant personnel. Manual operation (opening and closing) of the VTDB valves is estimated to expose plant personnel to 0.5 man-rem per test.

10 CFR 50.55a Request Number PR-11 (Continued)
System Pressure Test Boundary

Proposed Alternative and Basis for Use

Reactor coolant pressure boundary VTDB connections 1-inch NPS and smaller will be visually examined for leakage with the inboard isolation valve in the normally closed position during the system leakage test conducted at or near the end of each inspection interval [IWB-5222(b)]. This test provides reasonable assurance of structural integrity.

The 1-inch NPS and smaller VTDB connections are normally closed during plant operation. The outboard valves would only see pressure if the inboard valve is open or leaks by the seat. Seat leakage, although undesirable, is not indicative of a flaw in the pressure boundary. Furthermore, these valves are in close proximity to the main runs of pipe. The nonisolable portion of these VTDB connections is pressurized and VT-2 examined during the test. The VT-2 examination performed each refueling outage extends to the outboard valve, even though it is not pressurized.

In accordance with IWB-1220(a), 1989 Edition per 10 CFR 50.55a(b)(2)(xi), those portions of steam piping with an inside diameter of less than or equal to 2.64 inches, and water piping with an inside diameter of less than or equal to 1.34 inches, may be exempted from the surface and volumetric examination requirements of Table IWB-2500-1 based on the available makeup capacity. IWA-4131.1(a)(1) excludes items 1-inch NPS and smaller from the repair/replacement requirements. Thus the code recognizes that these small diameter piping connections are not significant challenges to the structural integrity of the reactor coolant pressure boundary.

The CNS technical specifications for reactor coolant pressure boundary leakage monitoring requires appropriate actions, including plant shutdown if leakage exceeded specified limits. Based on the above, NPPD requests authorization to use the proposed alternative in lieu of the ASME Section XI, IWB-5222(b), requirements.

Duration of Proposed Alternative

The proposed alternative will be used for the entire fourth ten-year interval of the Inservice Inspection Program for CNS.

Precedents

A similar request has been approved by the NRC for the Fitzpatrick Nuclear Power Plant:

USNRC letter (Safety Evaluation) from Richard J. Laufer to Michael Kansler, Entergy Nuclear Operations, Inc., "James A. Fitzpatrick Nuclear Power Plant – Request for Relief from Hydrostatic Test and Test Requirements for Small Bore Pipe (TAC No. MC7207)," dated November 1, 2005.

Enclosure to NLS2006015

Risk-Informed Inservice Inspection Program Plan

Cooper Nuclear Station

Revision 1

RISK-INFORMED INSERVICE INSPECTION PROGRAM PLAN COOPER NUCLEAR STATION,

REVISION 1

Table of Contents

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. CNS began implementing a risk-informed inservice inspection (RI-ISI) program during the third inspection period. CNS plans to continue implementing the RI-ISI Program during the fourth 10-year interval, which begins on March 1, 2006. The ASME Section XI Code of Record for the fourth ISI interval at CNS is the 2001 Edition, 2003 Addenda.

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."

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 1 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

- $\Delta T > 50^{\circ}F$,

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 inspection falls substantially below 10%, then the basis for selection needs to be investigated.

For 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	934	4	1584	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.

No other Relief Requests are modified as a result of this Relief Request.

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 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_0 and is expected to have a value less than 1E-08. Piping locations identified as medium failure potential have a likelihood of $20x_0$. These PBF likelihoods are consistent with References 9 and 14 of EPRI TR-112657. In addition, the analysis was performed both with and without taking credit for enhanced inspection effectiveness due to an increased POD from application of the RI-ISI approach.

Table 3.6-1 presents a summary of the RI-ISI Program versus ASME Section XI Code 2001 Edition 2003 Addenda 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.

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

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 provides a more robust selection process founded on actual service experience with nuclear plant piping failure data.

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

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

4. IMPLEMENTATION AND MONITORING PROGRAM

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

The applicable aspects of the ASME Code not affected by this change will be retained, such as examination 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.

PROPOSED ISI PROGRAM PLAN CHANGE

A comparison between the RI-ISI Program and ASME Section XI 2001 Edition, 2003 Addenda 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 fourth ISI interval 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.

5. REFERENCES/DOCUMENTATION

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

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

EE04-010, Rev 2; "Risk-Informed ISI (RI-ISI) Program Development for 4th Interval Update"

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	264[(1)]
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**	**1584[(1)]**

Note

1. CED6013961 added four (4) new welds to the Main Steam supply to the HPCI turbine therefore weld count has changed from Rev 0 to Rev 1.

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.
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	125[4]	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	876[4]	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.
3. CED6013961 added four (4) new welds to the Main Steam supply to the HPCI turbine therefore weld count has changed from Rev 0 to Rev 1.

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	13[5]	0	13[5]	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**

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**

Table 3.6-1 (Cont'd)

Risk Impact Analysis Results

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.

5. CED6013961 added four (4) new welds to the Main Steam supply to the HPCI turbine therefore weld count has changed from Rev 0 to Rev 1.

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						0	0	0	0						
	B-J						23	0	7	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		2	2	0	2						
	B-J						86	24	3	7		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		0	0	0	0						
	B-J						32	8	3	4		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		80[4]	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		8	5	3	8		2	0	2	0		
	B-J	4	4	0	1		433	96	28	44		191	12	25	0		
	C-F-2						36	2	1	4		898[4]	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.
4. CED6013961 added four (4) new welds to the Main Steam supply to the HPCI turbine therefore weld count has changed from Rev 0 to Rev 1.

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	0[4]	0	0[4]	0	
						B-J	17[4]	0	7[4]	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	2[4]	2[4]	0	2[4]	
						B-J	86[4]	24[4]	3	7[4]	
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	0(4)	0(4)	0	0(4)	
						B-J	32(4)	8(4)	3	4(4)	
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	80(5)	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.
4. Welds counts changed between Rev 0 and Rev 1 due to changes in Section XI from the 1989 Code in Interval 3 to 2001 Edition, 2003 Addenda for Interval 4. Dissimilar piping welds previously counted as B-F, Items B5.130 and B5.140 in the 1989 Code are now counted as B-J, Item B9.11 for example.
5. CED6013961 added four (4) new welds to the Main Steam supply to the HPCI turbine therefore weld count has changed from Rev 0 to Rev 1.

Correspondence Number: NLS2006015

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 Manager at Cooper Nuclear Station of any questions regarding this document or any associated regulatory commitments.

COMMITMENT	COMMITMENT NUMBER	COMMITTED DATE OR OUTAGE
CNS will incorporate the applicable NRC-approved final guidance of MRP-24 into the RI-ISI Program for assessing TASCS.	NLS2006015-01	Within 12 months of NRC acceptance of MRP-24