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Monticello Nuclear Generating Plant
Operated by Nuclear Management Company, LLC

SEP 1 6 2004

L-MT-04-050
10 CFR 50.55a

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Monticello Nuclear Generating Plant
Docket 50-263
License No. DPR-22

Monticello Fourth Interval Inservice Inspection Examination Plan, Revision 2

Enclosed with this letter is Revision 2 to the Monticello Nuclear Generating Plant (MNGP) Fourth Interval Inservice Inspection (ISI) Examination plan. Please post changes in your copy of the MNGP Fourth Interval ISI plan. The superseded pages should be destroyed.

This letter makes no new commitments or changes to any existing commitments.

John T. Conway for TJP

Thomas J. Palmisano
Site Vice President, Monticello Nuclear Generating Plant
Nuclear Management Company, LLC

Enclosure

cc: Administrator, Region III, USNRC
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A047

ENCLOSURE

MONTICELLO NUCLEAR GENERATING PLANT

INSERVICE INSPECTION EXAMINATION PLAN
REVISION 2

FOURTH INTERVAL

Remove	Insert
Title/Review and Approval, Rev. 1	Title/Review and Approval, Rev. 2
Record of Revision, pgs i and ii, Rev. 1	Record of Revision, pgs i, ii, iii, iv, Rev. 2
Introduction, pgs 1.2-1 through 1.2-4, Rev. 1	Introduction, pgs 1.2-1 through 1.2-4, Rev. 2
Source Documents, pgs 1.3-1 and 1.3-2, Rev. 1	Source Documents, pgs 1.3-1 and 1.3-2, Rev. 2
Requests for Relief, pgs 1.5-1 through 1.5-55, Rev. 0	Requests for Relief, pgs 1.5-1 through 1.5-71, Rev. 1

42 pages follow

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INSERVICE INSPECTION
EXAMINATION PLAN
REVISION 2

FOURTH INTERVAL
MAY 1, 2003 THROUGH MAY 31, 2012

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RECORD OF REVISIONS

RECORD OF REVISIONS (cont'd)

Summary of Changes, Plan Revision 1

Title Page	Modified Interval start date to May 1, 2003 and noted Revision 1
Page i	Noted Rev. 1 for affected sections
Page ii	Added page for Summary of Changes, Revision 1
Page 1.2-1	4th Ten Year Interval • Updated revised 3rd Interval extension dates • Added discussion regarding overlap of 3rd and 4th Intervals • Changed 4th Interval start dates Component Selection • Added requirement for examination of re-used CRD Bolting
Page 1.2-2	Code Edition Summary • Added requirement for examination of re-used CRD Bolting • Removed reference to NF (Supports). NF not applicable, supports are examined per Subsection IWF • Modified Appendix VIII Section to reflect latest modification of Appendix VIII implementation per 10CFR50.55a and remove references to specific Supplements and implementation dates
Page 1.2-3	Examination Personnel / Procedures • clarified description of examination personnel and procedure requirements. • Clarified to reflect additional use of Mandatory Appendix VIII requirements for UT personnel and procedures as modified by 10CFR50.55a dated September 26, 2002, except where relief has been granted. • Removed reference to Appendix VIII – Supplements.
Page 1.3-1	Removed reference to unpublished Reg. Guide 1.147 Rev. 13 (Draft Reg. Guide 1091)
Section 1.4	Modified entire section to remove references to Code Cases listed as approved or conditionally approved in unpublished Draft Reg. Guide 1091, but not found in published Reg. Guide 1.147, Rev. 12.

Summary of Changes, Plan Revision 2

Title Page	Modified revision number and revised titles/names
Page i	Changed Revision Number and pages for affected sections
Pages iii & iv	Added pages for Summary of Changes, Revision 2
Page 1.2-1	Background: • Format / spacing changes 4th Ten Year Interval • Format / spacing changes • Changed number of scheduled outages from Six to Five • Removed reference to maintenance outages Component Selection • Format / spacing changes
Page 1.2-2	Code Edition Summary • Added provisions for implementation of approved ISI Relief Request #7 to use 2001 Edition of Section XI for Repair/Replacement activities and associated Pressure Testing. NRC exception noted. (see Corrective Action Program OTH020219) Background for Plan / Schedule Development • changed intent of scheduling from “subject to allowing meaningful accumulation of service time for new components” to “to the extent practical
Page 1.2-3	ISI Plan Overall Description • Added RI-ISI to the description of how components are listed in the Plan and Schedule • Capitalized Item Number
Page 1.2-4	ISI Plan Overall Description (cont'd) • Added “Rev. B-A” to TR-112657
Page 1.3-1	Source Documents • Added 1995 Edition, 1995 Addenda of Section XI • Added 2001 Edition, No Addenda of Section XI

Page 1.3-2	Source Documents (cont'd) • Added NRC SER for Relief Request #7
Page 1.5-1	Relief Requests • Added Relief Request No.7
Page 1.5-35	Relief Request No.2 • Added Clarification in title regarding italicized text
Page 1.5-48	Relief Request No.3 • added Reference 11, NRC SER Title for Relief Request No.3 • updated Status as approved
Page 1.5-53	Relief Request No.5 • updated Status as approved and listed NRC SER Title for Relief Request No.5
Page 1.5-56	Relief Request No.6 • added Reference 4, NRC SER Title for Relief Request No.6 • updated Status as approved
Page 1.5-57	Relief Request No.7 • added Relief Request No. 7 including NRC exceptions

INTRODUCTION

Background:
The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code (hereafter referred to as ASME Section XI, Section XI, or the Code), Section XI Inservice Inspection (ISI) Program is prepared and maintained by the Nuclear Management Company (NMC). The Inservice Testing Program (IST) is maintained separately from this program and is submitted under separate cover. The Containment Inspection Program, as allowed by 10CFR55a(g)(6)(ii)(B), and the Repair/Replacement Program are maintained separately from this program, and, although they are not submitted, they are available at the plant site for audit and review.

4th Ten-Year Interval:
The Monticello 4th Ten-Year Inservice Inspection Interval is slightly less than 120 months due to an extension of the 3rd Interval (Letters to the NRC in May 2002 and January 2003 providing notification of 3rd Interval extension initially through March 8, 2003 (M2002057) and subsequently May 31, 2003). The 4th Interval will overlap the 3rd Interval as permitted by IWA-2430(d)(1),(2),(3), and (4) The 4th Interval begins May 1, 2003 and ends May 31, 2012. Five refueling outages are currently scheduled in this time frame.

Component Selection:
With the exception of Class 1 and 2 piping welds, components within the examination plan were selected and scheduled using criteria in the 1995 Edition of ASME Section XI with the1996 Addenda (Inspection Program B) and 10CFR50.55a(g)(6)(ii)(A), except where relief has been requested. Per 10CFR50.55a(b)(2)(xi), the requirements of IWB-1220 in the 1989 Edition of ASME Section XI, "Components Exempt from Examination," shall be used for Class 1 piping instead of the 1995 Edition of ASME Section XI with 1996 Addenda. Per 10CFR50.55a(b)(2)(xxi)(B) reused CRD Bolting must meet examination requirements for Table IWB-2500-1, Category B-G-2, Item B7.80 of ASME Section XI 1995 Edition with 1995 Addenda.

Selection of Class 1 and Class 2 piping welds in ASME Categories B-F, B-J, C-F-1 and C-F-2 are based on EPRI Topical Report 112657 Rev. B-A. "Revised Risk Informed Inservice Inspection Evaluation Procedure." The Risk Informed Class 1 and Class 2 application was also conducted in a manner consistent with ASME Code Case N-578 "Risk Informed Requirements for Class 1, 2, and 3 Piping, Method B." The use of the RI-ISI program was approved for use on July 27, 2002. (reference TAC MB3819 and Relief Request #1 for 4th ISI Interval)

INTRODUCTION (cont'd)

Code Edition Summary: The code editions implemented in the ISI Program can be summarized as follows:

Class 1 (Quality Group A)	1995 Edition with 1996 Addenda Risk-Informed Program (Relief #1) 1989 Edition IWB-1220 (10CFR50.55a)
Class 1 CRD Bolting (B7.80)	Augmented program GE SIL. No. 483R2, 10CFR50.55a(b)(2)(xxi)(B) dated September 26, 2002 specifies 1995 Edition with 1995 Addenda
Class 2 (Quality Group B)	1995 Edition with 1996 Addenda Risk-Informed Program (Relief #1)
Class 3 (Quality Group C)	1995 Edition with 1996 Addenda
MC (Metal Containment)	1992 Edition with 1992 Addenda, Subsection IWE
Appendix VIII - Mandatory	1995 Edition with 1996 Addenda as modified by 10CFR50.55a dated September 26, 2002
Repair / Replacement and associated Pressure Test	2001 Edition with No Addenda per ISI Relief Request No.7. NRC exception: must use IWA-4540(c) of the 1998 edition in lieu of the 2001 Edition requirement

Background for Plan/Schedule Development: The examination plan and schedule was developed from ASME Code requirements, Risk-Informed Methodology, individual component examination history and plant scheduling needs such as optimizing insulation removal and scaffolding needs. During the 2nd Interval, a substantial number of component replacements and alterations were made (e.g. the recirculation piping replacement). The intent of the 4th Interval scheduling was to be consistent with the 2nd and 3rd Interval, to the extent practical. For Class 1 (category B-F and B-J) and Class 2 Category C-F-1 and C-F-2) Piping Welds examined per the RI-ISI Plan, there may be little schedule correlation with previous ISI Intervals.

INTRODUCTION (cont'd)

Examination Personnel / Procedures: Inservice Inspection examination procedures and personnel certifications meet the requirements specified in the 1995 Edition of ASME Section XI with the 1996 Addenda. Additionally, UT personnel and procedures meet the requirements of Mandatory Appendix VIII as modified by 10CFR50.55a dated September 26, 2002, except where relief has been granted.

Reporting of Associated Section XI Programs: The Section XI Repair and Replacement Program, System Pressure Tests and Snubber Functional Tests are administered under separate program documents. Although these programs are administered separately, the activities required by the Repair and Replacement Program, System Pressure Tests and Snubber Functional Tests are reported in the "Inservice Inspection Summary Report" following each refueling outage.

ISI Plan Overall Description: The ASME Section XI Inservice Inspection Program is comprised of six parts: Introduction, Source Documents, Requests for Relief, ISI Boundary Drawings, ISI Isometric Drawings, and a table containing the Inservice Inspection Examination Plan and Schedule. The ISI Boundary Drawings outline Quality Group Classifications, (A, B and C). The ISI Isometric Drawings delineate ASME Section XI components or items that are included in the examination program.

The Inservice Inspection Examination Plan and Schedule lists the ASME Section XI components by Isometric Drawing Number, System, Code or RI-ISI Category, Code or RI-ISI Item, Component Description and Required Examination. The Examination Plan and Schedule identify the ASME Section XI Item Number listed in Tables IWB-2500-1, IWC-2500-1, IWD-2500-1 and Subsection IWF, and Item Number for Risk Informed Tables as identified in EPRI TR-112657, thus identifying the examination method. The examination schedule lists the anticipated period and outage for the examination of a given component. The examination schedule is intended to be flexible to allow for deviations in outage length and outage work scope. Therefore, the schedule may be changed, as allowed by the Code, without further notification. Examination distribution was developed in accordance with IWA-2432, Inspection Program B.

INTRODUCTION (cont'd)

The examination plan and schedule also contains certain non-code items to be examined, or examinations beyond Section XI Code requirements. These augmented items include licensee-initiated examinations on NC-7879-6/Tank and NC-ISI-37/W-1, W-2, W-3, W-4, W-12, W-12A shown in the plan and schedule. These items will be examined to the extent practical in accordance with the Section XI Code, 1995 Edition with 1996 Addenda, not the RI-ISI Program. Relief requests will not be submitted for these non-code exams if Section XI Code requirements cannot be met. Non-code exams are also subject to change without prior notification to the NRC.

The Monticello Plant was built prior to the implementation of Section XI Access Requirements. As a result, some components that require examination may not be completely accessible. Welds selected for examination under the Risk Informed Program were selected base on risk ranking, radiation area, and weld accessibility as allowed by EPRI TR-112657 Rev. B-A.

Source Documents:

The following referenced source documents described and listed below are basis documents used and applicable to the Monticello 4th Interval ISI Plan.

ASME BPV Code Section XI, 1992 Edition with 1992 Addenda, Subsection IWE

ASME BPV Code Section XI, 1995 Edition with 1996 Addenda

ASME BPV Code Section XI, 1995 Edition with 1995 Addenda

ASME BPV Code Section XI, 2001 Edition with No Addenda

10CFR50.55a (66FR16391)

10CFR-50.55a(g)(6)(ii)(A)(64FR51370) ASME Section XI, 1995 Edition with 1996 Addenda, Appendix VIII Supplements

10CFR-50.55a(g)(6)(ii)(A)(66FR16391) ASME Section XI, 1995 Edition with 1996 Addenda, Appendix VIII Supplement 4 Length Sizing Correction

Regulatory Guide 1.150, Rev. 1 & Generic Letter 83-15

Regulatory Guide 1.147, Rev. 12, May 1999

Monticello Inservice Inspection Licensee Control Program, 4 AWI-09.04.00

GE Nuclear Services Information Letter, SIL. No. 483R2 "CRD Cap Screw Crack Indications," September 5, 1992

Generic Letter 88-01 & NUREG 0313, Rev 2 (IGSCC (M88080A, M88082A)
**Note: All Monticello welds meet NUREG-0313, Rev. 2. Category A

NRC Letter, "Monticello Nuclear Generating Plant-Approval of Relief Request Number 8 of the Third 10 Year Inservice Inspection Program," (TAC No. M96255), November 19, 1997

NRC Letter, "MNGP-Evaluation of Relief Request No. 12 (for the 3rd 10-Year ISI Program Plan," (TAC No. MB0261), July 27, 2001

NRC Letter, "MNGP-Evaluation of Relief Request No. 13 (for the 3rd 10-Year ISI Program Plan," (TAC No. MB1833), August 22, 2001

Source Documents: (cont'd)

Monticello Notification Letter to NRC, "Notification of Extension of 3rd Ten-Year Inservice Testing and Inservice Inspection Intervals," May 30, 2002

NRC Letter, "MNGP-Third 10-Year Interval ISI Program Request for Relief from ASME Code, Section XI Requirements (TAC No. MB3904). (Relief Request #14 for 3rd ISI Interval), April 22, 2002

NRC Letter, "Monticello Nuclear Generating Plant – Risk-Informed Inservice Inspection Program (TAC MB3819)" (Relief Request #1 for 4th ISI Interval)

NRC Letter, "Monticello Nuclear Generating Plant – Fourth 10-Year Interval Inservice Inspection Program Plan Relief Request No. 7 (TAC NO. MB6897)"

EPRI Report TR-112657, Rev B-A, "Revised Risk-Informed Inservice Inspection Evaluation Procedure," December 1999

Requests for Relief

Relief Request No.	Description	Rev.
1*	Risk Informed Inservice Inspection Plan (Approved July 24, 2002 for 4th Interval)	0
2**	Reactor Vessel Circumferential Welds (Approved July 27, 2001 for remainder of current 40-Year Operating License)	1
3	Appendix VIII Supplement 4	0
4	Reactor Vessel Stabilizer Brackets	0
5	Leakage at Bolted Control Rod Drive (CRD) Housing Connections	0
6	Appendix VII Annual Training	0
7	Use of 2001 Addenda for Repair/Replacement Program	0

* Relief No. 1 is approved for use during the 4th ISI Interval and is not being submitted for further NRC Review or approval.

** Relief No. 2 is approved for use during the remaining time in the current operating license, including the 4th ISI Interval, and is not being submitted for further NRC Approval. It has been revised slightly to correct a weldname nomenclature error and update commitment statements made in Rev. 0.

Monticello Unit 1 - ISI Relief Request No. 1 (Rev. 0)

Risk Informed Inservice Inspection Plan

System: Various **Class:** 1 and 2

Category:		**Item:**
	B-F	ALL
	B-J	ALL
	C-F-1	ALL
	C-F-2	ALL

Alternative Examination Requirements:

Monticello has implemented Risk Informed Inservice Inspection program for Class 1 and Class 2 piping welds in accordance with EPRI Topical Report TR-112657 Rev. B-A, Final Report, December 1999.

Basis for Relief:

See attached Risk Informed Program Plan Submittal Rev. 0.

Status:

Approved July 24, 2002. NRC Letter, "Monticello Nuclear Generating Plant – Risk-Informed Inservice Inspection Program (TAC MB3819)"

RISK-INFORMED INSERVICE INSPECTION PROGRAM PLAN MONTICELLO NUCLEAR GENERATING PLANT - REVISION 0

Table of Contents

1. INTRODUCTION

The Monticello Nuclear Generating Plant (MNGP) is nearing the end of its 3rd 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. MNGP plans to implement a Risk-Informed Inservice Inspection (RI-ISI) Program concurrent with the start of the 4th ISI interval, which will begin on June 1, 2002. Pursuant to 10 CFR 50.55a(g)(4)(ii), the applicable ASME Section XI Code for the 4th ISI interval will be the 1995 Edition through 1996 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 risk-informed inservice inspection (RI-ISI) process used in this submittal is described in Electric Power Research Institute (EPRI) Topical Report (TR) 112657 Rev. B-A "Revised Risk-Informed Inservice Inspection Evaluation Procedure." The RI-ISI application was also conducted in a manner consistent with ASME Code Case N-578 "Risk-Informed Requirements for Class 1, 2, and 3 Piping, Method B."

1.1 Relation to NRC Regulatory Guides 1.174 and 1.178

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

1.2 PSA Quality

The Monticello Level 1 and Level 2 Probabilistic Safety Assessment (PSA) results that are based on the January 1999 update were used to evaluate the consequences of pipe ruptures for the RI-ISI assessment during power operation. The base PSA Core Damage Frequency (CDF) is 1.5E-5 events per year and the base PSA Large Early Release Frequency (LERF) is 5.5E-7 events per year for the 1999 update. The original IPE result was a CDF of 2.6E-5, which was reported to the NRC in 1992. The PSA model update history is discussed below.

The NRC review of the Monticello Individual Plant Examination (IPE) was issued in May 1994. The Staff Evaluation Report (SER) concluded the following regarding the Monticello IPE:

- The IPE is complete with respect to the information requested in Generic Letter 88-20 and associated Supplement 1;
- The IPE analytical approach is technically sound and capable of identifying plant-specific vulnerabilities;
- Monticello employed a viable means to verify that the IPE models reflect the current plant design and operation at the time of submittal to the NRC;
- The IPE had been peer-reviewed;
- Monticello participated in the IPE process;
- The IPE specifically evaluated the Monticello decay heat removal functions for vulnerabilities;
- Monticello had responded appropriately to the Containment Performance Improvement program recommendations.

There were no areas of improvement to the PSA model that were identified by the NRC in their review of the plant's IPE submittal.

The internal events PSA used for the RI-ISI evaluation is based on a more current version of the PSA than the version used for the IPE. The PSA model was updated in 1994, 1995 and 1999.

The major differences in the PSA model between the original IPE and the PSA updates through the 1995 update are that the updated model includes the following:

- Addition of a non-safety 480kv diesel generator that can backfeed through emergency bus 15 to supply battery charges;
- Installation of a hard piped vent that provides an additional means for containment heat removal;
- Improvements to safety relief valve pneumatics (including power supplies);
- Addition of a crosstie for alignment of the diesel fire pump as an additional source of low pressure makeup water;

- Replacement of an instrument air compressor with one that is not dependent on service water;
- More realistic success criteria for service water by changing from 2 of 3 pumps required for success to 1 of 3 pumps required for success;
- Internal floods initiating event frequency and effects were updated.

The 1999 PSA update was performed to incorporate the effects of power uprate conditions.

In 1997, a BWROG PSA Peer Certification Review was performed on the 1995 update PSA model. The overall conclusion was positive and said that the Monticello PSA can be effectively used to support applications involving relative risk significance. The "Facts and Observations" for Monticello have been evaluated, and are being addressed by the Monticello PSA Program. No substantial changes to the RI-ISI consequence conclusions are anticipated due to planned PSA model revisions to address these "Facts and Observations."

2. PROPOSED ALTERNATIVE TO CURRENT ISI PROGRAM REQUIREMENTS

2.1 ASME Section XI

ASME Section XI Examination Categories B-F, B-J, C-F-1 and C-F-2 currently contain the requirements for the nondestructive examination (NDE) of Class 1 and 2 piping components. The alternative RI-ISI program for piping is described in EPRI TR-112657. The RI-ISI program will be substituted for the current program for Class 1 and 2 piping (Examination Categories B-F, B-J, C-F-1 and C-F-2) in accordance with 10 CFR 50.55a(a)(3)(i) by alternatively providing an acceptable level of quality and safety. Other non-related portions of the ASME Section XI Code will be unaffected. EPRI TR-112657 provides the requirements for defining the relationship between the RI-ISI program and the remaining unaffected portions of ASME Section XI.

2.2 Augmented Programs

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

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

- The augmented inspection program for intergranular stress corrosion cracking (IGSCC) as addressed in NRC Generic Letter 88-01 and NUREG-0313, Rev. 2, have been resolved by Monticello's pipe replacement program wherein all susceptible material was replaced with resistant material. All welds are therefore classified as IGSCC Category "A". In accordance with EPRI TR-112657, piping welds identified as Category "A" are considered resistant to IGSCC, and as such are assigned a low failure potential provided no other damage mechanisms are present. Examination criteria for these welds will be in accordance with the RI-ISI process.

- The augmented inspection program for High Energy Line Break (HELB) piping includes 36 Class 1 welds that are classified as ASME Section XI, Examination Category B-J. Although MNGP is not committed to using the NUREG-0800 Standard Review Plan (SRP), Sections 3.6.1 and 3.6.2 of the SRP are used as guidance in determining appropriate design and examination requirements for specified high energy piping. The 36 Class 1 welds that require examination in accordance with the HELB augmented inspection program are between the containment penetration and the outboard isolation valve in the main steam, high pressure coolant injection, reactor core isolation cooling, reactor water clean-up, residual heat removal and core spray systems. Independent of the HELB program, the RI-ISI application selected 8 of these 36 HELB welds for examination. The remaining 28 HELB welds will continue to be examined in accordance with the HELB augmented inspection 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 MNGP. Table 3-16 of EPRI TR-112657 contains criteria for assessing the potential for thermal stratification, cycling and striping (TASCS). Key attributes for horizontal or slightly sloped piping greater than 1" nominal pipe size (NPS) include:

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

AND

$\Delta T > 50°F$,

AND

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

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

- **Turbulent penetration TASCS**

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

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

- **Low flow TASCS**

  In some situations, the transient startup of a system (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 in assessing the potential for TASCS effects. The above criteria has previously been submitted by EPRI for generic approval (Letter dated February 28, 2001, P.J. O'Regan (EPRI) to Dr. B. Sheron (USNRC), "Extension of Risk-Informed Inservice Inspection Methodology").

3.1 Scope of Program

The 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 impact on these measures due to both direct and indirect effects was considered using the guidance provided in EPRI TR-112657.

3.3 Failure Potential Assessment

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

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

3.4 Risk Characterization

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

The results of these calculations are presented in Table 3.4.

3.5 Element and NDE Selection

In general, EPRI TR-112657 requires that 25% of the locations in the high risk region and 10% of the locations in the medium risk region be selected for inspection using appropriate NDE methods tailored to the applicable degradation mechanism. In addition, per Section 3.6.4.2 of EPRI TR-112657, if the percentage of Class 1 piping locations selected for examination falls substantially below 10%, then the basis for selection needs to be investigated. For MNGP, the percentage of Class 1 welds selected per the RI-ISI process is 9.3% (76 of 817 welds), which is not a significant departure from 10%.

One additional factor that was considered during the evaluation was that the overall percentage of Class 1 selections included both socket and non-socket welds. Therefore, the percentage of Class 1 selections was 9.3% when both socket and non-socket piping welds were considered. This percentage increases to 13.2% (75 of 567 welds) when considering only those piping welds that are non-socket welded. It should be noted that non-socket welds are subject to volumetric examination, so this percentage does not rely upon welds that are solely subject to a VT-2 visual examination.

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

A brief summary is provided below, and the results of the selection 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	817	76	901	12	1718	88

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

3.5.1 Additional Examinations

The RI-ISI program in all cases will determine through an engineering evaluation the root cause of any unacceptable flaw or relevant condition found during examination. The evaluation will include the applicable service conditions and degradation mechanisms to establish that the element(s) will still perform their intended safety function during subsequent operation. Elements not meeting this requirement will be repaired or replaced.

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

3.5.2 Program Relief Requests

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

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

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

3.6 Risk Impact Assessment

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

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

3.6.1 Quantitative Analysis

Limits are imposed by the EPRI methodology to ensure that the change in risk of implementing the RI-ISI program meets the requirements of Regulatory Guides 1.174 and 1.178. The EPRI criterion requires that the cumulative change in core damage frequency (CDF) and large early release frequency (LERF) be less than 1E-07 and 1E-08 per year per system, respectively.

Monticello 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 (9E-03) and CLERP (9E-03), whereas, for medium consequence category segments, bounding estimates of CCDP (1E-04) and CLERP (1E-05) were used. The likelihood of pressure boundary failure (PBF) is determined by the presence of different degradation mechanisms and the rank is based on the relative failure probability. The basic likelihood of PBF for a piping location with no degradation mechanism present is given as x_0 and is expected to have a value less than 1E-08. Piping locations identified as medium failure potential have a likelihood of $20x_0$. In addition, the analysis was performed both with and without taking credit for enhanced inspection effectiveness due to an increased POD from application of the RI-ISI approach. The PBF likelihoods and POD values used in the analysis are consistent with those used in the approved RI-ISI pilot applications at Arkansas Nuclear One, Unit 2, and Vermont Yankee, as documented in References 9 and 14 of EPRI TR-112657.

Table 3.6-1 presents a summary of the RI-ISI program versus ASME Section XI Code requirements and identifies on a per system basis each applicable risk category. The presence of FAC was adjusted for in the performance of the quantitative analysis by excluding its impact on the risk ranking. However, in an effort to be as informative as possible, for those systems where FAC is present, 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. This is accomplished by enclosing the FAC damage mechanism, as well as all other resultant corresponding changes (failure potential rank, risk category and risk rank), in parenthesis. Again, this has only been done for information purposes, and has no impact on the assessment itself. The use of this approach to depict the impact of degradation mechanisms managed by augmented inspection programs on the risk categorization is consistent with that used in the delta risk assessment for the Arkansas Nuclear One, Unit 2 pilot application. An example is provided below.

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

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

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

Note

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

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

Risk Impact Results

System[1]	$\Delta Risk_{CDF}$		$\Delta Risk_{LERF}$	
	w/ POD	w/o POD	w/ POD	w/o POD
RPV	9.00E-11	9.00E-11	9.00E-11	9.00E-11
RWCU	4.50E-11	4.50E-11	4.50E-11	4.50E-11
MS	9.90E-10	9.90E-10	9.90E-10	9.90E-10
SLC	-4.50E-11	-4.50E-11	-4.50E-11	-4.50E-11
RCR	6.98E-09	6.98E-09	6.98E-09	6.98E-09
RCIC	-1.38E-10	-1.10E-10	-9.48E-11	-9.20E-11
RHR	-9.71E-09	-2.13E-09	-9.72E-09	-2.16E-09
CS	1.22E-09	1.22E-09	1.22E-09	1.22E-09
HPCI	-6.15E-10	2.69E-09	-5.88E-10	2.66E-09
FW	-6.20E-09	3.90E-09	-6.17E-09	3.91E-09
CCW	negligible	negligible	negligible	negligible
CRD	negligible	negligible	negligible	negligible
FPEC	no change	no change	no change	no change
PCAC	negligible	negligible	negligible	negligible
Torus	negligible	negligible	negligible	negligible
Total	**-7.40E-09**	**1.36E-08**	**-7.30E-09**	**1.36E-08**

Note

1. Systems are described in Table 3.1.

3.6.2 Defense-in-Depth

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

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

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

4. IMPLEMENTATION AND MONITORING PROGRAM

Upon approval of the RI-ISI program, procedures that comply with the guidelines described in EPRI TR-112657 will be prepared to implement and monitor the program. The new program will be integrated into the 4th Inservice Inspection Interval. No changes to the Technical Specifications or Updated Final Safety Analysis Report are necessary for program implementation.

The applicable aspects of the ASME Code not affected by this change will be retained, such as inspection methods, acceptance guidelines, pressure testing, corrective measures, documentation requirements, and quality control requirements. Existing ASME Section XI program implementing procedures will be retained and modified to address the RI-ISI process, as appropriate.

The monitoring and corrective action program will contain the following elements:

A. Identify
B. Characterize
C. (1) Evaluate, determine the cause and extent of the condition identified
   (2) Evaluate, develop a corrective action plan or plans
D. Decide
E. Implement
F. Monitor
G. Trend

The RI-ISI program is a living program requiring feedback of new relevant information to ensure the appropriate identification of high safety significant piping locations. As a minimum, risk ranking of piping segments will be reviewed and adjusted on an ASME period basis. In addition, significant changes may require more frequent adjustment as directed by NRC Bulletin or Generic Letter requirements, or by industry and plant specific feedback.

5. PROPOSED ISI PROGRAM PLAN CHANGE

A comparison between the RI-ISI program and ASME Section XI Code 1986 Edition program requirements for in-scope piping is provided in Tables 5-1 and 5-2. (Since no examination selections had been made for the 4th interval ISI Program prior to the development on the RI-ISI Program, the 3rd Interval selections were used for comparison purposes. The Code of record for the 3rd Interval was the 1986 Edition of ASME Section XI.) 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.

MNGP is implementing the RI-ISI program at the start of the 1st period of its 4th Inspection Interval. As such, 100% of the required RI-ISI program inspections will be completed in the 4th interval. Examinations shall be performed during the interval such that the period examination percentage requirements of ASME Section XI, paragraphs IWB-2412 and IWC-2412 are met.

6. REFERENCES/DOCUMENTATION

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

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

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

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

Supporting Onsite Documentation

Structural Integrity Calculation/File No. NMC-01-301, "Degradation Mechanism Evaluation for Class 1 and 2 Piping Welds at Monticello Nuclear Generating Plant," Revision 1

Structural Integrity Calculation/File No. NMC-01-302, "Risk-Informed Inservice Inspection Consequence Evaluation of Class 1 and 2 Piping for Monticello Nuclear Power Plant," Revision 1

Structural Integrity Calculation/File No. NMC-01-303, "Risk Ranking Summary, Matrix and Report for the Monticello Nuclear Generating Plant," Revision 0

Structural Integrity Calculation/File No. NMC-01-304, "Risk Impact Analysis for the Monticello Nuclear Generating Plant," Revision 1

Structural Integrity File No. NMC-01-103-4, Record of Conversation No. ROC-002, "Minutes of the Element Selection Meeting for the Risk-Informed ISI Project at the Monticello Nuclear Generating Plant," Revision 1, June 21, 2001

MNGP Calculation/File No. CA-01-216, "Monticello Nuclear Generating Plant, Risk-Informed Service History Report for Class I and II Piping Welds, ASME Categories B-F, B-J, C-F-1 and C-F-2," Revision 0

Table 3.1

System Selection and Segment / Element Definition

System Description	Number of Segments	Number of Elements
RPV – Reactor Pressure Vessel	19	112
RWCU – Reactor Water Clean-Up	10	85
MS – Main Steam	22	204
SLC – Standby Liquid Control	3	35
RCR – Reactor Coolant Recirculation	22	135
RCIC – Reactor Core Isolation Cooling	13	65
RHR – Residual Heat Removal	97	476
CS – Core Spray	36	191
HPCI – High Pressure Coolant Injection	20	158
FW – Feedwater	37	78
CCW – Component Cooling Water	2	18
CRD – Control Rod Drive	7	41
FPEC – Fuel Pool Emergency Cooling	10	54
PCAC – Primary Containment and Atmospheric Control	8	47
Torus – Torus Hard Vent	1	19
Totals	**307**	**1718**

NOTE: TABLE 3.2 was not part of the Risk-Informed ISI Program submittal and is intentionally excluded from this document.

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
RPV											
RWCU											
MS											X
SLC											
RCR									X		
RCIC		X									X
RHR		X									X
CS									X		X
HPCI		X									
FW	X	X							X		X
CCW											
CRD											
FPEC											
PCAC											
Torus											

Note

1. Systems are described in Table 3.1.

Table 3.4

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

System[1]	High Risk Region						Medium Risk Region				Low Risk Region			
	Category 1		Category 2		Category 3		Category 4		Category 5		Category 6		Category 7	
	With	Without	With	Without	With	Without	With	Without	With	Without	With	Without	With	Without
RPV							6	6			10	10	3	3
RWCU							9	9					1	1
MS	2[2]	0					5	7			14	14	1	1
SLC							1	1			2	2		
RCR			10	10			10	10					2	2
RCIC					3[3]	0	2	2	3	6	3	3	2	2
RHR			3	3	15[4]	0	13	13	5[5]	2	44	59	17	20
CS			2	2	1[6]	0	4	4	4[7]	0	6	7	19	23
HPCI			2	2			4	4	3	3	11	11		
FW	14[8]	0	14	21	2[9]	0	6	13	1	3				
CCW													2	2
CRD											2	2	5	5
FPEC											10	10		
PCAC											8	8		
Torus											1	1		
Total	16	0	31	38	21	0	60	69	16	14	111	127	52	59

Notes

1. Systems are described in Table 3.1.
2. These two segments become Category 4 after FAC is removed from consideration due to no other damage mechanisms being present.
3. These three segments become Category 5 after FAC is removed from consideration due to the presence of other "medium" failure potential damage mechanisms.
4. These fifteen segments become Category 6 after FAC is removed from consideration due to no other damage mechanisms being present.

Notes for Table 3.4 (cont'd)

5. Of these five segments, three segments become Category 7 after FAC is removed due to no other damage mechanisms being present.
6. This one segment becomes Category 6 after FAC is removed due to no other damage mechanisms being present.
7. These four segments become Category 7 after FAC is removed due to no other damage mechanisms being present.
8. Of these fourteen segments, seven segments become Category 2 after FAC is removed due to the presence of other "medium" failure potential damage mechanisms, and seven segments become Category 4 after FAC is removed due to no other damage mechanisms being present.
9. These two segments become Category 5 after FAC is removed due to no other damage mechanisms being present.

Table 3.5

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

System[1]	High Risk Region						Medium Risk Region				Low Risk Region			
	Category 1		Category 2		Category 3		Category 4		Category 5		Category 6		Category 7	
	Total	Selected	Total	Selected	Total	Selected	Total	Selected	Total	Selected	Total	Selected	Total	Selected
RPV							21	3			83	0	8	0
RWCU							84	9					1	0
MS							105	11[2]			95	0	4	0
SLC							8	1			27	0		
RCR			10	3			113	12					12	0
RCIC							12	2	28	3	12	0	13	0
RHR			31	8			67	7	10	1	269	0	99	0
CS			2	1			20	2			35	0	134	0
HPCI			8	2			27	3	33	4	90	0		
FW			36	10			38	4[3]	4	2				
CCW													18	0
CRD											10	0	31	0
FPEC											54	0		
PCAC											47	0		
Torus											19	0		
Total			87	24			495	54	75	10	741	0	320	0

Notes

1. Systems are described in Table 3.1.
2. One of these eleven welds was selected for examination by both the FAC and RI-ISI Programs. Since FAC was the only damage mechanism identified for this weld, the FAC examination will be credited toward both programs.
3. Two of these four welds were selected for examination by both the FAC and RI-ISI Programs. Since FAC was the only damage mechanism identified for these welds, the FAC examinations will be credited toward both programs.

Table 3.6-1
Risk Impact Analysis Results

System[1]	Category	Consequence Rank	Failure Potential		Inspections			CDF Impact[4]		LERF Impact[4]	
			DMs	Rank	Section XI[2]	RI-ISI[3]	Delta	w/ POD	w/o POD	w/ POD	w/o POD
RPV	4	High	None	Low	5	3	-2	9.00E-11	9.00E-11	9.00E-11	9.00E-11
RPV	6	Medium	None	Low	4	0	-4	negligible	negligible	negligible	negligible
RPV	7	Low	None	Low	2	0	-2	negligible	negligible	negligible	negligible
RPV Total								**9.00E-11**	**9.00E-11**	**9.00E-11**	**9.00E-11**
RWCU	4	High	None	Low	10	9	-1	4.50E-11	4.50E-11	4.50E-11	4.50E-11
RWCU	7	Low	None	Low	0	0	0	no change	no change	no change	no change
RWCU Total								**4.50E-11**	**4.50E-11**	**4.50E-11**	**4.50E-11**
MS	4 (1)	High	None (FAC)	Low (High)	2	0	-2	9.00E-11	9.00E-11	9.00E-11	9.00E-11
MS	4	High	None	Low	30	10	-20	9.00E-10	9.00E-10	9.00E-10	9.00E-10
MS	6	Medium	None	Low	21	0	-21	negligible	negligible	negligible	negligible
MS	7	Low	None	Low	0	0	0	no change	no change	no change	no change
MS Total								**9.90E-10**	**9.90E-10**	**9.90E-10**	**9.90E-10**
SLC	4	High	None	Low	0	1	1	-4.50E-11	-4.50E-11	-4.50E-11	-4.50E-11
SLC	6	Medium	None	Low	0	0	0	no change	no change	no change	no change
SLC Total								**-4.50E-11**	**-4.50E-11**	**-4.50E-11**	**-4.50E-11**
RCR	2	High	CC	Medium	10	3	-7	6.30E-09	6.30E-09	6.30E-09	6.30E-09
RCR	4	High	None	Low	27	12	-15	6.75E-10	6.75E-10	6.75E-10	6.75E-10
RCR	7	Low	None	Low	0	0	0	no change	no change	no change	no change
RCR Total								**6.98E-09**	**6.98E-09**	**6.98E-09**	**6.98E-09**

Table 3.6-1

Risk Impact Analysis Results

System[1]	Category	Consequence Rank	Failure Potential		Inspections			CDF Impact[4]		LERF Impact[4]	
			DMs	Rank	Section XI[2]	RI-ISI[3]	Delta	w/ POD	w/o POD	w/ POD	w/o POD
RCIC	4	High	None	Low	0	2	2	-9.00E-11	-9.00E-11	-9.00E-11	-9.00E-11
RCIC	5 (3)	Medium	TT, (FAC)	Medium (High)	1	1	0	-1.20E-11	no change	-1.20E-12	no change
RCIC	5	Medium	TT	Medium	0	2	2	-3.60E-11	-2.00E-11	-3.60E-12	-2.00E-12
RCIC	6	Medium	None	Low	1	0	-1	negligible	negligible	negligible	'negligible
RCIC	7	Low	None	Low	0	0	0	no change	no change	no change	no change
RCIC Total								**-1.38E-10**	**-1.10E-10**	**-9.48E-11**	**-9.20E-11**
RHR	2	High	TT	Medium	5	8	3	-1.03E-08	-2.70E-09	-1.03E-08	-2.70E-09
RHR	4	High	None	Low	19	7	-12	5.40E-10	5.40E-10	5.40E-10	5.40E-10
RHR	5	Medium	TT	Medium	4	1	-3	6.00E-12	3.00E-11	6.00E-13	3.00E-12
RHR	6 (3)	Medium	None (FAC)	Low (High)	5	0	-5	negligible	negligible	negligible	negligible
RHR	6	Medium	None	Low	20	0	-20	negligible	negligible	negligible	negligible
RHR	7 (5)	Low	None (FAC)	Low (High)	1	0	-1	negligible	negligible	negligible	negligible
RHR	7	Low	None	Low	8	0	-8	negligible	negligible	negligible	negligible
RHR Total								**-9.71E-09**	**-2.13E-09**	**-9.72E-09**	**-2.16E-09**
CS	2	High	CC	Medium	2	1	-1	9.00E-10	9.00E-10	9.00E-10	9.00E-10
CS	4	High	None	Low	9	2	-7	3.15E-10	3.15E-10	3.15E-10	3.15E-10
CS	6 (3)	Medium	None (FAC)	Low (High)	0	0	0	no change	no change	no change	no change
CS	6	Medium	None	Low	6	0	-6	negligible	negligible	negligible	negligible
CS	7 (5)	Low	None (FAC)	Low (High)	0	0	0	no change	no change	no change	no change
CS	7	Low	None	Low	18	0	-18	negligible	negligible	negligible	negligible
CS Total								**1.22E-09**	**1.22E-09**	**1.22E-09**	**1.22E-09**

Table 3.6-1

Risk Impact Analysis Results

System(1)	Category	Consequence Rank	Failure Potential		Inspections			CDF Impact(4)		LERF Impact(4)	
			DMs	Rank	Section XI(2)	RI-ISI(3)	Delta	w/ POD	w/o POD	w/ POD	w/o POD
HPCI	2	High	TT	Medium	5	2	-3	-5.40E-10	2.70E-09	-5.40E-10	2.70E-09
HPCI	4	High	None	Low	2	3	1	-4.50E-11	-4.50E-11	-4.50E-11	-4.50E-11
HPCI	5	Medium	TT	Medium	7	4	-3	-3.00E-11	3.00E-11	-3.00E-12	3.00E-12
HPCI	6	Medium	None	Low	7	0	-7	negligible	negligible	negligible	negligible
HPCI	6	Low	TT	Medium	1	0	-1	negligible	negligible	negligible	negligible
HPCI Total								**-6.15E-10**	**2.69E-09**	**-5.88E-10**	**2.66E-09**
FW	2 (1)	High	TASCS, TT, (FAC)	Medium (High)	0	1	1	-1.62E-09	-9.00E-10	-1.62E-09	-9.00E-10
FW	2 (1)	High	TASCS, (FAC)	Medium (High)	4	1	-3	5.40E-10	2.70E-09	5.40E-10	2.70E-09
FW	2 (1)	High	TT, (FAC)	Medium (High)	2	1	-1	-5.40E-10	9.00E-10	-5.40E-10	9.00E-10
FW	2	High	TASCS, TT	Medium	0	1	1	-1.62E-09	-9.00E-10	-1.62E-09	-9.00E-10
FW	2	High	TASCS	Medium	6	4	-2	-3.24E-09	1.80E-09	-3.24E-09	1.80E-09
FW	2	High	TT	Medium	0	0	0	no change	no change	no change	no change
FW	2	High	CC	Medium	2	2	0	no change	no change	no change	no change
FW	4 (1)	High	None (FAC)	Low (High)	6	0	-6	2.70E-10	2.70E-10	2.70E-10	2.70E-10
FW	4	High	None	Low	3	2	-1	4.50E-11	4.50E-11	4.50E-11	4.50E-11
FW	5 (3)	Medium	TASCS, TT, (FAC)	Medium (High)	0	1	1	-1.80E-11	-1.00E-11	-1.80E-12	-1.00E-12
FW	5 (3)	Medium	TASCS, (FAC)	Medium (High)	0	0	0	no change	no change	no change	no change
FW	5	Medium	TASCS	Medium	0	1	1	-1.80E-11	-1.00E-11	-1.80E-12	-1.00E-12
FW Total								**-6.20E-09**	**3.90E-09**	**-6.17E-09**	**3.91E-09**
CCW	7	Low	None	Low	1	0	-1	negligible	negligible	negligible	negligible
CCW Total								**negligible**	**negligible**	**negligible**	**negligible**
CRD	6	Medium	None	Low	10	0	-10	negligible	negligible	negligible	negligible
CRD	7	Low	None	Low	21	0	-21	negligible	negligible	negligible	negligible
CRD Total								**negligible**	**negligible**	**negligible**	**negligible**

Table 3.6-1

Risk Impact Analysis Results

System(1)	Category	Consequence Rank	Failure Potential		Inspections			CDF Impact(4)		LERF Impact(4)	
			DMs	Rank	Section XI(2)	RI-ISI(3)	Delta	w/ POD	w/o POD	w/ POD	w/o POD
FPEC	6	Medium	None	Low	0	0	0	no change	no change	no change	no change
FPEC Total								**no change**	**no change**	**no change**	**no change**
PCAC	6	Medium	None	Low	4	0	-4	negligible	negligible	negligible	negligible
PCAC Total								**negligible**	**negligible**	**negligible**	**negligible**
Torus	6	Medium	None	Low	1	0	-1	negligible	negligible	negligible	negligible
Torus Total								**negligible**	**negligible**	**negligible**	**negligible**
Grand Total								**-7.40E-09**	**1.36E-08**	**-7.30E-09**	**1.36E-08**

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 were 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. Risk Category 4 (1) inspection locations selected for examination by both the FAC and RI-ISI Programs are not included in the count since they do not represent additional examinations.
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. Hence, the word "negligible" is given in these cases in lieu of values for CDF and LERF Impact. In those cases where no inspections were being performed previously via Section XI, and none are planned for RI-ISI purposes, "no change" is listed instead of "negligible."

NOTE: TABLE 4 was not part of the Risk-Informed ISI Program submittal and is intentionally excluded from this document.

Table 5-1

Inspection Location Selection Comparison Between 1986 ASME Section XI Code and EPRI TR-112657 by Risk Region

System[1]	Code Category[2]	High Risk Region					Medium Risk Region					Low Risk Region				
		Weld Count	1986 Section XI[2]		EPRI TR-112657		Weld Count	1986 Section XI[2]		EPRI TR-112657		Weld Count	1986 Section XI[2]		EPRI TR-112657	
			Vol/Sur	Sur Only	RI-ISI	Other[3]		Vol/Sur	Sur Only	RI-ISI	Other[3]		Vol/Sur	Sur Only	RI-ISI	Other[3]
RPV	B-F						5	3	2	1		3	1	2	0	
	B-J						16	2	3	2		88	5	24	0	
RWCU	B-F						1	1	0	1						
	B-J						83	9	15	8		1	0	0	0	
MS	B-J						105	32	1	11[4]		99	21	21	0	
SLC	B-F											1	0	1	0	
	B-J						8	0	3	1		26	0	6	0	
RCR	B-F	10	10	0	3		2	2	0	0						
	B-J						111	25	5	12		12	0	3	0	
RCIC	B-J											14	0	5	0	
	C-F-2						40	1	0	5		11	1	0	0	
RHR	B-F	1	1	0	0		2	2	0	0						
	B-J	30	4	0	8		75	21	0	8		7	4	0	0	
	C-F-2											361	30	2	0	
CS	B-F	2	2	0	1											
	B-J						20	9	0	2		8	2	0	0	
	C-F-2											161	22	0	0	
HPCI	B-F	2	2	0	0											
	B-J	6	3	0	2							9	1	0	0	
	C-F-2						60	9	0	7		81	7	0	0	
FW	B-J	29	9	0	10		41	8	0	6[5]						
	C-F-2	7	5	0	0		1	1	0	0						

Table 5-1 (cont'd)

Inspection Location Selection Comparison Between 1986 ASME Section XI Code and EPRI TR-112657 by Risk Region

System[1]	Code Category[2]	High Risk Region					Medium Risk Region					Low Risk Region				
		Weld Count	1986 Section XI[2]		EPRI TR-112657		Weld Count	1986 Section XI[2]		EPRI TR-112657		Weld Count	1986 Section XI[2]		EPRI TR-112657	
			Vol/Sur	Sur Only	RI-ISI	Other[3]		Vol/Sur	Sur Only	RI-ISI	Other[3]		Vol/Sur	Sur Only	RI-ISI	Other[3]
CCW	C-F-2											18	1	0	0	
CRD	C-F-1											31	28	0	0	
	C-F-2											10	3	0	0	
FPEC	C-F-2											54	0	0	0	
PCAC	C-F-2											47	4	0	0	
Torus	C-F-2											19	1	0	0	
Total	B-F	15	15	0	4		10	8	2	2		4	1	3	0	
	B-J	65	16	0	20		459	106	27	50		264	33	59	0	
	C-F-1											31	28	0	0	
	C-F-2	7	5	0	0		101	11	0	12		762	69	2	0	

Notes

1. Systems are described in Table 3.1.
2. Since no examination selections had been made for the 4th interval ISI Program prior to the development of the RI-ISI Program, the 3rd Interval selections were used for comparison purposes. The Code of record for the 3rd Interval was the 1986 Edition of ASME Section XI. The Code Categories listed in the table are therefore in accordance with the 1986 Edition of ASME Section XI.
3. The column labeled "Other" is generally used to identify augmented inspection program locations credited per Section 3.6.5 of EPRI TR-112657. The EPRI methodology allows augmented inspection program locations to be credited if the inspection locations selected strictly for RI-ISI purposes produce substantially less than a 10% sampling of the overall Class 1 weld population. As stated in Section 3.5 of this template, MNGP achieved a 9.2% sampling without relying on augmented inspection program locations beyond those selected by the RI-ISI process. The "Other" column has been retained in this table solely for uniformity purposes with the other RI-ISI application template submittals.
4. One of these eleven welds was selected for examination by both the FAC and RI-ISI Programs. Since FAC was the only damage mechanism identified for this weld, the FAC examination will be credited toward both programs.
5. Two of these six welds were selected for examination by both the FAC and RI-ISI Programs. Since FAC was the only damage mechanism identified for these welds, the FAC examinations will be credited toward both programs.

Table 5-2
Inspection Location Selection Comparison Between 1986 ASME Section XI Code and EPRI TR-112657 by Risk Category

System[1]	Risk		Consequence Rank	Failure Potential		Code Category[2]	Weld Count	1986 Section XI[2]		EPRI TR-112657	
	Category	Rank		DMs	Rank			Vol/Sur	Sur Only	RI-ISI	Other[3]
RPV	4	Medium	High	None	Low	B-F	5	3	2	1	
						B-J	16	2	3	2	
RPV	6	Low	Medium	None	Low	B-F	3	1	2	0	
						B-J	80	3	22	0	
RPV	7	Low	Low	None	Low	B-J	8	2	2	0	
RWCU	4	Medium	High	None	Low	B-F	1	1	0	1	
						B-J	83	9	15	8	
RWCU	7	Low	Low	None	Low	B-J	1	0	0	0	
MS	4 (1)	Medium (High)	High	None (FAC)	Low (High)	B-J	6	2	0	1[4]	
MS	4	Medium	High	None	Low	B-J	99	30	1	10	
MS	6	Low	Medium	None	Low	B-J	95	21	18	0	
MS	7	Low	Low	None	Low	B-J	4	0	3	0	
SLC	4	Medium	High	None	Low	B-J	8	0	3	1	
SLC	6	Low	Medium	None	Low	B-F	1	0	1	0	
						B-J	26	0	6	0	
RCR	2	High	High	CC	Medium	B-F	10	10	0	3	
RCR	4	Medium	High	None	Low	B-F	2	2	0	0	
						B-J	111	25	5	12	
RCR	7	Low	Low	None	Low	B-J	12	0	3	0	

Table 5-2 (cont'd)

Inspection Location Selection Comparison Between 1986 ASME Section XI Code and EPRI TR-112657 by Risk Category

System[1]	Risk		Consequence Rank	Failure Potential		Code Category[2]	Weld Count	1986 Section XI[2]		EPRI TR-112657	
	Category	Rank		DMs	Rank			Vol/Sur	Sur Only	RI-ISI	Other[3]
RCIC	4	Medium	High	None	Low	C-F-2	12	0	0	2	
RCIC	5 (3)	Medium (High)	Medium	TT, (FAC)	Medium (High)	C-F-2	8	1	0	1	
RCIC	5	Medium	Medium	TT	Medium	C-F-2	20	0	0	2	
RCIC	6	Low	Medium	None	Low	B-J	5	0	2	0	
						C-F-2	7	1	0	0	
RCIC	7	Low	Low	None	Low	B-J	9	0	3	0	
						C-F-2	4	0	0	0	
RHR	2	High	High	TT	Medium	B-F	1	1	0	0	
						B-J	30	4	0	8	
RHR	4	Medium	High	None	Low	B-F	2	2	0	0	
						B-J	65	17	0	7	
RHR	5	Medium	Medium	TT	Medium	B-J	10	4	0	1	
RHR	6 (3)	Low (High)	Medium	None (FAC)	Low (High)	C-F-2	42	5	0	0	
RHR	6	Low	Medium	None	Low	C-F-2	227	20	0	0	
RHR	7 (5)	Low (Medium)	Low	None (FAC)	Low (High)	C-F-2	10	1	0	0	
RHR	7	Low	Low	None	Low	B-J	7	4	0	0	
						C-F-2	82	4	2	0	
CS	2	High	High	CC	Medium	B-F	2	2	0	1	
CS	4	Medium	High	None	Low	B-J	20	9	0	2	
CS	6 (3)	Low (High)	Medium	None (FAC)	Low (High)	C-F-2	4	0	0	0	
CS	6	Low	Medium	None	Low	B-J	8	2	0	0	
						C-F-2	23	4	0	0	
CS	7 (5)	Low (Medium)	Low	None (FAC)	Low (High)	C-F-2	13	0	0	0	
CS	7	Low	Low	None	Low	C-F-2	121	18	0	0	

Table 5-2 (cont'd)
Inspection Location Selection Comparison Between 1986 ASME Section XI Code and EPRI TR-112657 by Risk Category

System[1]	Risk		Consequence Rank	Failure Potential		Code Category[2]	Weld Count	1986 Section XI[2]		EPRI TR-112657	
	Category	Rank		DMs	Rank			Vol/Sur	Sur Only	RI-ISI	Other[3]
HPCI	2	High	High	TT	Medium	B-F	2	2	0	0	
						B-J	6	3	0	2	
HPCI	4	Medium	High	None	Low	C-F-2	27	2	0	3	
HPCI	5	Medium	Medium	TT	Medium	C-F-2	33	7	0	4	
HPCI	6	Low	Medium	None	Low	C-F-2	81	7	0	0	
HPCI	6	Low	Low	TT	Medium	B-J	9	1	0	0	
FW	2 (1)	High (High)	High	TASCS, TT, (FAC)	Medium (High)	B-J	1	0	0	1	
FW	2 (1)	High (High)	High	TASCS, (FAC)	Medium (High)	B-J	1	1	0	1	
						C-F-2	4	3	0	0	
FW	2 (1)	High (High)	High	TT, (FAC)	Medium (High)	B-J	4	1	0	1	
						C-F-2	1	1	0	0	
FW	2	High	High	TASCS, TT	Medium	B-J	2	0	0	1	
						C-F-2	1	0	0	0	
FW	2	High	High	TASCS	Medium	B-J	12	5	0	4	
						C-F-2	1	1	0	0	
FW	2	High	High	TT	Medium	B-J	1	0	0	0	
FW	2	High	High	CC	Medium	B-J	8	2	0	2	
FW	4 (1)	Medium (High)	High	None (FAC)	Low (High)	B-J	18	5	0	2[5]	
						C-F-2	1	1	0	0	
FW	4	Medium	High	None	Low	B-J	19	3	0	2	
FW	5 (3)	Medium (High)	Medium	TASCS, TT, (FAC)	Medium (High)	B-J	1	0	0	1	
FW	5 (3)	Medium (High)	Medium	TASCS, (FAC)	Medium (High)	B-J	1	0	0	0	
FW	5	Medium	Medium	TASCS	Medium	B-J	2	0	0	1	

Table 5-2 (cont'd)
Inspection Location Selection Comparison Between 1986 ASME Section XI Code and EPRI TR-112657 by Risk Category

System[1]	Risk		Consequence Rank	Failure Potential		Code Category[2]	Weld Count	1986 Section XI[2]		EPRI TR-112657	
	Category	Rank		DMs	Rank			Vol/Sur	Sur Only	RI-ISI	Other[3]
CCW	7	Low	Low	None	Low	C-F-2	18	1	0	0	
CRD	6	Low	Medium	None	Low	C-F-1	10	10	0	0	
CRD	7	Low	Low	None	Low	C-F-1	21	18	0	0	
						C-F-2	10	3	0	0	
FPEC	6	Low	Medium	None	Low	C-F-2	54	0	0	0	
PCAC	6	Low	Medium	None	Low	C-F-2	47	4	0	0	
Torus	6	Low	Medium	None	Low	C-F-2	19	1	0	0	

Notes
1. Systems are described in Table 3.1.
2. Since no examination selections had been made for the 4th interval ISI Program prior to the development of the RI-ISI Program, the 3rd Interval selections were used for comparison purposes. The Code of record for the 3rd Interval was the 1986 Edition of ASME Section XI. The Code Categories listed in the table are therefore in accordance with the 1986 Edition of ASME Section XI.
3. The column labeled "Other" is generally used to identify augmented inspection program locations credited per Section 3.6.5 of EPRI TR-112657. The EPRI methodology allows augmented inspection program locations to be credited if the inspection locations selected strictly for RI-ISI purposes produce substantially less than a 10% sampling of the overall Class 1 weld population. As stated in Section 3.5 of this template, MNGP achieved a 9.2% sampling without relying on augmented inspection program locations beyond those selected by the RI-ISI process. The "Other" column has been retained in this table solely for uniformity purposes with the other RI-ISI application template submittals.
4. This one weld was selected for examination by both the FAC and RI-ISI Programs. Since FAC was the only damage mechanism identified for this weld, the FAC examination will be credited toward both programs.
5. These two welds were selected for examination by both the FAC and RI-ISI Programs. Since FAC was the only damage mechanism identified for these welds, the FAC examinations will be credited toward both programs.

Monticello Unit 1 - ISI Relief Request No. 2 (Rev. 1)
Reactor Vessel Circumferential Shell Welds
(note - italicized text clarifies / corrects typographical errors and omissions or describes actions taken to address implementation)

System: **Reactor Vessel** **Class:** **1**

Category: **B-A** **Item:** **B1.11**

Reactor Vessel Circumferential Welds: VCBB-4, VCBB-3 and *VCBA-2 (errantly named VCBB-2 on Rev.0)*

Examination Requirements:

A September 8, 1992 revision to 10 CFR 50.55a(g)(6)(ii)(A) contains an augmented examination requirement to perform a one time volumetric examination of essentially 100% (>90%) of all circumferential and axial reactor pressure vessel (RPV) shell assembly welds. This rule revokes previously granted relief requests regarding the extent of volumetric examination on circumferential (B1.11) and longitudinal (B1.12) reactor pressure shell vessel welds. 10 CFR 50.55a(g)(6)(ii)(A) requires the augmented examinations to be performed as specified in the ASME Code Section XI (1989 Edition).

Monticello requests relief from the inspection of Reactor Vessel Circumferential (B-A) Welds Item B1.11 for the remaining term of the current license for Monticello *(during the 4th ISI Interval).*

Basis For Relief:

Monticello reactor vessel circumferential welds were not inspected to the essentially 100% volumetric requirements during the 1st and 2nd ISI inspection intervals. A relief request (RR-01) was granted on the basis of inadequate accessibility and unnecessary radiation exposure during the first two 10 year inspection intervals. Upon submittal of the 3rd Interval ISI Inspection Plan, Rev. 1 (July 29, 1993), continuance for the 1st and 2nd interval relief request (RR-01) was requested. That relief request (RR-01) was denied on the basis of 10 CFR 50.55a(g)(6)(ii)(A), effective September 8, 1992, requiring augmented examination for reactor vessel shell assembly welds.

On November 10, 1998, the NRC issued Generic Letter 98-05 "BOILING WATER REACTOR LICENSEES USE OF BWRVIP-05 REPORT TO REQUEST RELIEF FROM AUGMENTED EXAMINATION REQUIREMENTS ON REACTOR PRESSURE VESSEL CIRCUMFERENTIAL WELDS." This generic letter permits licensees to request permanent relief from the inservice inspection requirements of 10 CFR 50.55a(g)(6) for the volumetric examination of circumferential reactor pressure vessel welds if it can be demonstrated that: (1) at the expiration of the license, the circumferential welds will continue to satisfy the limiting conditional failure probability for circumferential welds in the staff's July 28, 1998, safety evaluation, and (2) operator training and procedures limit the frequency of cold over-pressure events to the amount specified in the staff's July 28, 1998, safety evaluation (Reference 1). The following is our evaluation of these two criteria.

(1) Limiting Conditional Failure Probability

The values established in Attachment 1 were calculated in accordance with the guidelines of Regulatory Guide 1.99, Revision 2. The chemistry factor for the limiting circumferential weld recorded in Attachment 1 is Monticello (manufactured by Chicago Bridge & Iron (CB&I)) plant specific (Reference 3). This value is slightly higher than the USNRC's value which utilizes Table 1 of Regulatory Guide 1.99, Revision 2. As a result, the Monticello mean RT_{NDT} value of 46.9° F is slightly higher than the USNRC's limiting plant specific analysis mean RT_{NDT} value of 44.5° F listed in Reference 5 for the CB&I reference case. A recent safety evaluation (Reference 6) identified a Brunswick Unit 1 (manufactured by CB&I) mean RT_{NDT} value of 46.5° F which also exceeded the corresponding CB&I mean RT_{NDT} value specified in Reference 5. To validate the acceptability of the failure probability in this case, the staff performed calculations using the Brunswick Unit 1 value of 46.5° F. The calculations showed only a small increase in failure probability (6 x 10^{-7}/yr for Brunswick vs. 2 x 10^{-7}/yr for the reference case). Since the Monticello mean RT_{NDT} is only slightly higher than the Brunswick Unit 1 mean RT_{NDT} (46.9° F vs. 46.5° F), it is expected that only a small increase in failure probability will result for Monticello.

The overall limiting conditional failure probability for circumferential welds across the BWR fleet listed in Reference 5 is 8.17 x 10^{-5}/yr (calculated by the staff for the Babcock & Wilcox (B&W) reference case). This limiting conditional failure probability is based on reactor vessel data that produced a calculated mean RT_{NDT} of 99.8° F (Reference 5). Since the Monticello mean RT_{NDT} (46.9° F) is less than 99.8° F, it follows that the Monticello conditional failure probability will also be less than the limiting failure probability listed in Reference 5. Attachment 2 provides a plot of mean RT_{NDT} against failure probability using results documented in References 5 and 6. Based on this trend, the conditional failure probability for Monticello is estimated to be less than 1 x 10^{-6}/yr.

In conclusion, the above discussion demonstrates that the circumferential welds of the Monticello RPV will continue to satisfy the limiting conditional failure probability listed in Reference 5.

(2) Training and Procedures

The cold pressurization events considered in Reference 1 (i.e., inadvertent injections, condensate injection, CRD injection, loss of RWCU, actual event) were reviewed to identify the critical operator actions that were assumed to occur to mitigate these events. Procedures and training were reviewed to ensure that those critical operator actions would occur with a high degree of certainty so that the low temperature over pressurization (LTOP) event frequency is maintained less than the amount specified in Reference 1 (i.e., 1×10^{-3}/yr). System design was also considered in this review to assure that the associated systems function as described in Reference 1. Results of our review indicate that in general, procedures, training and system design ensure that the evaluations contained in Reference 1 are valid for Monticello. Following are the detailed results of our review:

1. Inadvertent Injections.

   The evaluation provided in Reference 1 (paragraph 2.6.1.1) is applicable to Monticello with one exception. The evaluation considered the availability of automatic trips of high pressure injection systems on high water level. Review of Monticello procedures identified that during performance of reactor feedwater pump (RFP) testing during cold shutdown, the high reactor water level trip is bypassed. Measures are taken procedurally to close valves that prevent water from getting to the vessel. *Monticello enhanced Operations Procedure B.06.05-05 to further assure the isolation of flow to the vessel.*

2. Condensate Injection.

   The evaluation provided in Reference 1, (paragraph 2.6.1.2) is applicable to Monticello. Operating procedures provide precautions which indicate that reactor water level is to be closely monitored when starting a condensate pump. This aids in assuring that an overfill event which could lead to an LTOP event does not occur. In order to assure that operations personnel understand that an overfill event has the potential to lead to an LTOP event, *Monticello enhanced Operations Procedure B.06.05-05 to identify an LTOP event as a potential consequence of an overfill event.* Monticello also has high reactor water level and high reactor pressure alarms in the control room that warn operators when high level/pressure limitations are being exceeded which provides further assurance that an LTOP event will not occur due to condensate injection.

3. CRD Injection.

The evaluation provided in Reference 1, (paragraph 2.6.1.3), is applicable to Monticello. The evaluation notes that the risk of cold over pressurization due to CRD injection may be higher if a loss of station power were to occur during reactor vessel pressure testing. *Monticello revised vessel pressure testing procedures 0255-20-IIA-1 and 0255-20-IIC-1 to provide precautions that ensure proper response to a loss of station power (i.e., RWCU and Recirculation pumps are restored along with restoration of CRD).*

4. Loss of Reactor Water Cleanup (RWCU)

The evaluation provided in Reference 1, (paragraph 2.6.1.4), is applicable to Monticello. Monticello has procedures in place to provide guidance for recovery measures following a scram. In the event that a scram occurs that results in a RWCU isolation, procedural guidance is provided which consists of restoring the RWCU system as soon as the cause of the isolation is identified and resetting the reactor scram as soon as possible in order to limit cold water injection into the vessel. Also, procedural guidance is provided for dealing with recirculation loop or vessel stratification so that an excessive amount of cold water is not distributed throughout the reactor vessel during the restart of a tripped recirculation pump(s). *Monticello added a precaution in the Operations Procedure C.4-A for RWCU restoration in order to further inform the operations personnel of the potential of an LTOP event occurring during SCRAM recovery.*

5. Actual Event.

General Electric issued RICSIL No. 049, Inadvertent Vessel Pressurization, in response to the actual event discussed in Reference 1, (paragraph 2.6.1.5). Our assessment of the RICSIL indicated that the likelihood of a similar event occurring at Monticello is very low. Procedures require that the reactor vessel remain vented at all times during cold shutdown except as permitted by approved procedures. The reactor vessel pressure test procedure allows the vent valves to be closed during cold shutdown. During the pressure test, strict procedural guidance is provided for administratively monitoring vessel pressure and temperature while controlling CRD injection and RWCU reject in order to assure a smooth, controlled method of increasing or decreasing pressure while vessel temperature is being maintained above the required P-T limits. If reactor pressure exceeds the specified limits, during the test, the CRD pump is immediately tripped. In addition to the above mentioned procedural guidance, a requirement is included to perform an "Infrequent Test or Evolution Briefing" with all essential personnel. This briefing details the anticipated testing evolution with special emphasis on conservative decision making, plant safety awareness, lessons learned from similar in-house or industry operating experiences, the importance of open communications, and the process in which the test would be aborted if plant systems responded in an adverse manner.

The above evaluations show that system design and procedures, including the proposed enhancements, minimize the probability of LTOP events at Monticello. Our review of training indicated that licensed operator training addresses LTOP events. Initial licensed operator simulator training, for example, includes performance of surveillance tests which ensure pressure-temperature curve compliance during plant heatup and cooldown. *Additionally Monticello created Request for Training (RFT) 20012810 to provide training to operations personnel on the specific scenarios and events evaluated in Reference 1, (paragraph 2.6.1.1-5), including the features of system design and procedural controls that prevent such events at Monticello.*

<u>Conclusion</u>:

The Monticello mean RT_{NDT} value of 46.9° F is less than the mean RT_{NDT} value of 99.8° F corresponding to the B&W limiting reference case. Since the Monticello RT_{NDT} is much less than the limiting RT_{NDT}, the Monticello conditional failure probability will be well below the limiting conditional failure probability of 8.17×10^{-5}/yr calculated by the Staff for the corresponding B&W reference case.

A thorough review of existing procedures, operator training and system design identified improvement opportunities that Monticello has committed to implement. With the recommended enhancements to existing procedures and operator training and with the current design capabilities of the associated systems, the LTOP event frequency is limited to the amount specified in Reference 1, (1×10^{-3}/yr).

Based on these evaluations the conditions for requesting relief from the inservice inspection requirements of 10 CFR 50.55a(g)(6)(ii)(A), for the volumetric examination of circumferential reactor pressure vessel welds in accordance with ASME Code Section XI (1995 Edition with 1996 Addenda), Table IWB-2500-1, Examination Category B-A, Item B1.11, Circumferential Welds, are satisfied. Relief is hereby requested in accordance with 10 CFR 50.55a(a)(3)(I). The proposed alternative examinations provide an adequate level of quality and safety.

Alternate Examination:

As an alternative to the inspection requirements of ASME Code Section XI (1995 Edition with the 1996 Addenda) Category B-A, Item B1.11, 100% volume requirement, we propose that the following examination methodology be used. The alternative examination requested maintains essentially 100% (>90%) examination of reactor vessel longitudinal (axial) shell welds, Code Category B-A, Item B.1.12. Two to three percent of the circumferential RPV shell welds Code Category B-A, Item B1.11, Code Category B-A, Item B1.11 will be inspected at the intersections of the axial and circumferential welds. This is consistent with the alternate inspection requirements as specified in GL 98-05. This alternative is capable of detecting weld degradation sufficient to insure the integrity of the reactor pressure vessel boundary, and is the same as that described in the NRC SER (Reference 1).

Time Period Relief is Requested For:

Relief is presently approved by the NRC for the remaining term of the current Monticello license during the 4th 10 year interval. (Reference 7)

References:

1. NRC Safety Evaluation Report of Topical Report by the Boiling Water Reactor Vessel and Internals Project: "BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations, BWRVIP-5," (TAC No. M93925), July 28, 1998.

2. General Electric Report SASR 87-61, DRF137-0010, "Revision of Pressure-Temperature Curves to Reflect Improved Beltline Weld Toughness Estimate for the Monticello Nuclear Generating Plant – Rev. 1," December 1987.

3. NSP Letter to NRC, Submittal of Report on Reactor Pressure Vessel Specimen Test, December 21, 1998.

4. General Electric Report GENE-B13-01796-1, "Reactor Vessel Fracture Toughness Engineering Evaluation – Task 5.4," March 13, 1996

5. NRC Safety Evaluation Report of Topical Report by the Boiling Water Reactor Vessel and Internals Project: "Supplement to Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-5 Report (TAC No. MA3395)," March 7, 2000.

6. Brunswick Steam Electric Plant, Unit No's 1 and 2 – Safety Evaluation for Proposed Alternative in Accordance with 10 CFR 50.55a(a)(3)(i) for Reactor Vessel Circumferential Shell Weld Examinations (TAC No's MA9299 and MA9300).

7. NRC Letter, "Monticello Nuclear Generating Plant - Approval of Relief Request Number 12 of the Third 10 Year Inservice Inspection Program," (TAC No. MB0261), July 27, 2001.

<u>Status</u>:

Approved July 27, 2001 for continued use in 4th Interval (*...'remainder of current 40-year operating license for the unit'*), (See Reference #7 above).

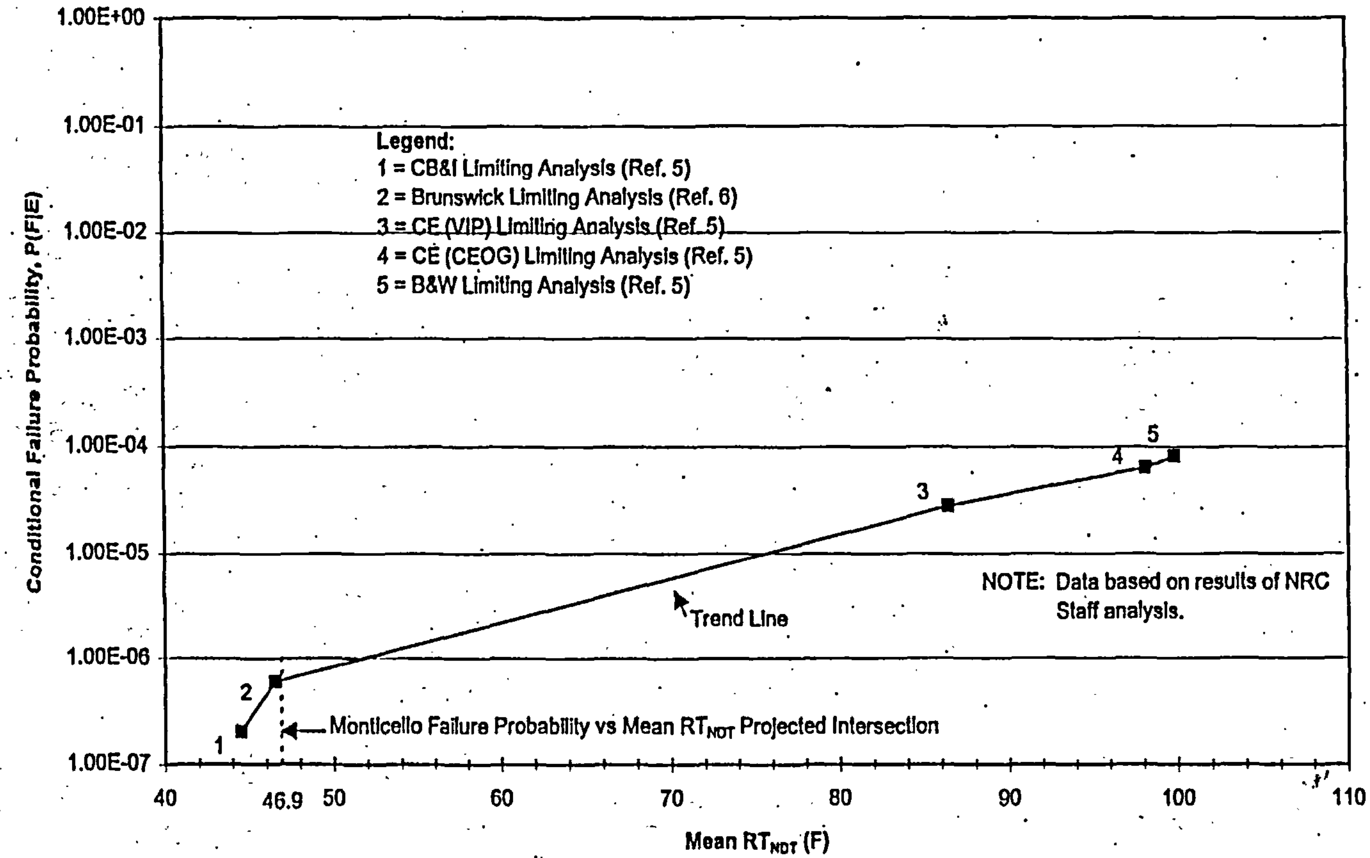
ATTACHMENT 1

Comparison of Monticello RPV Parameters to NRC Limited Plant Specific Parameters

Parameter Description	Monticello Parameters for the Bounding Circumferential Weld	USNRC Limiting Plant Specific Analyses Parameters SER Table 2.6-4 (Reference 5)	
		CB&I	B&W
Cu, wt%	0.10 (Reference 2)	0.10	0.31
Ni, wt%	0.99 (Reference 2)	0.99	0.59
CF (Chemistry factor)	138.5 (Reference 3)	134.9	196.7
EOL ID Fluence, x 10^{19} n/cm^2	0.51 (Reference 4)	0.51	0.095
ΔRT_{NDT}, °F	112.5	109.5	79.8
RT_{NDT} (u) °F	-65.6 (Reference 2)	-65	20
Mean RT_{NDT}, °F	46.9	44.5	99.8
Conditional Failure Probability P(FIE)	$<1x10^{-6}$ Attachment 2	$2x10^{-7}$	$8.17x10^{-5}$

ATTACHMENT - 2

Circ. Weld Failure Probability vs Mean RT_{NDT} Trend Using Limiting CE, CB&I, B&W and Brunswick Data



Monticello Unit 1 - Relief Request No. 3 (Rev. 0)

Appendix VIII Supplement 4

System/Component(s) For Which Relief Request Will Be Used

Code Class:	Class 1
Reference:	ASME, Section XI, Tables IWB-2500-1 (1995 Edition, 1996 Addenda)
Examination Category:	B-A
Item Number:	B1.10, B1.20
Description:	Alternative Requirement to Appendix VIII, Supplement 4 "Qualification Requirements for the Clad/Base Metal Interface of Reactor Vessel"
Component Numbers:	All

Code and 10 CFR 50.55a Requirements:

10 CFR 50.55a(b)(2) was amended on September 22, 1999 to reference Section XI of the ASME Code through the 1995 Edition with the 1996 Addenda (64 FR 51370). This amendment provides an implementation schedule for the supplements to Appendix VIII of Section XI to the 1995 Edition with the 1996 Addenda.

Supplement 4 to Appendix VIII, Subparagraph 3.2(c) imposes three statistical parameters for depth sizing. The first parameter, 3.2(c)(1), pertains to the slope of a linear regression line. The linear regression line is the difference between measured versus true value plotted along a through-wall thickness. The second parameter, 3.2(c)(2), pertains to the mean deviation of flaw depth. The third parameter, 3.2(c)(3), pertains to a correlation coefficient.

The Final Rule was amended by Federal Register Notice (66FR16391) dated March 26, 2001. This amendment specified the use of a flaw length sizing tolerance criterion of 0.75 inch Root Mean Square (RMS) for reactor vessel qualification to be used in conjunction with the 0.15 inch RMS for depth sizing specified in the Rule in lieu of paragraphs 3.2(a) and 3.2(b). In the Notice, there was no reference to the elimination of the statistical parameters of Paragraph 3.2(c), which were intended for use with paragraphs 3.2(a) and 3.2(b) of Appendix VIII, Supplement 4. There was no amendment statement included to reflect the use of the RMS error calculations for depth and length sizing in lieu of Paragraph 3.2(c).

Basis for Alternative Examination:

This relief request was developed using the Electric Power Research Institute (EPRI) Performance Demonstration Initiative (PDI) ASME Section XI, Appendix VIII Implementation Guideline. It is modeled after the sample request for relief associated with the Supplement 4 published discrepancies: Appendix D, "Sample Request for Relief – Alternative Length Sizing Criteria (Revised)." (Reference 5)

The U.S. nuclear utilities created PDI to implement demonstration requirements contained in Appendix VIII. PDI developed a performance demonstration program for qualifying UT techniques. PDI does not use paragraph 3.2(c) for sizing qualifications. The solution for resolving the differences between the PDI program and the Code was for PDI to participate in the development of a Code case that reflected PDI's program. The Code case was presented to ASME for discussion and consensus building. NRC representatives participated in this process. ASME approved the Code Case and published it as Code Case N-622, "Ultrasonic Examination of RPV and Piping, Bolts and Studs, Section XI, Division 1." (Reference 6) The NRC first approved the use of Code Case N-622 for Florida Power and Light Company's St. Lucie Plant Unit 2 (TAC No. MA5041). (Reference 7)

Operating in parallel with the actions of PDI, the Staff incorporated most of Code Case N-622 criteria in the Rule published in the Federal Register, 64 FR 51370 dated September 22, 1999. This amendment requires the implementation of the ASME Code Section XI, Appendix VIII, Supplement 4, 1995 Edition with the 1996 Addenda. The required implementation date for Supplement 4 was November 22, 2000. Appendix IV to Code Case N-622 contains the proposed alternative sizing criteria which has been authorized by the Staff. However, the sizing parameters printed in the published Rule differed from the sizing parameters implemented by the PDI Program and Code Case N-622.

On January 12, 2000, NRC Staff, representatives from the EPRI Nondestructive Examination Center, and representatives from PDI participated in a conference call. The discussion during the conference call included the differences between Supplement 4, "Qualification Requirements for the Clad/Base Metal Interface of Reactor Vessel," to Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems," Paragraph 10 CFR 50.55a(b)(2)(xv)(C)(1) in the rule (Federal Register, 64 FR 51370), and the implementation of Supplement 4 by the PDI Program. (Reference 8)

In a public meeting on October 11, 2000 at NRC offices in White Flint, MD, the PDI identified the discrepancy between the PDI Program and statistical parameters required by Subparagraph 3.2(c). The Staff agreed that the inclusion of the statistical parameters of Paragraph 3.2(c) of Supplement 4 to Appendix VIII was an oversight. The NRC agreed that Paragraph 10 CFR 50.55a(b)(2)(xv)(C)(1) should have excluded Subparagraph 3.2(c) as a requirement. (Reference 9)

In Subparagraph 3.2(c), the linear regression line is the difference between measured versus true value plotted along a through-wall thickness. For Supplement 4 performance demonstrations, a linear regression line of the data is not applicable because the performance demonstrations are performed on test specimens with flaws located in the inner 15% through-wall. The difference between measured versus true value produce a tight grouping of results that resemble a shotgun pattern. The slope of a regression line from such data is extremely sensitive to small variations, thus making the parameter of Subparagraph 3.2(c)(1) a poor and inappropriate acceptance criterion.

The value used in the 3.2(c)(2) is too lax with respect to evaluating flaw depths within the inner 15% of wall thickness. Therefore, Monticello proposes to use the more appropriate criterion of 0.15 inch RMS of 10 CFR 50.55a(b)(2)(xv)(C)(1), that modifies Subparagraph 3.2(a) as the acceptance criteria.

Subparagraph 3.2(c)(3) pertains to a correlation coefficient. This value of correlation coefficient is inappropriate for this application since it is based on the linear regression from Subparagraph 3.2(c)(1).

The NRC Staff previously approved MNGP use of this Alternative to the Code and 10 CFR 50.55a on August 22, 2001 (TAC No. MB1833) for use during the 3rd ISI Interval. (Reference 10)

Alternative Examination:

Pursuant to 10 CFR 50.55a(a)(3)(i), relief is requested to use the RMS Error calculations in lieu of the statistical parameters of Subparagraph 3.2(c) in Supplement 4 of the 1995 Edition 1996 Addenda of ASME Section XI Appendix VIII. As discussed above and demonstrated by the PDI, this will provide an acceptable level of quality and safety.

Implementation Schedule:

This alternative is requested for continued use for the 4th Ten-Year Interval of the Inservice Inspection Program for Monticello.

References:

1. ASME Boiler and Pressure Vessel Code, Section XI, 1995 Edition with 1996 Addenda

2. Federal Register, Rules and Regulations, September 22, 1999 (64 FR 51370)

3. Federal Register Notice, Industry Codes and Standards, Amended Requirements, March 26, 2001 (66 FR 16391)

4. Federal Register, Rules and Regulations, September 26, 2002 (67 FR 60520)

5. Performance Demonstration Initiative (PDI), "Guideline for Implementation of Appendix VIII and 10CFR50.55a," Volume One, Programmatic Implementation, Rev. 2, Appendix D, October 14, 2000

6. ASME Section XI Nuclear Code Case N-622, "Ultrasonic Examination of RPV and Piping, Bolts, and Studs"

7. NRC Staff letter to Mr. T. F. Plunkett, Florida Power and Light Company, September 23,1999.

8. Meeting Summary, Teleconference between NRC and representatives from PDI, D.G. Naujock, Metallurgist, NDE & Metallurgy Section, to Edmund J. Sullivan, Chief NDE & Metallurgy Section, Chemical Engineering Branch, Division of Engineering, U.S. NRC, March 6, 2000.

9. NRC Memo, "Summary of Public Meeting Held on October 11, 2000, with PDI Representatives," November 13, 2000

10. NRC Letter to Nuclear Management Company, "MNGP – Evaluation of Relief Request No. 13 for the Third 10-Year Interval Inservice Inspection Program," (TAC No. MB1833), August 22, 2001

11. NRC Letter to Nuclear Management Company, "Relief Request Nos. 3 and 6 for the Fourth 10-Year Interval of the Inservice Inspection Examination Plan" (TAC No. MB6896), March 28, 2003

Status:

Approved on March 28, 2003 for use during the 4th Interval. (See Reference 11 above)

Monticello Unit 1 - ISI Relief Request No. 4 (Rev. 0)

Reactor Vessel Stabilizer Brackets

System: Reactor Vessel **Class:** 1

Category: B-K **Item:** B10.10

Code Examination Requirements (ASME Section XI, 1995 Edition with 1996 Addenda):

Perform surface examination on 100% of the vessel stabilizer bracket to vessel integral attachment welds.

Basis for Relief:

The vessel stabilizer brackets are surrounded by mirror insulation secured with cable hangers and buckles, ventilation ductwork and electrical installations.

The stabilizer brackets do not provide support during normal operation. The brackets stabilize the vessel against local and seismic loads.

Alternative Examination:

Pursuant to 10 CFR 50.55a(a)(3)(i), Monticello proposes to perform a surface examination on the stabilizer brackets if local (jet reaction forces) or seismic loads are experienced. This proposed alternative to the requirements of Table IWB-2500-1, Category B-K, Item B10.10 will provide an acceptable level of quality and safety.

Status:

This Alternative to the Code was previously approved for 2nd and 3rd Intervals:

- NRC Letter, "Monticello - Second Ten-Year Inservice Inspection (ISI) Program," (TAC No. 46510), November 29, 1990, Relief Request No. 51
- NRC Letter, "Evaluation of the Third 10-Year Interval Inservice Inspection Program Plan and Associated Requests for Relief for Monticello," (TAC No. M82545), October 18, 1994, Relief Request No. 2

Not yet approved for 4th interval.

Requested for continued use during 4th interval.

Monticello Unit 1 - ISI Relief Request No. 5 (Rev. 0)

Leakage at Bolted Control Rod Drive (CRD) Housing Connections

SYSTEM: Bolted CRD Housing Joint

Class: 1

Category: B-P

Item: B15.10

Code Examination Requirements:

IWA-5250(a)(2): If leakage occurs at a bolted connection on other than a gaseous system, one of the bolts shall be removed, VT-3 examined, and evaluated in accordance with IWA-3100.

Basis for Relief:

10 CFR Part 50, Section 50.55a(a)(3), which states, (in part):

> *"Proposed alternatives to the requirements of paragraphs (c), (d), (e), (f), (g), and (h) of this section or portions thereof may be used when...*
>
> *(ii) Compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety."*

The CRD (Control Rod Drive) housings are flanged connections beneath the reactor vessel that are used to secure the 121 CRD mechanisms in position below the vessel. Each of the 121 CRD to CRD housing bolted joints utilizes eight bolts, washers, and nuts to hold the CRD mechanism in position. The joint also utilizes three hollow metal O-rings to provide a watertight seal capable of withstanding full reactor pressure at normal operating temperatures.

The CRD housing joints are VT-2 examined as part of the periodic Reactor Pressure Vessel Leakage and Hydrostatic pressure tests. These tests are conducted with the vessel temperature much less than the design operating temperature. For a typical test, the vessel temperature would be <212° F, as compared to a normal operating temperature of about 540° F. It is not unusual for these bolted joints to leak slightly during periodic reactor vessel pressure tests conducted at test temperatures below normal operating temperature.

This is a condition identified in the original design of the connection by the Architect/Engineer, General Electric (GE). GE developed guidance to permit evaluation of a leaking CRD housing bolted connection over a period of time, while at test pressure, to determine whether the leak will stop once the vessel heats up to normal operating pressure. This leakage evaluation criteria is incorporated into the VT-2 tests for these joints.

Compliance with Code Requirement IWA-5250(a)(2) represents a hardship (burden) in the case of the CRD housing bolted joints because:

1) Examining the bolting would involve the accumulation of considerable personnel radiation exposure, since the work must be performed in a relatively high dose rate area inside the drywell, immediately below the reactor vessel. Typical shutdown dose rates in the vicinity of the bolting flanges would be on the order of 50 to 100 mr/hr.

2) Since the reactor pressure vessel test is critical path item, the additional time needed to depressurize the vessel, remove the bolting, perform the exam, and then re-pressurize the vessel to retest the joint would delay plant startup from an outage by an equivalent amount of time. The cost of such delays is significant, since it is estimated that the cost of extending the duration of an outage is $379,000 per day (including replacement power costs)(this is estimated cost submitted in 1993 (see TAC No. M82545 referenced in "Status" section)

Compliance with Code requirement IWA-5250(a)(2) would not result in a compensating increase in quality or safety because:

1) CRD Housing joint leakage during (relatively) low temperature testing is not unexpected due to the design of the bolted joint. This joint is unusual in that it has hollow metal o-rings that require the CRD housing bolts to be tightened within a specific torque range in order to function properly at normal operating temperature. Thus, the bolts cannot simply be tightened to stop leakage as might be done for a conventional gasketed joint. As noted previously, GE developed guidance to evaluate any CRD housing leakage to determine if the leakage will persist at normal operating temperature/pressure and should therefore be corrected.

2) Leakage that is found to be acceptable per the guidance is not considered adverse to quality or safety and need not be corrected prior to startup. This type of analysis is consistent with Section XI.

3) Code paragraph IWB-3142 allows analysis of the leakage for acceptability. Performance of the VT-3 bolting examination does not represent a corrective action for the joint leakage and will not reduce the likelihood of joint leakage upon retest. Therefore, the VT-3 bolting examination does not contribute to increased quality or safety.

4) The bolts in the CRD housing connection are periodically examined when the joint is disassembled, per Table IWB-2500-1, Item B7.80 (1995 Edition with no Addenda per 10CFR50.55A Paragraph *(b)(2)(xxi)(B))* and Procedure 9309, "Changeout Selected CRD's – Maintenance" and Commitment No. M92076A. Four of the eight bolts on each housing joint were replaced with new bolts in 1991 under Work Control Record (WCR) 91-01909. It was also reported in General Electric SIL 483 that only three uniformly distributed housing bolts are required to support the CRD mechanism. These factors provide a high degree of confidence in the long term safety and integrity of the CRD housing joints.

Earlier Section XI code editions invoked by Monticello's 1st and 2nd Ten-Year Inspection Interval Programs did not include the subject examination requirement. During the 3rd Inspection Interval, Relief Request 7 was granted by the NRC in an SER dated October 18, 1994.

Alternate Examination:

Pursuant to 10 CFR 50.55a(a)(3)(ii), the following alternative is proposed. Any leakage found at a CRD housing bolted joint during a periodic pressure test performed at a temperature much less than operating temperature will be evaluated to determine whether it will stop leaking at operating temperature. If this evaluation shows the leak will stop as temperature increases to normal operating temperature, no further action will be taken. The acceptance criteria is based on guidance provided by General Electric and is included in the VT-2 tests for the joint (Note: This criteria was submitted for NRC review during the Request for Relief process previously approved on October 18, 1994, therefore it is not included in this submittal). If the leakage is determined to be unacceptable according to the General Electric guidelines and the joint is disassembled to correct the leak, any CRD bolting that is reused will be examined by the VT-1 examination method (10 CFR 50.55a(b)(2)(xxi)(B) dated September 26, 2002).

Upon approval of this Relief Request, MNGP commits to revise the applicable pressure test procedure to perform a VT-1 exam in lieu of a VT-3 exam specified by IWA-5250(a)(2) on all CRD bolting that will be reused when the GE acceptance criteria has been exceeded and disassembly is required to correct the leak.

Status:

Approved on June 9, 2003 for use during the 4th Interval, NRC Letter to Nuclear Management Company, "Fourth 10-Year Interval Inservice Inspection Program Plan Relief Request No. 5" (TAC No. MB6956)

Monticello Unit 1 - ISI Relief Request No. 6 (Rev. 0)

Appendix VII Annual Training

System/Component(s) For Which Relief Will Be Used:

Code Class:	All
Reference:	ASME, Section XI 1995 Edition 1996 Addenda, Appendix VII, VII-4240
Examination Category:	All
Item Number:	All
Description:	All NDE Examiners performing ultrasonic volumetric examination in accordance with ASME Section XI, 1995 Edition 1996 Addenda and Appendix VII, Annual Training.
Component Numbers:	All

Code and 10 CFR 50.55a Requirement:

ASME Section XI, 1995 Edition, 1996 Addenda, Mandatory Appendix VII, Paragraph VII-4240: Supplemental training is required on an annual basis to impart knowledge of new developments, material failure modes, and any pertinent technical topics as determined by the Employer. The extent of this training shall be a minimum of 10 hours per year. A record of attendance and the topics covered during the training shall be maintained; however no examination is required.

10 CFR 50.55a, paragraph (b)(2)(xiv): All personnel qualified for performing ultrasonic examinations in accordance with Appendix VIII shall receive 8 hours of annual hands-on training on specimens that contain cracks. This training must be completed no earlier than 6 months prior to performing ultrasonic examinations at a licensee's facility.

Basis For Relief Request:

10 CFR 50.55a was amended in the Federal Register (Volume 64, No. 183 dated September 22, 1999) to require Appendix VIII – Supplements for accelerated implementation in accordance with ASME Section XI 1995 Edition, 1996 Addenda.

Basis For Relief Request (continued):

Paragraph 2.4.1.1.1 in the Federal Register (Volume 64, No. 183 dated September 22, 1999) during rule making contained the following statement: "The NRC had determined that this requirement (10 hours of training on an annual basis) was inadequate for two reasons. The first reason was that the training does not require laboratory work and examination of flawed specimens. Signals can be difficult to interpret and as detailed in the regulatory analysis for this rulemaking, experience and studies indicate that the examiner must practice on a frequent basis to maintain the capability for proper interpretation. The second reason is related to the length of training and its frequency. Studies have shown that an examiner's capability begins to diminish within approximately 6 months if skills are not maintained."

Thus, the NRC has determined that 10 hours of annual training is not sufficient practice to maintain skills and that annual Ultrasonic training shall be conducted in accordance with 10 CFR 50.55a(b)(2)(xiv) as amended in the Federal Register (Volume 64, No. 183 dated September 22, 1999) in lieu of ASME Section XI, 1995 Edition, 1996 Addenda, Appendix VII, Subparagraph VII-4240."

The latest amendment to 10 CFR 50.55a (Volume 67, No. 187 dated September 26, 2002), paragraph (b)(2)(xiv) further recognizes, and permits use of, analyzing prerecorded data from material or welds that contain cracks for meeting annual training requirements. However, these provisions apply to those sites implementing use of the 1999 Addenda through the latest Edition and Addenda referenced in paragraph (b)(2) of the Rule; Monticello is using the 1995 Edition with the 1996 Addenda as the Code of Record for the 4th ISI Interval.

Alternative Requirement:

Pursuant to 10 CFR 50.55a(a)(3)(i), Monticello proposes to use the more rigorous and detailed annual training requirements of 10 CFR 50.55a(b)(2)(xiv) in lieu of annual training requirements Appendix VII, paragraph VII-4240.

Therefore, all personnel qualified for performing Ultrasonic examinations in accordance with Appendix VIII - Supplements ASME Section XI, 1995 Edition, 1996 Addenda shall receive 8 hours of annual hands-on training on specimens that contain cracks or by analyzing prerecorded data from material or welds that contain cracks. This training will be completed no earlier than 6 months prior to performing ultrasonic examinations at the Monticello Nuclear Generating Unit.

Justification for Granting Relief:

This relief improves the performance of Appendix VIII - Supplement examinations by requiring NDE examiner performing Appendix VIII examinations to demonstrate proficiency by analyzing specimens that contain cracks or prerecorded ultrasonic data from material or welds that contain cracks prior to performing actual examinations. The proposed alternative will simplify record keeping, satisfy the needs of maintaining Ultrasonic examiner skills, and also provides an acceptable level of quality and safety.

Implementation Schedule:

The proposed alternative is requested for the 4th Ten-Year Interval of the Inservice Inspection Program for Monticello Nuclear Generating Unit.

References:

1. ASME Boiler and Pressure Vessel Code, Section XI, 1995 Edition with 1996 Addenda

2. Federal Register, Rules and Regulations, September 22, 1999 (64 FR 51370)

3. Federal Register, Rules and Regulations, September 26, 2002 (67 FR 60520)

4. NRC Letter to Nuclear Management Company, "Relief Request Nos. 3 and 6 for the Fourth 10-Year Interval of the Inservice Inspection Examination Plan" (TAC No. MB6896), March 28, 2003

Status:

Approved on March 28, 2003 for use during the 4th Interval. (See Reference 4 above)

RELIEF REQUEST NUMBER: ISI No. 7

COMPONENT IDENTIFICATION

Code Classes:	1, 2, and 3
References:	IWA, IWB, IWC, IWD, and IWF -4000 (IWX-4000)
Examination Category:	Not Applicable
Item Number:	Not Applicable
Description:	Use of the 2001 Edition of Section XI to Govern Repair/Replacement Activities and Procedures (IWX-4000).
Component Numbers:	All Class 1, 2, 3 and MC pressure retaining components and their supports.

CODE REQUIREMENT

IWX-4000 (ASME Section XI 1995 Edition with the 1996 Addenda, used for Class 1, 2, and 3 components) provides the rules and requirements for repair/replacement activities associated with pressure retaining components and their supports, including appurtenances, subassemblies, parts of a component, core support structures, metal containments and their integral attachments, and metallic portions of Class CC containments and their integral attachments.

IWX-4000 (ASME Section XI 1992 Edition with the 1992 Addenda, used for IWE components) provides the rules and requirements for the repair of pressure retaining components and their supports, including appurtenances, subassemblies, parts of a component, core support structures, metal containments and their integral attachments, and metallic portions of Class CC containments and their integral attachments, by welding, brazing, or metal removal. This article also provides the rules and requirements for the specification and construction of items to be used for replacements and installation of replacement items.

10 CFR 50.55a dated September 6, 1996 required the implementation of Subsections IWE and IWL of the 1992 Edition with the 1992 Addenda.

BASIS FOR RELIEF

The 1992 Edition with the 1992 Addenda to Section XI made several changes to Articles IWX-4000. Very few of these changes were technical in nature. Instead, the changes restructured some of the requirements, (ie. Combined IWX-4000 and IWX-7000 into one section) clarified others that were difficult to interpret, and eliminated redundant requirements. Of the actual technical changes made, these changes either added enhancements to the program or added requirements not applicable to Monticello.

Meeting both the 1995 with the 1996 Addenda and the 1992 with the 1992 Addenda of ASME Section XI would require the maintenance of two separate repair and replacement programs (one for the IWB, IWC, and IWD components per the 1996 Addenda of ASME Section XI and one for the 1992 Addenda for the containment vessel). Duplicate records to demonstrate compliance with the 1996 Addenda and the 1992 Addenda would also be required. This duplication of programs and records increases the man-hours necessary to maintain the Monticello Repair/Replacement Program without providing any increase in quality or safety.

The final rule (Federal Register/Vol. 67, No. 187, dated September 26, 2002) incorporates reference to the 1998 Edition through 2000 Addenda. Attached is a reconciliation of the changes made and a comparison of the 2001 Edition to the 2000 Addenda of Section XI. Each change related to Repair/Replacement Activities is addressed in the attachment to show it will be implemented at Monticello.

ALTERNATE EXAMINATION

This alternative is requested in accordance with 10CFR 50.55a(a)(3)(ii). Monticello Nuclear Generating Plant will use the 2001 Edition of ASME Section XI, to govern Repair/Replacement Procedures (IWX-4000) for Class 1,2,3, and MC pressure retaining components and their supports. Using the requirements contained in the 2001 Edition of ASME Section XI for Repairs/Replacements at the Monticello Nuclear Generating Plant will maintain the safety of the plant. The following table indicates the implementation of the 2001 Edition for Repair/Replacement Activities.

Article	Topic	Bases
IWA-1000	Scope and Responsibility	1996 Addenda
IWA-2000	Examination and Inspection	1996 Addenda
IWA-3000	Acceptance Standards	1996 Addenda
IWA-4000	**Repair/Replacements**	**2001 Edition**
IWA-5000	Pressure Tests (Periodic)	1996 Addenda
IWA-5000	**Pressure Tests (Repair/Replacements)**	**2001 Edition**
IWA-6000	**Records**	**2001 Edition**
IWA-9000	**Glossary**	**2001 Edition**
IWB-1000	Scope and Responsibility	1996 Addenda
IWB-2000	Examination and Inspection	1996 Addenda
IWB-3000	Acceptance Standards	1996 Addenda
IWB-5000	Pressure Tests (Periodic)	1996 Addenda
IWB-5000	**Pressure Tests (Repair/Replacements)**	**2001 Edition**
IWC-1000	Scope and Responsibility	1996 Addenda
IWC-2000	Examination and Inspection	1996 Addenda
IWC-3000	Acceptance Standards	1996 Addenda
IWC-5000	Pressure Tests (Periodic)	1996 Addenda
IWC-5000	**Pressure Tests (Repair/Replacements)**	**2001 Edition**
IWD-1000	Scope and Responsibility	1996 Addenda
IWD-2000	Examination and Inspection	1996 Addenda
IWD-3000	Acceptance Standards	1996 Addenda
IWD-5000	Pressure Tests (Periodic)	1996 Addenda
IWD-5000	**Pressure Tests (Repair/Replacements)**	**2001 Edition**

Article	Topic	Bases
IWE-1000	Scope and Responsibility	1992 Addenda
IWE-2000	Examination and Inspection	1992 Addenda
IWE-3000	Acceptance Standards	1992 Addenda
IWE-5000	Pressure Tests (Periodic)	Appendix J
IWE-5000	**Pressure Tests (Repair/Replacements)**	**2001 Edition w/ Appendix J**
IWF-1000	Scope and Responsibility	1996 Addenda
IWF-2000	Examination and Inspection	1996 Addenda
IWF-3000	Acceptance Standards	1996 Addenda
IWF-5000	Snubber Examinations and Tests	1996 Addenda

APPLICABLE TIME PERIOD

Relief is requested for the fourth ten-year interval of the Inservice Inspection Program for Monticello Nuclear Generating Plant.

Certificate of Reconciliation

The Certificate of Reconciliation provides the basis for revisions to the Monticello Nuclear Generating Plant's (MNGP) ASME Section XI "Repair/Replacement Program" (4AWI-09.04.03) in order to meet the 2001 Edition of ASME Section XI. On September 9, 1996, the Nuclear Regulatory Commission (NRC) issued a revision to 10 CFR 50.55a, implementing subsections IWE and IWL (IWL "Requirements for Class CC Concrete Components of Light-Water Cooled Plants" is not applicable to the Monticello Nuclear Generating Plan) is not of the 1992 edition, including the 1992 addenda of Section XI of the ASME Code. This required utilities to develop and implement a program for the examination of containments by September 9, 2001. Additionally, it required implementation of an IWE/IWL repair/replacement program effective September 9, 1996. The NMC is updating the MNGP Inservice Inspection (ISI) Program for the fourth ten-year interval to meet the 1995 Edition with the 1996 Addenda. Because of the hardship to maintain two separate Repair/Replacement Programs, this alternative is proposed to allow the use of the 2001 Edition of ASME Section XI. This reconciliation is completed to provide justification for allowing the use of the 2001 Edition for Class 1, 2, 3 and MC pressure retaining components and their supports.

The current revision of 10CFR50.55a requires ASME Section XI Programs to follow the 1995 Edition as amended by the 1996 Addenda of ASME Section XI for Class 1, 2, and 3 components and the 1992 Edition as amended by the 1992 Addenda for Class MC components. There are some general issues to discuss prior to delineating the specific changes that have been made to the ASME Section XI Code (2000 Addenda to 2001 Edition). By performing the reconciliation from the 1992 Addenda, the reconciliation from the 1996 Addenda is covered as well.

1) The NRC has reviewed and approved with some exceptions the 1998 Edition through 2000 Addenda of the code. This has been included in the Final Rule (dated September 26, 2002). Those specific exceptions made to the rules for repair/replacement activities are included in the implementation of the 2001 Edition.
2) The NMC ISI requirements for MNGP will be based on the 1995 Edition as amended by the 1996 Addenda.
3) The Periodic Pressure Testing requirements will be based on the 1995 Edition as amended by the 1996 Addenda. While the pressure testing requirements for repair/replacement activities will be based on the 2001 Edition.
4) The reconciliation attached addresses the changes contained within the IWA-4000 paragraphs. In addition, any significant changes identified within any related requirements are addressed.

Each change is categorized as:

Editorial (E) – Those changes that are of an editorial nature like typographical errors or misspelled words.
Technical Significant (TS) – Those changes that effect the technical requirements and either reduce or increase those requirements. These changes are described in more detail as to their applicability to MNGP.

Technical (T) – Technical changes that are only used for clarification of an existing requirement.

Non-significant (TN) – Those changes that are not technical in nature, but could not be classified as editorial or just a relocation of existing requirements.

ISI RELIEF REQUEST NUMBER: No. 7 Certificate of Reconciliation		
2001 Edition		
IWA-4110(b)	Revised to insert the words "Thermal metal removal" to clarify that thermal metal removal activities fall within the scope of IWA-4000	TS (Note 1)
IWA-4230	This was added to relocate the requirements of IWA-4451 "Helical Coil Threaded Inserts". This relocation places these requirements in IWA-4200 "Material" which is appropriate since they deal primarily with helicoil material requirements.	TN
IWA-4400	Retitled to "Welding, Brazing, Metal Removal, and Installation". This was retitled specify that metal removal rules apply to all Section XI repair activities.	TN
IWA-4410	This was rewritten to make its contents consistent with the revised title. It is also revised to clarify that mechanical metal removal not associated with defect removal is not within the scope of IWA-4400.	T
IWA-4411	This is a new paragraph titled "Welding and Brazing". This new paragraph serves to consolidate the requirements applicable only to welding and brazing, and to clarify the distinction between when Construction Code requirements apply and when IWA-4400 requirements apply.	T
IWA-4412	This is a new paragraph titled "Defect Removal". This new paragraph serves to clarify that the requirements of IWA-4420 are mandatory for all defect removal activities, and to direct the user to these requirements.	T

ISI RELIEF REQUEST NUMBER: No. 7 Certificate of Reconciliation		
IWA-4413	This is a new paragraph titled "Thermal Metal Removal". This new paragraph serves to clarify that the requirements of IWA-4461 are mandatory for all thermal metal removal activities, and to direct the user to these requirements.	T
IWA-4420	Revised title to "Defect Removal Requirements". This revision makes the title consistent with the changes described below.	TN
IWA-4421	Revised to "General Requirements" with the following specific changes: i) The second sentence of para. (a) is moved to IWA-4421. ii) The last sentence of para. (a) is dropped, since IWA-4412 now invokes requirements for defect removal and associated NDE. iii) The remainder of the text from IWA-4421(a), (b), and (c) is reorganized and moved to IWA-4411(a) and (b), except that the final sentence, "A Report of Reconciliation shall be prepared." has been deleted to make this paragraph consistent with the changes made.	TN

ISI RELIEF REQUEST NUMBER: No. 7 Certificate of Reconciliation		
IWA-4422	Revised to "Defect Evaluation and Examination". This change makes the title consistent with the content changes described for IWA-4422.1. IWA-4421.1 was changed as follows: i) Title changed to "Defect Evaluation" ii) The first sentence of IWA-4422.1(a) is deleted. The requirement that the defect removal process comply with 4421 is unneeded, as it is redundant with the new IWA-4421 (a) through (d) iii) The third sentence of IWA-4422.1(a) is deleted. This deleted sentence stated, "The component is acceptable for continued service if the resulting section thickness created by the cavity is at least the minimum required thickness." This sentence is deleted for two reasons: 1) It is redundant with the proceeding sentence in IWA-4422.1(a) and 2) It implies that all defect removal operations involve metal removal and creation of a cavity. Several repair types do not involve metal removal or cavity creation.	TN
IWA-4430	This paragraph was deleted. Its contents were reworded and relocated to IWA-4411(f).	TN
IWA-4450	This was deleted from the Code in its entirety. Use of the ASME Code to mandate compliance with manufacturer's recommendations is considered inappropriate and constitutes the basis for deleting this requirement.	TN
IWA-4451	This was renumbered as IWA-4134 and is relocated accordingly. This relocation is consistent with the contents of IWA-4451, which address installation of helical-coil threaded inserts. The installation of helical-coil threaded inserts does not fall within the scope of IWA-4400.	TN
Table IWA-4461.1-1	This table was revised to delete reference to P-1 materials. This revision is editorial in nature, and is incorporated to make Table-4461.1 consistent with IWA-4461.1 and 4461.2. The revision for preheat of P-1 materials prior to thermal metal removal was deleted by a prior revision to IWA-4460, but Table IWA-4461.1 was not revised to reflect this revision.	E

ISI RELIEF REQUEST NUMBER: No. 7 Certificate of Reconciliation		
IWA-4461.4	Title was revised to "Alternatives to Mechanical Processing". This change is necessary to accommodate a newly added alternative to mechanical processing after thermal metal removal, which is addressed in IWA-4461.4.2. The two alternatives are addressed in new paragraphs IWA-4461.4.1 and IWA-4461.4.2. IWA-4461.4.1 describes the qualification process whereby thermal metal removal is permitted without subsequent mechanical processing. No changes were made to these requirements other than paragraph renumbering. IWA-4461.4.2 describes the evaluation process where by thermal metal removal is permitted without subsequent mechanical processing. This alternative enables an Owner to perform a documented evaluation to determine whether elimination of mechanical processing is acceptable. A footnote was added to define the term "Mechanical Processing"	TS (Note 1)
IWA-4462	This was revised to "Mechanical Defect Removal Processes". IWA-4462(a) is replaced with wording that clarifies the applicability of this paragraph to defect removal activities only.	TN
IWA-4500	Title changed to "Examination and Testing"	TN
IWA-4520(a)	This was revised to add two specific exceptions. These exceptions are as follows: i) IWA-4521(a)(1) was revised to exempt Class 3 base material repairs from volumetric examination when full-penetration butt welds in the same location do not require volumetric examination. ii) IWA-4521(a)(2) was revised to invoke the examination requirements of IWA-4600 and 4700 in lieu of Construction Code examinations for all repairs using IWA-4600 or 4700. This exception invokes IWA-4600 NDE requirements for all IWA-4600 welding, and invokes IWA-4700 NDE requirements for IWA-4700 welding. This change clarifies that use of IWA-4600 and IWA-4700 welding alternatives and also mandates use of the associated NDE requirements.	TS (Note 1)

ISI RELIEF REQUEST NUMBER: No. 7 Certificate of Reconciliation		
IWA-4600(a)	This was revised to delete the words "and nondestructive examination requirements". These words are deleted for clarification. The underwater welding alternative requirements of IWA-4660 apply in lieu of Construction Code requirements; however, IWA-4660 invokes Construction Code NDE requirements. Since IWA-4660 invokes Construction Code NDE requirements, it is incorrect to state that 4660's requirements are "in lieu of" Construction Code NDE requirements.	TN
IWA-4610	This was revised to "General Requirements for Temperbead Welding of all Materials"	TN

ISI RELIEF REQUEST NUMBER: No. 7 **Certificate of Reconciliation**		
IWA-4611	IWA-4611.1(a), (b) and (c) were deleted and alternative requirements were added. i) The defect removal requirements of 4611.1(a) have been moved to IWA-4421.1. The existing 4611(a), therefore is redundant and is no longer needed. ii) The IWA-4611.1(b) requirement that "the original defect shall be removed" has been revised to match what the original intent was by the words "the original defect shall be reduced in size to a level that meets the applicable Construction Code NDE acceptance criteria. The requirement for compliance with Construction Code acceptance criteria was added to IWA-4624.2, 4634.2, 4644.2 and 4654.2. iii) The IWA-4611.1(c) requirements for the Repair/Replacement Program and Plan are redundant with IWA-4150. Deletion of this paragraph eliminates this redundancy. IWA-4611.1(a), (b), and (c) additions are as follows: i) IWA-4611.1(a) now consists of a reference to IWA-4422.1. Use of this reference enables all defect removal activities to rely on a single set of defect removal requirements, eliminating redundancy and reducing complexity. ii) IWA-4611.1(b) now includes a reference to the NDE requirements applicable to each of the various repair methods authorized by IWA-4600. This reference is needed because each repair method includes its own unique NDE requirements, and these requirements are different from those used for welding and brazing activities that are not within the scope of IWA-4600. i) IWA-4611.1(c) now includes a reference to the thermal metal removal requirements of IWA-4413. This reference is needed because the requirements for thermal metal removal apply to all IWA-4600 processes, and because thermal metal removal requirements have been consolidated into IWA-4461, which is referenced by IWA-4413.	TS (Note 1)

ISI RELIEF REQUEST NUMBER: No. 7 **Certificate of Reconciliation**		
IWA-4611 (cont'd)	IWA-4611.2(a) was changed as follows: i) In the first line, the word "grinding" is replace with "processing". This change is necessary to acknowledge that final grinding is not always required for defect removal. ii) In the sixth line, "IWA-3000" is replaced with "IWB-3500, IWC-3500, or IWD-3000". This change adds a direct reference to the NDE acceptance criteria tables of IWB and IWC (Note: Since IWD tables are 'in course of preparation', the IWD-3000 reference invokes permission to use IWB requirements). By referencing these tables, IWA-4611.2(a) clarifies that the indication may be considered 'reduced to an acceptable level' only when the respective table's acceptance criteria has been met. A new sentence states, "For supports and containment vessels, the provisions of IWA-4422.1(b) may be used." This sentence is added because ASME Section III Subsections NE and NF do not contain surface examination acceptance criteria for base materials, therefore, no criteria exist for these exams. IWA-4422.1(b) provides an evaluation alternative for these applications.	TS (Note 1)
IWA-4620	Title was revised to "Temperbead Welding of Similar Materials"	TN
IWA-4624	A) IWA-4624.1(a) was added to invoke IWA-4611.2(a), which mandates surface examination prior to welding for all temperbead repairs. This paragraph is added to assure that Section XI, IWA-3000 acceptance criteria is used for NDE of existing metal. B) IWA-4624.2 invokes Construction Code or Section III NDE acceptance criteria on in-processing welding and on the final weld. This assures that all newly installed weld metal complies with Construction Code requirements during installation and at the time of weld completion.	TS (Note 1)
IWA-4630	Title was revised to "Temperbead Welding of Dissimilar Materials"	TN
IWA-4634	This was revised similar to that discussed in IWA-4624 above.	TS (Note 1)
IWA-4644	This was revised similar to that discussed in IWA-4624 above.	TS (Note 1)

ISI RELIEF REQUEST NUMBER: No. 7 Certificate of Reconciliation		
IWA-4654	This was revised similar to that discussed in IWA-4624 above.	TS (Note 1)
IWA-4666	This was revised to impose Construction Code NDE requirements on completed underwater welds. This paragraph also provides an alternative to these NDE requirements when the underwater environment renders normal NDE practical.	TS (Note 1)
IWA-4711.4	This was revised to clarify the final visual examination was to be a VT-1 examination.	TS (Note 1)
IWA-4712	This was revised to make its wording consistent with IWA-4711. This change states that use of these requirements is mandatory for Class 1 applications, but use of these requirements in Class 2 and Class 3 applications is also acceptable.	TN
IWA-4721.1	This was revised to make its wording consistent with IWA-4711. This change states that use of these requirements is mandatory for Class 1 applications, but use of these requirements in Class 2 and Class 3 applications is also acceptable.	TN
IWA-4131.1(a)	The change deleted the word "welded" located in before the reference to plugs.	TS (Note 2)
IWA-4713	This revision adds new requirements for qualification of Class 1 mechanical tube plugs. These requirements represent a compilation of the standards and methods that have been used for twenty years to design, qualify, and install steam generator tube plugs. They have proven to provide safe installation and service for mechanical steam generator tube plugs. These requirements include development and qualification of the plug design and of a Plugging Procedure Specification (PPS), and performance qualification for individuals who install the tube plugs	TS (Note 2)

ISI RELIEF REQUEST NUMBER: No. 7 Certificate of Reconciliation		
IWA-4132	This revision deletes the requirement for pressure testing and VT-2 visual examination of relief valves rotated from stock and installed by mechanical means. In the 1999 Addenda, the requirement to pressure test mechanical joints made in installation of pressure retaining items was deleted from IWA-4540, because Owner's operation and maintenance personnel post-installation inspections are adequate without an additional Code-required examination. With the deletion of pressure tests for mechanical connections, a similar exemption is warranted for installation of relief valves by mechanical means. The revision also clarifies that no other IWA-4000 requirements apply to rotation of snubber and relief valves, except those of IWA-4132, and clarifies that use of an ANII is not required. This revision incorporates the provisions of Case N-508-2, "Rotation of Serviced Snubbers and Pressure Relief Valves for the Purpose of Testing, Section XI, Division 1."	TS (Note 3)

NOTE 1. It is important to apply the correct acceptance criteria to each repair/replacement activity completed. As reflected in the Final Rule, the NRC recognizes the difference between the NDE of the Construction Codes and ASME Section XI. The other changes were made to clarify the rules as they apply to the mechanical removal process and of a non-technical nature with reordering of paragraphs or moving of requirements to different paragraphs. The MNGP Repair/Replacement Program incorporates these requirements.

NOTE 2. NMC has determined that it is important to have all special processes qualified and/or demonstrated to verify the application. Because of the elimination of the word "welded," the alternative requirements provided in IWA-4131.1 are no longer applicable to any tube plugging (mechanical or welded). The MNGP Repair/Replacement Program incorporates these provisions.

NOTE 3. Since the code no longer requires a VT-2 Examination on installation of mechanical joints, the NMC has determined that the installation of relief valves rotated from MNGP stock and installed by mechanical means would not require a VT-2 examination.

NRC Limitation / NMC Commitments:

The NRC staff requires implementation of paragraph IWA-4540(c) of the 1998 edition in lieu of that of the 2001 edition when implementing the 2001 edition of ASME Code, Section XI, Article IWX-4000 for repair and replacement activities.

The NRC is planning revisions to the Final rule which may have an effect on this Relief Request. NMC has committed to implement the limitations and modifications to the 1998 edition through 2000 addenda of the ASME Code, Section XI, as stated in 10 CFR 50.55a(b)(2) when implementing the 2001 edition. NMC has further committed to implement any limitations and modifications to the 2001 edition of the ASME Code for its repair and replacement program when the NRC incorporates, by reference, this edition into the regulations.

References:

1. NRC Letter to Nuclear Management Company, "Fourth 10-Year Interval Inservice Inspection Program Plan Relief Request No.7" (TAC No. MB6897), October 3, 2003

2. NRC Letter to Nuclear Management Company, "Issuance of Corrected Page Fourth 10-Year Interval Inservice Inspection Program Plan Relief Request No.7" (TAC No. MB6897), December 31, 2003

Status:

Approved on October 3, 2003 for use during the 4th Interval, (See References 1 and 2 above)