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TECHNICAL EVALUATION REPORT ON THE THIRD 10-YEAR INTERVAL INSERVICE INSPECTION PROGRAM PLAN: NORTHERN STATES POWER COMPANY, MONTICELLO NUCLEAR GENERATING PLANT, DOCKET NUMBER 50-263

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ABSTRACT

This report presents the results of the evaluation of the *Monticello Nuclear Generating Plant, Third 10-Year Interval Inservice Inspection (ISI) Program Plan*, Revision 1, submitted July 29, 1993, including the requests for relief from the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI requirements that the licensee has determined to be impractical. The *Monticello Nuclear Generating Plant, Third 10-Year Interval Inservice Inspection Program Plan* is evaluated in Section 2 of this report. The ISI Program Plan is evaluated for (a) compliance with the appropriate edition/addenda of Section XI, (b) acceptability of the examination sample, (c) correctness of the application of system or component examination exclusion criteria, and (d) compliance with ISI-related commitments identified during the Nuclear Regulatory Commission (NRC) review before granting an operating license. The requests for relief are evaluated in Section 3 of this report.

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SUMMARY

The licensee, Northern States Power Company, has prepared the *Monticello Nuclear Generating Plant, Third 10-Year Interval Inservice Inspection (ISI) Program*, Revision 1, to meet the requirements of the 1986 Edition of the ASME Code Section XI. The third 10-year interval began June 1, 1992 and ends May 31, 2002.

The information in the *Monticello Nuclear Generating Plant, Third 10-Year Interval Inservice Inspection Program*, Revision 1, submitted July 29, 1993, was reviewed. Included in the review were the requests for relief from the ASME Code Section XI requirements that the licensee has determined to be impractical. As a result of this review, a request for additional information was prepared describing the information and/or clarification required from the licensee in order to complete the review. The licensee provided the requested information in the submittal dated December 20, 1993.

The results of the *Monticello Nuclear Generating Plant, Third 10-Year Interval Inservice Inspection Program*, Revision 1, review showed compliance with the Code, except as discussed in paragraph 2.2.2.

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1. INTRODUCTION

Throughout the service life of a water-cooled nuclear power facility, 10 CFR 50.55a(g)(4) (Reference 1) requires that components (including supports) that are classified as American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Class 1, Class 2, and Class 3 meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in the ASME Code Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components" (Reference 2), to the extent practical within the limitations of design, geometry, and materials of construction of the components. This section of the regulations also requires that inservice examinations of components and system pressure tests conducted during successive 120-month inspection intervals comply with the requirements in the latest edition and addenda of the Code incorporated by reference in 10 CFR 50.55a(b) on the date 12 months prior to the start of the 120-month inspection interval, subject to the limitations and modifications listed therein. The components (including supports) may meet requirements set forth in subsequent editions and addenda of this Code that are incorporated by reference in 10 CFR 50.55a(b) subject to the limitations and modifications listed therein. The licensee, Northern States Power Company, has prepared the *Monticello Nuclear Generating Plant, Third 10-Year Interval Inservice Inspection (ISI) Program*, Revision 1 (Reference 3), to meet the requirements of the 1986 Edition of the ASME Code Section XI, except that the extent of examination for Class 1, Examination Category B-J welds has been determined by the requirements of the 1974 Edition through Summer 1975 Addenda as permitted by 10 CFR 50.55a(b)(2)(ii). The third 10-year interval began June 1, 1992 and ends May 31, 2002.

As required by 10 CFR 50.55a(g)(5), if the licensee determines that certain Code examination requirements are impractical and requests relief from them,

the licensee shall submit information and justification to the Nuclear Regulatory Commission (NRC) to support that determination.

Pursuant to 10 CFR 50.55a(g)(6), the NRC will evaluate the licensee's determination that Code requirements are impractical to implement. The NRC may grant relief and may impose alternative requirements that are determined to be authorized by law, will not endanger life, property, or the common defense and security, and are otherwise in the public interest, giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility.

Alternatively, pursuant to 10 CFR 50.55a(a)(3), the NRC will evaluate the licensee's determination that either (i) the proposed alternatives provide an acceptable level of quality and safety or that (ii) Code compliance would result in hardship or unusual difficulty without a compensating increase in safety. Proposed alternatives may be used when authorized by the NRC.

The information in the *Monticello Nuclear Generating Plant, Third 10-Year Interval ISI Program*, Revision 1, submitted July 29, 1993, was reviewed, including the requests for relief from the ASME Code Section XI requirements that the licensee has determined to be impractical. This review was performed using the Standard Review Plans of NUREG-0800 (Reference 4), Section 5.2.4, "Reactor Coolant Boundary Inservice Inspections and Testing," and Section 6.6, "Inservice Inspection of Class 2 and 3 Components."

In a letter dated October 18, 1993 (Reference 5), the NRC requested additional information that was required in order to complete the review of the ISI Program Plan. The licensee provided the requested information in a letter dated December 20, 1993 (Reference 6). In this response, the licensee, Northern States Power Company, submitted revised Request for Relief No. 7 (Revision 1).

The *Monticello Nuclear Generating Plant, Third 10-Year Interval ISI Program Plan* is evaluated in Section 2 of this report. The ISI Program Plan is evaluated for (a) compliance with the appropriate edition/addenda of Section XI, (b) acceptability of the examination sample, (c) correctness of

the application of system or component examination exclusion criteria, and (d) compliance with ISI-related commitments identified during the NRC's previous reviews. The requests for relief are evaluated in Section 3 of this report.

Unless otherwise stated, references to the Code refer to the ASME Code, Section XI, 1986 Edition. Specific inservice test (IST) programs for pumps and valves are being evaluated in other reports.

2. EVALUATION OF INSERVICE INSPECTION PROGRAM PLAN

This evaluation consisted of a review of the applicable program documents to determine whether or not they are in compliance with the Code requirements and any previous license conditions pertinent to ISI activities. This section describes the submittals reviewed and the results of the review.

2.1 Documents Evaluated

Review has been completed on the following information from the licensee:

(a) *Monticello Nuclear Generating Plant, Third 10-Year Interval Inservice Inspection (ISI) Program*, Revision 1 (Reference 3); and

(b) The response to the NRC's request for additional information dated December 20, 1993 (Reference 6).

2.2 Compliance with Code Requirements

2.2.1 Compliance with Applicable Code Editions

The Inservice Inspection Program Plan shall be based on the Code edition defined in 10 CFR 50.55a(g)(4) and 10 CFR 50.55a(b). Based on the starting date of June 1, 1992, the Code applicable to the third interval ISI program is the 1986 Edition. As stated in Section 1 of this report, the licensee has prepared the *Monticello Nuclear Generating Plant Third 10-Year ISI Program* to meet the requirements of 1986 Edition of the Code, except that the extent of examination for Class 1, Examination Category B-J welds has been determined by the requirements of the 1974 Edition through Summer 1975 Addenda (74S75) as permitted by 10 CFR 50.55a(b)(2)(ii).

2.2.2 Acceptability of the Examination Sample

Inservice volumetric, surface, and visual examinations shall be performed on ASME Code Class 1, 2, and 3 components and their

supports using sampling schedules described in Section XI of the ASME Code and 10 CFR 50.55a(b).

Regulatory Guide (RG) 1.26, *Quality Group Classification and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants* (Reference 7), provides guidance criteria for the classification of Class 2 and 3 systems. Paragraph C.1.c of RG 1.26 states that Group B (Class 2) quality standards should be applied to:

> "Those portions of the steam systems of boiling water reactors extending from the outermost containment isolation valve up to but not including the turbine stop and bypass valves and connected piping up to and including the first valve that is either normally closed or capable of automatic closure during all modes of normal reactor operation. Alternately, for boiling water reactors containing a shutoff valve (in addition to the two containment isolation valves) in the main steam line and in the main feedwater line, Group B quality standards should be applied to those portions of the steam and feedwater systems extending from the outermost containment isolation valves up to and including the shutoff valve or the first valve that is either normally closed or capable of automatic closure during all modes of normal operation."

Additional clarification can be found in USNRC Standard Review Plan 3.2.2 (Reference 4), Appendix A, *Classification of Main Steam Components Other Than the Reactor Coolant Pressure Boundary for BWR Plants*, which states that the main steam line from the second isolation valve to the turbine stop valve and the main turbine bypass line to the bypass valve should be Quality Group B.

In the licensee's Program, those portions of the main steam system from the outboard isolation valves are shown as non-classed. In accordance with RG 1.26 and Standard Review Plan 3.2.2, the subject portions of the main steam system to the turbine stop and bypass valves should receive surface and volumetric examinations as required by Subsection IWC of ASME Section XI.

Prior to 1991, the subject piping was treated as safety related (Class 2) and consequently, received Code-required examinations

during Monticello's previous 10-Year ISI intervals. Reclassification of the main steam system will exclude the subject portions of the main steam system from Class 2 volumetric and surface examinations and, therefore, does not meet the intent of RG 1.26. This change of Code classification should be considered unacceptable until the change is reviewed and accepted by the NRC.

In addition, ASME Code Class 2 piping downstream of the core spray system pumps is being excluded from volumetric examination based on pipe wall thickness. The licensee committed to performing a volumetric examination on a 7.5% weld sample in this system during the second 10-year interval. However, only six of the welds in each of the two loops have wall thicknesses greater than 3/8 inch. Consequently, the licensee's sample is being only applied to these 3/8-inch-wall welds. The decision to exclude the "thin-wall" welds from volumetric examination leaves approximately 40 welds in each loop downstream of the core spray system pump susceptible to undetected inservice degradation. A more random distribution would be a better indicator of the system's overall inservice condition.

2.2.3 Exemption Criteria

The criteria used to exempt components from examination shall be consistent with Paragraphs IWB-1220, IWC-1220, IWC-1230, IWD-1220, and 10 CFR 50.55a(b). The exemption criteria have been applied by the licensee in accordance with the Code, as discussed in the ISI Program Plan, and appear to be correct.

2.2.4 Augmented Examination Commitments

In addition to the requirements specified in Section XI of the ASME Code, the licensee has committed to perform the augmented examinations described in the following documents:

(a) NUREG 0313, *Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping*, Revision 2 (Reference 8), as required by Generic Letter 88-01, *Intergranular*

Stress Corrosion Cracking in BWR Austenitic Stainless Steel Piping (Reference 9);

(b) NUREG 0619, *BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking* (Reference 10);

(c) NUREG/CR-3052, *BWR Jet Pump Assembly Failure* (Reference 11); and

(d) Regulatory Guide 1.150, *Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations* (Reference 12).

2.3 Conclusion

Based on the review of the documents listed above, the *Monticello Nuclear Generating Plant, Third 10-Year Interval ISI Program*, Revision 1, showed compliance with the Code, except for the items discussed in paragraph 2.2.2. The licensee should review these items and make changes to the ISI Program, where appropriate.

3. EVALUATION OF RELIEF REQUESTS

The requests for relief from the ASME Code requirements that the licensee has determined to be impractical for the third 10-year inspection interval are evaluated in the following sections.

3.1 Class 1 Components

3.1.1 Reactor Pressure Vessel

3.1.1.1 Request for Relief No. 1 (Revision 1), Examination Category B-A, Item B1.10, Reactor Pressure Vessel (RPV) Shell Welds

Code Requirement: Section XI, Table IWB-2500-1, Examination Category B-A, Item B1.10, requires a 100% volumetric examination of one circumferential and one longitudinal beltline region weld as defined by Figures IWB-2500-1 and IWB-2500-2.

Licensee's Code Relief Request: The licensee requested relief from examining RPV beltline shell welds to the extent required by the Code.

Licensee's Stated Basis for Requesting Relief:

"Only limited outside diameter access to the beltline welds is available through the shield wall nozzle openings.

"Accessibility from the inside diameter is limited due to interference from core spray piping, spargers, guide rods, and welded attachments. Portions of the vessel clad are too rough for ultrasonic examinations."

Licensee's Proposed Alternative Examination (as stated):

"Volumetrically examine one circumferential and one longitudinal beltline weld to the extent practical. This amounts to 12% of the circumferential beltline weld and 40% partial examination of the longitudinal beltline weld.

"In addition, perform a sampling volumetric examination of other longitudinal and circumferential welds to the extent practical. This amounts to 47% of the remaining reactor vessel welds.

"Note: Partial examination means that the weld has not been scanned in all four directions."

Evaluation: In addition to the requirements of the 1986 Code, which requires examination of beltline region welds, 10 CFR 50.55a(g)(6)(ii)(A) mandates an augmented volumetric examination of all RPV shell welds (Item B1.10, Examination Category B-A to the 1989 Edition of the Code) during the third interval at Monticello. The new regulation is requiring that licensees examine essentially 100% of those welds.

10 CFR 50.55a(g)(6)(ii)(A)(1) states that all previously granted reliefs under § 50.55a for the extent of volumetric examination of reactor vessel shell welds are revoked. Because of this regulatory requirement, this relief request cannot be granted. In addition, the augmented RPV examination can serve as a substitute for the examination from which Monticello is requesting relief. The RPV augmented examination should be implemented and, if necessary, an alternative should be proposed under the provisions of 10 CFR 50.55a(g)(6)(ii)(A)(5).

Conclusion: 10 CFR 50.55a(g)(6)(ii)(A) requires an augmented volumetric examination of all RPV shell welds. Furthermore, the NRC Regulatory Analysis of the augmented examination rule concludes that the new requirements will result in a substantial increase in the overall protection of the public health and safety, and that the cost of implementation is justified in view of this increase. Therefore, it is recommended that relief be denied.

3.1.1.2 Request for Relief No. 2 (Rev. 1), Examination Category B-H, Item B8.10, Reactor Pressure Vessel Stabilizer Brackets

Code Requirement: Section XI, Table IWB-2500-1, Examination Category B-H, Item B8.10, requires a 100% surface or volumetric

examination of integrally welded attachments, as defined by Figures IWB-2500-13, -14, or -15.

Licensee's Code Relief Request: The licensee requested relief from performing the Code-required surface examination of the integrally welded RPV stabilizer brackets.

Licensee's Stated Basis for Requesting Relief:

"The vessel stabilizer brackets are surrounded by non-removable insulation, 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."

Licensee's Proposed Alternative Examination: The stabilizer brackets will be examined if they experience local or seismic loads.

Evaluation: The Code requires that the RPV stabilizer brackets receive a surface or volumetric examination each inspection interval. Non-removable insulation, ventilation ductwork, and electrical installations surrounding the RPV make performance of the Code-required examinations impractical. The RPV stabilizer brackets do not provide support during normal operation but are designed to allow thermal movement of the RPV without restraint. Because there is no loading during normal operation, the licensee's proposal, to examine the stabilizer bracket welds if local or seismic loads are experienced, will assure the continued structural integrity of the component.

Conclusion: Based on the impracticality of performing the Code-required examination, it is recommended that relief be granted as requested, pursuant to 10 CFR 50.55a(g)(6)(i).

3.1.1.3 Request for Relief No. 5 (Rev. 1), Examination Category B-E, Item B4.10, Reactor Vessel Partial Penetration Nozzle Welds

Code Requirement: Section XI, Table IWB-2500-1, Examination Category B-E, Item B4.10, requires a VT-2 visual examination of external surfaces of pressure-retaining partial penetration welds in the RPV nozzles during the system hydrostatic test.

Licensee's Code Relief Request: The licensee requested relief from performing the Code-required VT-2 visual examination of Closure Head Flange Leakage Sensor Nozzles N-13 and N-14.

Licensee's Stated Basis for Requesting Relief:

"These nozzles do not see pressure during the vessel pressure test; unless the vessel flange o-ring leaks. Leakage past the o-ring is detected by the RPV head-seal leak detection system. Therefore, a visual examination for leaks during the vessel pressure test will not detect leaks because the nozzles are not pressurized.

"Pressurizing the nozzle in the wrong direction to perform a leak test will damage the vessel flange o-ring.

"The nozzles are not accessible without damaging area insulation."

Licensee's Proposed Alternative Examination: No examinations are proposed unless the insulation is removed for maintenance or other activities.

Evaluation: The subject nozzle welds are isolated from the RPV by the vessel flange o-ring and, consequently, are not exposed to pressure during normal operation or the vessel pressure test, unless the flange o-ring leaks. A localized pressure test that would pressurize the nozzles in the opposite direction would damage the vessel o-ring. Since there is no practical way to pressurize these nozzle welds, the Code requirement is impractical. To perform the examination as required by the Code, the subject nozzles and RPV would have to be redesigned and replaced, which would be a burden on the licensee.

The function of these nozzles is to detect leakage at the RPV head-to-vessel seal by detecting the build-up of pressure and/or liquid. Since any leakage in this region will be detected and annunciated in the control room, immediate action can be taken to correct an abnormal condition. Therefore, the existing leak detection system provides reasonable assurance of inservice integrity.

Conclusion: Pursuant to 10 CFR 50.55a(g)(6)(i), it is concluded that the Code-required VT-2 visual examination of the closure head flange leakage sensor nozzles is impractical to perform at Monticello Nuclear Generating Plant. Based on the above evaluation, it is recommended that relief be granted as requested.

3.1.2 Pressurizer (Does not apply to BWRs)

3.1.3 Heat Exchangers and Steam Generators (No relief requests)

3.1.4 Piping Pressure Boundary (No relief requests)

3.1.5 Pump Pressure Boundary

3.1.5.1 Request for Relief No.3 (Rev. 1), Examination Category B-L-2, Item B12.20, Reactor Recirculation Pump Internal Surfaces

Code Requirement: Section XI, Table IWB-2500-1, Examination Category B-L-2, Item B12.20 requires a VT-3 visual examination of pump internal surfaces of at least one pump in each group of pumps performing similar functions in a system.

Licensee's Code Relief Request: The licensee requested relief from performing the Code-required VT-3 visual examination of the reactor recirculation pump casing internal surfaces.

Licensee's Stated Basis for Requesting Relief:

"Inspection of a reactor recirculation pump internal surface requires disassembly of the pump. Disassembly of the pump results in a large amount of personnel radiation exposure. Disassembly of the pump may result in damage to the pump."

Licensee's Proposed Alternative Examination: The internal surface of a reactor recirculation pump will be visually examined if a pump is disassembled for maintenance. If a recirculation pump is not disassembled for maintenance in the third ten-year interval, this fact will be reported at the end of the interval.

Evaluation: The visual examination requirement for internal surfaces of pumps necessitates complete disassembly of the pump. Performing a visual examination of the reactor recirculation pump casing internal surface is a major effort requiring many hours by skilled maintenance and inspection personnel, which results in excessive radiation exposure. The VT-3 visual examination is performed to determine if unanticipated degradation of the casing is occurring due to phenomena such as erosion, corrosion, or cracking. Later editions and addenda of the ASME Code (after 1988) have eliminated disassembly of pumps for the sole purpose of examining the internal surfaces and state that the internal surface VT-3 visual examination requirement is only applicable for pumps that are disassembled for reasons such as maintenance, repair, or volumetric examination. Therefore, the concept of visual examination of the internal surfaces of the pump casing only if the pump is disassembled for maintenance is acceptable.

Conclusion: It is concluded that disassembly of a pump for the sole purpose of inspection is impractical at Monticello. The licensee's proposed alternative will provide reasonable assurance of the continued operational readiness. Therefore, pursuant to 10 CFR 50.55a(g)(6)(i), it is recommended that relief be granted as requested.

3.1.6 Valve Pressure Boundary

3.1.6.1 Request for Relief No. 4 (Rev. 1), Examination Category B-M-2, Item B12.50, Class 1 Valve Body Internal Surfaces

Code Requirement: Section XI, Table IWB-2500-1, Examination Category B-M-2, Item B12.50, requires a VT-3 visual examination of valve body internal surfaces of valves exceeding NPS 4 inches. Examinations are limited to at least one valve within each group of valves that are of the same size, type (such as globe, gate, or check valves), and manufacturing method, and that perform similar functions in the system (such as containment isolation and system overprotection).

Licensee's Code Relief Request: The licensee requested relief from performing the Code-required VT-3 visual examinations on Class 1 valves.

Licensee's Stated Basis for Requesting Relief:
"Inspection of the internal surfaces of Class 1 valves requires disassembly of the valves. Disassembly of these valves results in a large amount of personnel radiation exposure. Disassembly of the valves may result in damage to the valves."

Licensee's Proposed Alternative Examination: The Class 1 valve internal surfaces will be visually examined if a valve is disassembled for maintenance. If the valves are not disassembled for maintenance, this will be reported at the end of the interval.

Evaluation: The VT-3 visual examination requirement for internal surfaces of valves dictates the disassembly of the valve. Disassembly of the valve for the sole purpose of visual examination of the valve internal surface is a major effort and requires many hours by skilled maintenance and inspection personnel, resulting in excessive radiation exposure. Therefore, the Code requirement is impractical. The VT-3 visual examination

is performed to determine if unanticipated degradation of the casing is occurring due to phenomena such as erosion, corrosion, or cracking. Later editions and addenda of the ASME Code (after 1988) have eliminated disassembly of valves for the sole purpose of examining the internal surfaces and state that the internal surface visual examination requirement is only applicable to valves that are disassembled for reasons such as maintenance, repair, or volumetric examination. Therefore, the concept of visual examination of the internal surfaces of the valve only if the valve is disassembled for maintenance is acceptable.

Conclusion: It is concluded that disassembly of a valve for the sole purpose of inspection is impractical to perform at Monticello. Since no problems have been reported in the industry with regard to valve internal surfaces, the licensee's proposal will provide reasonable assurance of the continued operational readiness. Therefore, pursuant to 10 CFR 50.55a(g)(6)(i), it is recommended that relief be granted as requested.

3.1.7 General (No relief requests)

3.2 Class 2 Components (No relief requests)

3.3 Class 3 Components (No relief requests)

3.4 Pressure Tests

3.4.1 Class 1 System Pressure Tests (No relief requests)

3.4.2 Class 2 System Pressure Tests (No relief requests)

3.4.3 Class 3 System Pressure Tests (No relief requests)

3.4.4 General

3.4.4.1 Request for Relief No. 7 (Rev. 1), Paragraph IWA-5250(a)(2), Corrective Measures for Leakage at Bolted Connections in Control Rod Drive (CRD) Housing Joints

Code Requirement: Section XI, Paragraph IWA-5250(a)(2) requires removal and VT-3 visual examination of all bolting at bolted connections if leakage occurs during a system pressure test. The bolting removed is to be evaluated in accordance with IWA-3100.

Licensee's Code Relief Request: The licensee requested relief from removing the bolts and performing the Code-required VT-3 visual examination at a leaking CRD bolted connection without first performing an evaluation of the leakage.

Licensee's Stated Basis for Requesting Relief:

"The CRD (Control Rod Drive) housing 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 water tight 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 the 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 a critical path item, the additional time needed to depressurize the vessel, remove the bolting, perform the exam, and then repressurize 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)."

"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 conventional gasketed joint.

2) 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. Leakage that is found to be acceptable per the guidance is not considered adverse to quality or safety and need not be corrected before startup. This type of analysis is consistent with Section XI Code paragraph IWB-3142, which allows analysis of the leakage for acceptability.

3) 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. Four of the eight bolts on each housing joint were placed with new bolts in 1991. 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 first and second ten-year inspection interval programs did not include the subject examination requirement. A subsequent code revision which has been approved for use by the NRC, the 1990 Addenda to the 1989 Edition, limits the exam requirement to one bolt nearest the leak. Current Code committee activities include an effort to write a code case that limits this examination even further to specific joints and materials. All of these changes support the conclusion that this code requirement, as written, is overly restrictive and represents a hardship without a compensating increase in the level of quality or safety."

Licensee's Proposed Alternative Examination (as stated):

"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 will be based on guidance provided by General Electric and will be included in the VT-2 tests for the joint (Note: this criteria has been submitted for NRC review).* If the leak is determined to be unacceptable and the joint is disassembled to correct the leak, one of the bolts will be VT-3 examined in accordance with the 1990 Addenda requirement."

Evaluation: The 1986 Edition of the Code requires that the source of leakages detected during the conduct of a system pressure test be located and evaluated for corrective measures. If leakage occurs at a bolted connection, the bolting is to be removed and VT-3 visually examined. The CRD housings, designed by General Electric, have an inherent characteristic in which, "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." Therefore, GE developed guidance for the evaluation of a leaking CRD housing bolted connection to determine if disassembly of the joint is necessary to decrease the leak rate. The licensee is requesting

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*An excerpt from *CRD Operation and Maintenance Guidelines* by N. J. Biglieri, General Electric Co., Nuclear Energy Division, San Jose, CA. 8/30/70, was included in the licensee's submittal but is not included in this document.

relief from the requirement to remove the bolting at a leaking CRD housing connection if an evaluation indicates that the leak would eventually stop when normal operating temperature is reached. In addition, if the joint is disassembled to correct the leak, one of the bolts will be VT-3 examined in accordance with the 1990 Addenda.

The evaluation criteria, developed by GE, state that drip-type leakages with decreasing leak rates will eventually stop leaking and would not necessarily be corrected by disassembly. All other leakage situations signify that maintenance should be performed. The burden associated with imposition of this Code requirement include: considerable personnel radiation exposure (50 to 100 mR/hr), increased outage time, and possibility of leakage upon retest.

The licensee has also proposed to perform a VT-3 visual examination of only one bolt when the joint is disassembled to correct a leak, in accordance with the 1990 Addenda. Because a Boiling Water Reactor does not contain borated water, the corrosion resulting from boric acid is not a concern. The removal of the bolt closest to any leakage should provide a sufficient indication of corrosion if it has occured. The use of the 1990 Addenda for corrective measures for leaking bolted connections will provide an acceptable level of quality and safety.

Conclusions: The licensee's proposed alternative, to evaluate CRD housing leakage to the General Electric acceptance criteria, provides reasonable assurance of operational readiness because any leakage not requiring corrective action will seal after reaching operating temperature. Complying with the Code requirement would cause a burden on the licensee due to the increased personnel radiation exposure and outage time without a compensating increase in the probability of correcting the

leakage. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), it is recommended that the proposed alternative be authorized.

3.5 General

3.5.1 Ultrasonic Examination Techniques (No relief requests)

3.5.2 Exempted Components (No relief requests)

3.5.3 Other

3.5.3.1 Request for Relief No. 6 (Rev. 0), Paragraphs IWA-6220(d)(10) and IWA-7520(a)(8), Use of NIS-2 Forms

Code Requirement: Section XI, Paragraphs IWA-6220(d)(10) and IWA-7520(a)(8) require the use of Form NIS-2 Owner's Report for Repairs and Replacements, as shown in Appendix II.

Licensee's Code Relief Request: The licensee requested relief from the use of the Code-required NIS-2 Form.

Licensee's Stated Basis for Requesting Relief:

"The Monticello plant was designed and built prior to the existence of ASME Section III piping codes. As a result, there is no design basis requirements to have Code stamped pressure piping and components. The NIS-2 Form is intended as a record keeping device for pressure boundary repairs and replacements to piping and components that are Code stamped (i.e., to maintain the Code stamp). Section XI Code editions invoked in Monticello's first and second inspection intervals did not require completion of the NIS-2 Form.

"The 1986 Edition to the Code requires completion of the NIS-2 Form as shown in Appendix II. This is an administrative burden to the Monticello plant with no compensatory increase in quality or safety. The intent of the requirement is to maintain a Code stamp through the use of the NIS-2 Form, and is not applicable to the Monticello plant."

Licensee's Proposed Alternative Examination:

"The Inservice Inspection Summary Report, which is prepared in accordance with the requirements of IWA-6220, will contain a summary of the repairs and replacements conducted since the preceding summary report. The summary will include the work document number, system identification, and a description for each repair and replacement. The repair and replacement summary will include a signature block for the ANII."

Evaluation: The NIS-2 Form is a mandatory appendix to Section XI. As such, its consistent industry-wide use is helpful in establishing a uniform mechanism for evaluating the extent of repairs and replacements for regulatory purposes. The professed administrative burden for Monticello has not been effectively demonstrated.

Conclusions: Based on the above discussion, it is recommended that relief be denied.

4. CONCLUSION

Pursuant to 10 CFR 50.55a(g)(6) or, alternatively, 10 CFR 50.55a(a)(3), it has been determined that certain inservice examinations cannot be performed to the extent required by Section XI of the ASME Code. In the cases for which relief is requested, except Request for Relief No. 1 (Rev. 1), No. 6 (Rev. 0), and No. 7 (Rev. 1), the licensee has demonstrated that specific Section XI requirements are impractical. For Request for Relief No. 1 (Rev. 1) and No. 6 (Rev. 0), it is concluded that the licensee has not provided information to support the determination that the Code requirement is impractical and that requiring the licensee to comply with the Code would result in hardship.

This technical evaluation has not identified any practical method by which the licensee can meet all the specific inservice inspection requirements of Section XI of the ASME Code for the existing Monticello Nuclear Generating Plant. Compliance with all the exact Section XI required inspections would necessitate redesign of a significant number of plant systems, sufficient replacement components to be obtained, installation of the new components, and a baseline examination of these components. Even after the redesign efforts, complete compliance with the Section XI examination requirements probably could not be achieved. Therefore, it is concluded that the public interest is not served by imposing certain provisions of Section XI of the ASME Code that have been determined to be impractical. Pursuant to 10 CFR 50.55a(g)(6), relief is allowed from the requirements that are impractical to implement, or alternatively, pursuant to 10 CFR 50.55a(a)(3), alternatives to the Code-required examinations may be granted provided that either (i) the proposed alternatives provide an acceptable level of quality and safety or that (ii) Code compliance would result in hardship or unusual difficulty without a compensating increase in safety. Relief may be granted only if granting the relief will not endanger life, property, or the common defense and security and is otherwise in the public interest, giving due consideration to the burden upon the licensee, that could result if the requirements were imposed on the facility.

The review of the *Monticello Nuclear Generating Plant, Third 10-Year Interval Inservice Inspection Program*, Revision 1, the licensee's response to the NRC's

request for additional information, and the recommendations for granting relief from the ISI examination requirements that have been determined to be impractical has been completed. The review results showed compliance with the Code, except for the issues discussed in paragraph 2.2.2 of this report. The licensee should review these items and make changes to the ISI Program where appropriate. These changes should include the volumetric and surface examinations required by the Code for Class 2 piping systems for the portions of the main steam system between the outboard containment isolation valves and the turbine stop and bypass valves.

5. REFERENCES

1. Code of Federal Regulations, Title 10, Part 50.

2. American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, Division 1:

   1986 Edition
   1974 Edition through Summer 1975 Addenda

3. *Monticello Nuclear Generating Plant, Third 10-Year Interval Inservice Inspection Program*, Revision 1, submitted July 29, 1993.

4. NUREG-0800, Standard Review Plans, Section 3.2.2, Appendix A, "Classification of Main Steam Components other than the Reactor Coolant Pressure Boundary for BWR Plants," Section 5.2.4, "Reactor Coolant Boundary Inservice Inspection and Testing," and Section 6.6, "Inservice Inspection of Class 2 and 3 Components," July 1981.

5. Letter, dated October 18, 1993, M. Gamberoni (NRC) to R. O. Anderson (Northern States Power), containing request for additional information on the Third 10-Year Interval ISI Program Plan.

6. Letter, dated December 20, 1993, R. O. Anderson (NSP) to Document Control Desk, containing the response to the NRC's request for additional information.

7. Regulatory Guide (RG) 1.26, *Quality Group Classification and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants*, Revision 3, February 1976.

8. NUREG 0313, *Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping*, Revision 2, January 1988.

9. NRC Generic Letter 88-01, *NRC Position on Intergranular Stress Corrosion Cracking in BWR Austenitic Stainless Steel Piping*, January 25, 1988.

10. NUREG 0619, *BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking*, November 1980.

11. NUREG/CR-3052 (Closeout of IE Bulletin 80-07), *BWR Jet Pump Assembly Failure*, November 1984.

12. NRC Regulatory Guide 1.150, *Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations*, Revision 1, February 1983.