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Monticello Nuclear Generating Plant
2807 West County Road 75
Monticello, MN 55362-9637

Operated by Nuclear Management Company LLC

October 10, 2000

10 CFR Part 50
Section 50.55a

US Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

MONTICELLO NUCLEAR GENERATING PLANT
Docket No. 50-263 License No. DPR-22

Request for Relief No. 12 for the
Third 10-Year Interval Inservice Inspection Program

On April 14, 2000, Northern States Power (now Nuclear Management Company - NMC) submitted for NRC review the latest revision (No. 3) of Monticello's third 10-year Inservice Inspection Examination Plan. The purpose of this letter is to request review and approval of ISI Relief Request No. 12 to the third 10-year plan (see attached).

Relief Request No. 12 seeks relief from the inspection of reactor vessel circumferential welds, ASME Code Section XI (1986) Category B-A, Item B.1.11, 100% volume requirement, for the third interval inspections and for the remaining term of the current Monticello license. The requested alternative examination meets requirements for reactor vessel longitudinal (axial) welds, Code Category B-A, Item B.1.12, and calls for inspection of 2-3% of reactor vessel circumferential welds, Code Category B-A, Item B1.11 at the intersection points with the axial welds. This alternative is capable of detecting weld degradation sufficient to insure the integrity of the reactor pressure vessel boundary.

As part of the preparation for this relief request, procedures and operator training associated with systems that could cause low temperature over pressure events were reviewed. As a result of this review, procedure and training enhancements are being made. Accordingly, this letter contains the following commitments which will be completed prior to the next refueling outage and vessel inspection, currently scheduled for the fall of 2001:

1. *Monticello commits to enhance* the reactor feedwater pump (RFP) testing *procedure to further assure the isolation of flow to the vessel.*

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2. *Monticello commits to enhance condensate system operating procedures to identify an LTOP event as a potential consequence of an overfill event.*

3. *Monticello commits to revise vessel pressure testing procedures to provide precautions that ensure proper response to a loss of station power.*

4. *Monticello commits to add a precaution in the RWCU restoration procedure in order to further inform the operations personnel of the potential of an LTOP event occurring during SCRAM recovery.*

5. *Monticello commits to provide training 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.*

Please contact Sam Shirey, Senior Licensing Engineer, at (763) 295-1449 if you require further information.

Byron D. Day

Byron D. Day
Plant Manager
Monticello Nuclear Generating Plant

c: Regional Administrator - III, NRC
NRR Project Manager, NRC
Sr. Resident Inspector, NRC
Minnesota Department of Commerce
J Silberg

Attachment: ISI Relief Request No. 12

ISI Relief Request No. 12
Reactor Vessel Circumferential Shell Welds

SYSTEM: Reactor Vessel
Category: B-A

Class: 1
Item: B 1.11

RV Circumferential Welds: VCBB-4, VCBB-3 and VCBB-2

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 third interval inspections and for the remaining term of the current license for Monticello.

Basis For Relief:

Monticello reactor vessel circumferential welds were not inspected to the essentially 100% volumetric requirements during the first and second 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 third interval ISI inspection plan, Rev. 1 (July 29, 1993), continuance for the first and second interval relief request (RR-01) was requested. This 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 (Ref. 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 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 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 x 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 commits to enhance this procedure 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 commits to enhance condensate system operating procedures 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 commits to revise vessel pressure testing procedures 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 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 commits to add a precaution in the RWCU restoration procedure 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 commits to provide training 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.*

Conclusion:

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 (1989 Edition), 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 (1986) Category B-A, Item B.1.11, 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 (Ref. 1).

Time Period Relief is Requested For:

Relief is requested for the third 10 year interval and for the remaining term of the current Monticello license.

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)(a)(i) for Reactor Vessel Circumferential Shell Weld Examinations (TAC No's MA9299 and MA9300).

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 (Ref. 5)	
		CB&I	B&W
Cu, wt%	0.10 (Ref. 2)	0.10	0.31
Ni, wt%	0.99 (Ref. 2)	0.99	0.59
CF (Chemistry factor)	138.5 (Ref. 3)	134.9	196.7
EOL ID Fluence, x 10^{19} n/cm^2	0.51 (Ref. 4)	0.51	0.095
ΔRT_{NDT}, °F	112.5	109.5	79.8
RT_{NDT} (u) °F	-65.6 (Ref. 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