NMC
Committed to Nuclear Excellence

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

Operated by Nuclear Management Company LLC

May 3, 2001

10 CFR Part 50
Section 50.55a(g)

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

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

Response to NRC Request for Additional Information for
Request for Relief No. 12 for the Third 10-Year
Interval Inservice Inspection Program

Reference 1: NSP letter to NRC, "Request for Relief No. 12 for the Third 10-Year Interval Inservice Inspection Program," dated October 10, 2000.

In Reference 1, Nuclear Management Company (NMC) requested approval of Inservice Inspection (ISI) Relief Request No. 12 to the third 10-year plan for the Monticello Nuclear Generating Plant.

On January 19, 2001, February 13, 2001, and February 22, 2001, conference calls were held between members of the NRC Staff and the Monticello Nuclear Generating Plant Staff. In these calls, the NRC presented several questions that are restated and answered in Attachment 1.

On March 9, 2001, the NRC e-mailed six additional questions which are restated and answered separately in Attachment 2.

Attachment 1 clarifies the basis of Relief Request No. 12. In accordance with 10 CFR 50.55a(g)(6)(ii)(A) and10 CFR 50.55a(g)(4), the examination requirements of Table IWB-2500-1 of the 1989 Edition of Section XI, Division 1, of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code are applicable to Monticello's RPV shell welds. However, the Table IWB-2500-1 requirement to examine 100% of the length of each of the circumferential RPV shell welds is impractical at Monticello and would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Boiling Water Reactor Vessel Inspection Program (BWRVIP) report, BWRVIP-5 and Reference 1, provide an acceptable alternative to these requirements for Monticello's circumferential RPV shell welds. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), permanent relief is requested from the routine examination requirements of 10 CFR 50.55a(g)(4) and the augmented

A047

RPV examination requirements of 10 CFR 50.55a(g)(6)(ii)(A) for the Monticello circumferential RPV shell welds.

Within this letter, Monticello commits to the following:

> *The procedure utilized for bypassing the reactor feed pump (RFP) high reactor level trip to allow reactor feed pump testing will be enhanced to include additional isolation of valves and breakers, to assure that an inadvertent injection cannot occur. This enhancement will be completed prior to the upcoming 2001 refueling outage.*

Please direct any questions on this matter to Sam Shirey, Sr. Licensing Engineer, at (763) 295-1449.

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, Esq.

Attachments 1: Response to NRC Verbal Request for Additional Information to Request for Relief No. 12, Reactor Vessel Circumferential Shell Welds

2. Response to NRC E-mail Request for Additional Information to Request for Relief No. 12, Reactor Vessel Circumferential Shell Welds

Response to NRC Verbal Request for Additional Information to
Request for Relief No. 12,
Reactor Vessel Circumferential Shell Welds

In Reference 1, Nuclear Management Company (NMC) requested approval of Inservice Inspection (ISI) Relief Request No. 12 to the third 10-year ISI plan. On January 19, 2001, February 13, 2001, and February 21, 2001, conference calls were held between members of the NRC Staff, and the Monticello Nuclear Generating Plant Staff. In these calls, the NRC presented Monticello with the following questions:

1. **Provide technical justification to show fluence and chemistry factors used to calculate the mean nil ductility reference temperature (RT_{ndt}) are conservative:**

Fluence:

The projected end of life fluence of 0.51×10^{19} n/cm^2 was calculated based on a power value of 1670 MWt through cycle 18. Starting with cycle 19, a 6% uprated value of 1775 MWt was utilized. The following conservative factors were utilized in calculating the end of life fluence for the current license period including power uprate conditions:

A. The relationship between fluence and effective full power years (EFPY) was assumed to be constant throughout plant life based on an early proportionality factor determined over cycles 1 through 9 (Reference 4). Fluence = C x EFPY, where C is defined as the proportionality factor/conversion constant. Utilizing this same proportionality factor for later cycles is conservative for the following reasons:

   a. In more recent cycles, core fuel enrichment has increased as bundle designs have become more efficient. This would lower the value of C, making the value for fluence more conservative.

   b. Fuel loading pattern design philosophy has evolved since cycle 9, resulting in lower neutron leakage, and lower vessel fluence. This would also make the value for fluence more conservative.

B. A 20% correction factor was added to the nominal specimen fluence to correct for azimuth variations and instrument inaccuracies.

Therefore, the utilized fluence value of 0.51×10^{19} is conservative.

Initial RT_{ndt}/Chemistry Factor:

The following conservative factors were utilized in determining the initial RT_{ndt} and chemistry factor for the Monticello reactor pressure vessel (RPV) beltline seam welds:

The Monticello RPV was made using the shielded metal arc welding (SMAW) process. Since records provided with the weld rod did not include Charpy test data, it was not possible to determine specific RT_{ndt} values for the Monticello beltline welds. Reference 2 identified the mean RT_{ndt} for the beltline welds as –65.6 °F. That value was determined from mean nil ductility transition temperature (NDTT) data, using drop weight testing for a large number of SMAW weld heats produced by Alloy Rods Corporation. Alloy Rods Company provided the weld rod used for the Monticello RPV seam welds.

The Charpy data for irradiated weld metal from the first Monticello surveillance program capsule (Reference 3) indicated an RT_{ndt} of –75°. This provides further justification that the utilized initial RT_{ndt} value of –65°F is acceptable.

Alloy Rods (later "The ESAB Group") and GE were also contacted to obtain data on copper and nickel content in the weld rod. This was used to determine a chemistry factor following Regulatory Guide 1.99, Revision 2, guidelines. The ESAB Group reported that (Reference 8) the most limiting copper content in their population of applicable SMAW weld rods was 0.10%. The most limiting nickel content was reported by GE (Reference 3) as 0.99%. These values were used as the basis for determining the Monticello chemistry factor. Note that actual measured weld metal specimen values from the first Monticello surveillance program capsule were lower at 0.06% copper and 0.95% nickel (Reference 4). Therefore, the utilized chemistry factor is conservative.

Table 3-4 of Reference 5 demonstrates the method used to calculate plant specific chemistry factors. This method is in accordance with Regulatory Guide 1.99, Revision 2, Position 2.1, for plate and weld metal. For the C2220 Monticello plate material, a plant specific chemistry factor of 130.8°F was determined. This was slightly larger (more conservative) than the chemistry factor predicted from Table 2 of the Reg. Guide. The ratio of the Reg. Guide chemistry factor to the reference plate chemistry factor was applied to each material to "adjust" measured shift to the reference plate. The methodology of Position 2.1 of the Reg. Guide was then used to formulate a plant specific chemistry factor for the other plates and for the weld metal. The plant specific chemistry factor of 138.5°F was determined for weld metal of the limiting chemistry (0.10% Cu and 0.99% Ni). That value is slightly larger than the 134.9°F chemistry factor from Table 1 of the Reg. Guide, and is much larger than the 82°F chemistry factor that would apply if the actual Monticello weld metal analysis of 0.06% Cu and 0.96% Ni were used.

Based on the above discussion, the fluence and chemistry factors used to calculate the mean nil ductility reference temperature (RT_{ndt}) are conservative.

2. **Identify that Monticello is considered the limiting Chicago Bridge and Iron (CB&I) vessel based on the data listed in Reference 6, or identify which CB&I vessel is most limiting so that an accurate comparison to the bounding conditions can be made.**

   Reference 7 reviewed the effects of plant-to-plant variation in material chemistry, RT_{ndt}, and fluence for all domestic operating BWR plants. Reference 7 does not specifically align data with plants. However, the Monticello data for copper, nickel, fluence, and RT_{ndt} initial, as reported in Reference 7, match the limiting data of Reference 6. Therefore, Monticello is considered to be the most limiting CB&I vessel.

3. **Clarification is required with the legal aspects of the relief request. This deals with identifying the proper sections of the Code of Federal Regulations for which relief is being requested for the permanent deferral of all volumetric inspections of the circumferential shell welds in the reactor vessel.**

   Reference 1 indicated that relief was being requested from 10 CFR 50.55a(g)(6)(ii)(A). This section contains the augmented examination requirement to perform a one time volumetric examination of essentially 100% (>90%) of all circumferential and axial RPV shell assembly welds. Reference 1, however, failed to mention that relief is also required from the normal ISI volumetric examinations required by 10 CFR 50.55a(g)(4).

   In accordance with 10 CFR 50.55a(g)(6)(ii)(A) and10 CFR 50.55a(g)(4), the examination requirements of Table IWB-2500-1 of Subsection IWB of the 1989 Edition of Section XI, Division 1, of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code are applicable to Monticello's RPV shell welds. However, the Table IWB-2500-1 requirement to examine 100% of the length of each of the RPV shell welds cannot be met at Monticello. Boiling Water Reactor Vessel Inspection Program (BWRVIP) report, BWRVIP-5 and our Relief Request No. 12 (Reference 1), provide an acceptable alternative to these requirements for Monticello's circumferential RPV shell welds. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), relief is requested for the third 120-month inspection interval from the routine examination requirements of 10 CFR 50.55a(g)(4). Relief is also requested for the augmented RPV examination requirements of 10 CFR 50.55a(g)(6)(ii)(A) for the Monticello circumferential RPV shell welds.

4. Clarification is required for identifying an acceptable alternative to the examination requirements as specified in 10 CFR 50.55a(g) that would provide an acceptable level of quality and safety.

Generic Letter98-05 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 and the additional information provided herein demonstrate that Monticello conforms to these two criteria. Therefore, the alternative approach provided by GL 98-05 is justified.

References:

1. NSP letter to NRC, "Request for Relief No. 12 for the 3rd 10-Year Interval Inservice Inspection Program," dated October 10, 2000.

2. General Electric Report SASR 88-99 Rev 1, January 1989, Implementation of Regulatory Guide 1.99, Revision 2 for the Monticello Nuclear Generating Plant

3. Monticello Nuclear Generating Plant Information on Reactor Vessel Material Surveillance Program, GE Report NEDO-24197, Revision 1, dated October 1979

4. Examination Testing and Evaluation of Irradiated Pressure Vessel Surveillance Specimens from the Monticello Nuclear Generating Plant, Battelle Columbus Laboratories Report BCL-585-84-2, Revision 1, dated November 1984

5. Structural Integrity Report No. SIR–97-003 Rev. 2, "Review of the Test Results of Two Surveillance Capsules and Recommendations for the Materials Properties and Pressure – Temperature Curves to be Used for the Monticello Reactor Pressure Vessel," dated October 1998

6. NRC Supplement to Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report (TAC No. MA3395), dated March 7, 2000

7. BWRVIP Response to NRC Request for Additional Information on BWRVIP-05, dated December 18, 1997

8. The ESAB Group letter to Northern States Power Company, "Copper Content of 8018-NM for 1967 and 1968," dated May 17, 1994

Response to NRC E-mail Request for Additional Information to
Request for Relief No. 12,
Reactor Vessel Circumferential Shell Welds

In Reference 1, Nuclear Management Company (NMC) requested approval of Inservice Inspection (ISI) Relief Request No. 12 to the third 10-year ISI plan. On March 9, 2001, E-mail was received at Monticello from the NRC Staff. This E-mail presented questions that are repeated and answered below:

1. **What is the CRD flowrate?**

   The nominal flowrate of each control rod drive (CRD) pump is 74 gallons per minute (gpm) at a discharge pressure of 1625 psig. During a vessel pressure test, the flow rate to the vessel varies and is dependent on the rate of flow being discharged through RWCU. The operators maintain RPV pressure by balancing CRD and RWCU flow. (Reference 1)

2. **What is the SLCS flowrate?**

   The nominal flowrate of each standby liquid control system (SBLC) pump is 28.5 gpm at a discharge pressure of 1500 psig. (Reference 2)

3. **What are the discharge pressures of the LPCI pumps, the core spray pumps, the feedwater pumps, and the condensate pumps?**

Pump	Discharge Pressure	Reference
Low Pressure Coolant Injection (LPCI)	Shutoff head is 352 psig	Tech Manual NX- 7905-18
Core Spray	Shutoff head is 365 psig	Tech Manual NX-7833-33
Feed Water	Shutoff head is 1,429 psig	Design change 77M090
Condensate	Shutoff head is 433 psig	Tech Manual NX-17440

4. **Regarding the "Inadvertent Injections" section on Page 3 of 7** [of Reference 1]**, provide additional details about the procedures and controls that will be put in place to prevent a cold over-pressure event from occurring during reactor feed pump testing.**

   For the reactor feed pump to be tested at water levels above the high water trip setpoint of 48", the high level trip interlock must be bypassed. To accomplish this, a specific procedure is utilized which currently provides a single isolation to prevent inadvertent injection.

This procedure will be enhanced to provide additional isolations to assure an inadvertent injection cannot occur. The revision of this procedure will be completed prior to the upcoming 2001 refueling outage.

5. What procedures are in place to prevent inadvertent injection by reactor feed pump or condensate pump?

For the reactor feed pumps, an inadvertent injection with the vessel water level greater than +48 inches is controlled by a high water level interlock. Generally during outages, the vessel level is maintained greater than +48 inches therefore preventing a feedwater pump from starting unless the bypass switch is placed in bypass. This switch position is controlled and would only be bypassed if the proper isolation were in place to prevent injecting into the vessel (see question 4 above). During the vessel pressure test, the bypass switch and the feedwater pumps are isolated and tagged.

For the condensate pumps, precautions are provided in the operating procedures which indicate to the operators that they need to monitor reactor water level closely when the pumps are supplying feed to the reactor vessel in order to prevent an overfill event. However, since the shutoff head of the condensate pumps is only 350 psig, a concern with an LTOP event occurring from the condensate system is minimal. Monticello also has high reactor water level and high reactor pressure alarms in the control room that warn operators when level/pressure limitations are being exceeded.

6. Address the potential for overfill due to LPCI or core spray injection.

Pumps in the Low Pressure Coolant Injection (LPCI) and Core Spray (CS) systems have shutoff heads less than 370 psig. Because of this overpressurizing the reactor pressure vessel (RPV) is not possible since the pressure temperature concerns associated with brittle fracture during cold shutdown conditions do not come into affect until reactor pressure is greater than 500 psig.

The potential for overfilling the reactor pressure vessel with LPCI or CS below 370 psi is low due to plant procedural controls, alarms and instrumentation which are closely monitored by licensed operators.

References:

1. NSP letter to NRC, "Request for Relief No. 12 for the Third 10-Year Interval Inservice Inspection Program," dated October 10, 2000.

2. Operations Manual Section: CRD Hydraulic System B.1.3-02 Description of Equipment, page 3.

3. Operations Manual Section: Standby Liquid Control System B.3.5-02 Description of Equipment, page 5.