NMC
Committed to Nuclear Excellence

Monticello Nuclear Generating Plant
Operated by Nuclear Management Company, LLC

April 28, 2003

L-MT-03-030
10 CFR Part 50
Section 50.55a(a)(3)

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

MONTICELLO NUCLEAR GENERATING PLANT
DOCKET 50-263
LICENSE No. DPR-22

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING INSERVICE INSPECTION RELIEF REQUEST NO. 5 FOR THE FOURTH 10-YEAR INSERVICE INSPECTION INTERVAL – LEAKAGE AT BOLTED CONTROL ROD DRIVE (CRD) HOUSING CONNECTIONS (TAC No. MB6956)

Reference 1: NMC Letter to NRC, "Request For Review and Approval of Relief Requests Associated with Fourth 10-Year Interval Inservice Inspection Examination Plan Submittal", dated December 6, 2002

Reference 2: NRC letter to NMC, "Monticello Nuclear Generating Plant – Request for Additional Information Related to ISI Relief Request No. 5 (TAC No. MB6956)," dated April 24, 2003

Reference 1 requested NRC approval of an alternative examination in lieu of the examination requirements in American Society of Mechanical Engineers (ASME) Section XI, IWA-5250(a)(2) for leakage at bolted connections. The proposed alternative examination would allow the use of an acceptance criteria methodology for leakage at Control Rod Drive (CRD) bolted connections based on the unique design of three hollow metal O-rings in the Monticello Nuclear Generating Plant (MNGP) CRD design, coincident with the low temperature testing of the CRD connections. This relief had previously been granted for the MNGP Third 10-Year Interval Inservice Inspection Plan.

Reference 2 requested the NMC to provide additional information regarding the subject relief request.

Attachment A to this letter contains the NMC response to the requests for additional information.

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This letter does not contain any new commitments.

If you have any questions regarding this submittal please contact John Fields, Senior Licensing Engineer at (763) 295-1663.

David L. Wilson
Site Vice President
Monticello Nuclear Generating Plant

cc: Regional Administrator-III, NRC
NRR Project Manager, NRC
Sr. NRC Resident Inspector, NRC
State of Minnesota Boiler Inspector
Hartford Insurance

Attachment A - NMC RESPONSES TO NRC REQUEST FOR ADDITIONAL INFORMATION

Attachment A

NUCLEAR MANAGEMENT COMPANY, LLC
MONTICELLO NUCLEAR GENERATING PLANT
DOCKET 50-263

April 28, 2003

NMC RESPONSES TO NRC REQUEST FOR ADDITIONAL INFORMATION

3 pages follow

NMC RESPONSES TO NRC REQUEST FOR ADDITIONAL INFORMATION

NRC Request #1:

As a basis for Request for Relief No. 5 dated December 6, 2002, NMC stated that the control rod drive 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 degrees F, as compared to a normal operating temperature of about 540 degrees F.

Please confirm that NMC is following American Society of Mechanical Engineers Code, Section XI, IWA-5000, IWB-5240 temperature requirements when performing periodic reactor pressure vessel leakage and hydrostatic pressure tests.

NMC Response:

American Society of Mechanical Engineers (ASME) Section XI, IWA-5212(g) states "*The system test pressure and temperature may be obtained by using any means that comply with the plant technical specifications.*" The Monticello pressure-temperature (P-T) curves referenced in Monticello Technical Specifications (TS) 3.6/4.6.B (with the associated Reference Temperature Nil Ductility Transition (RT_{NDT}) Shift) are used to develop the temperature during the system leakage test to prevent brittle fracture. This results in a temperature of <212°F.

IWB-5240 states:
"(a) The minimum test temperature for either the system leakage or system hydrostatic test shall not be lower then the minimum temperature for the associated pressure specified in the plant Technical Specifications."

"(b) The system test temperature shall be modified as required by the results obtained from each set of material surveillance specimens withdrawn from the reactor vessel during the service lifetime"

"(c) For tests of systems or portions of systems constructed entirely of austenitic steel, test temperature limitations are not required to meet fracture prevention criteria. In cases where the components of the system are constructed of ferritic and austenitic steels that are nonisolable from each other during a system leakage or system hydrostatic test, the test temperature shall be in accordance with IWB-5230(a)".

Under IWB-5230(a) it states that the requirements of IWB-5240 need to be met for all ferritic steel components within the boundary of the system (or portion of system) subject to the test pressure.

The Monticello TS state in section 3.6.B.1 *"During inservice hydrostatic or leak testing, the reactor vessel shell temperatures specified in 4.6.B.1, except for the reactor vessel bottom head, shall be at or above the temperatures shown on the two curves of Figure 3.6.2, where the dashed curve, 'RPV Core Beltline,' is increased by the core beltline*

temperature adjustment from Figure 3.6.1. The reactor vessel bottom head temperature shall be at or above the temperatures shown on the solid curve of Figure 3.6.2, 'RPV Remote from Core Beltline,' with no adjustment from Figure 3.6.1."

Figure 3.6.1 is the Core Beltline Operating Limits Curve Adjustment vs. Fluence. The X axis is the ¼ Vessel Wall Thickness Fluence (10^{18} n/cm^2) with the Y axis the RT_{NDT} Shift (deg F). This is calculated every cycle to determine the amount of shift to apply for each outage.

Figure 3.6.2 is the Minimum Temperature vs. Pressure for Pressure Tests. The X axis is the Pressure Limit in Vessel Top Head (psig) with the Y axis the Minimum Vessel Metal Temperature (°F). Once the adjustment is calculated from Figure 3.6.1, that adjustment is added to the temperature for the corresponding pressure for the applicable pressure test. Once every refueling outage, ASME, Section XI requires a system leakage pressure test on the Class 1 system per Category B-P. The corresponding pressure for Monticello is 1010 psig. The current estimate for the fluence at the end of the current cycle is 2.33×10^{18} n/cm^2. This equates to an adjustment of 113.5°F. When this is added to the corresponding temperature for 1010 psig the resultant temperature required per Monticello's TS for the system pressure test is 178.5°F (including instrument uncertainty), which is less than the normal operating temperature of 540°F.

The discussion above confirms that Monticello follows the ASME Code, Section XI, IWA-5000, IWB-5240, temperature requirements when performing periodic Reactor Pressure Vessel Leakage and Hydrostatic pressure test.

NRC Request #2:

Please provide the numbers and results of your inspections of the control rod drive mechanism (CRDM) bolting. How many of new types bolts as required by GE SIL-483 were installed in the CRDMs?

NMC Response:

At Monticello there are a total of 121 control rod drives (CRDs) with 8 cap screws for each CRD housing. In 1991 a total of 72 cap screws were inspected with 37 (51%) being rejected. During that same outage 484 cap screws (4 on each CRD) were replaced using the same design as the existing cap screws. These replacements were made because of information provided by the supplier General Electric (GE) that indicated that 3 uniformly spaced cap screws were sufficient to support a CRD.

In 1995, GE introduced a new cap screw design of a different material and configuration. In 1996, a total of 112 cap screws were inspected with 18 (16%) rejected, 14 (13%) of these were rejected for linear indications. In 2000, a total of 88 cap screws were inspected with 24 (27%) being rejected. Currently, Monticello has installed 200 cap screws of the new GE design to meet the recommendation of GE SIL-483. Monticello intends to use the new cap screw design or an improved cap screw design for future replacements.

Additionally, Monticello will only examine those cap screws that are intended for reuse, per the requirements of Table IWB-2500-1, Item B7.80, as required by 10 CFR 50.55a(b)(2)(xxi)(B).

NRC Request #3:

Please provide the material of the bolts and environmental conditions that these bolts are subject to during operation? Also, what is the degradation mechanism of these bolts?

NMC Response:

The material for the original designed cap screws is American Society for Testing and Materials (ASTM) SA-193 Grade B7 material while the new designed cap screws are ASTM SA-540 Grade B23, Class 4, which has higher mechanical properties and more controlled chemistry. The normal environmental conditions that the cap screws are subjected to is containment atmosphere <200°F based on temperature records of center of drives being <200°F.

The degradation mechanism for the original designed cap screws is attributed to stress corrosion cracking in a creviced region of the cap screw with possible aggravation by fabrication irregularities. Magnesium sulfide inclusions and surface pitting may have contributed in some cases. It was also probable that water leakage into the bolted connection contributed to stress corrosion. The new style cap screws are intended to be less susceptible to these degradation mechanisms.

NRC Request #4:

How does NMC determine that the leakage from the CRDMs has stopped when the operating pressure and temperature is reached?

NMC Response:

There is no way to absolutely assure that leakage from the CRDs has stopped when the operating pressure and temperature has been reached. However, GE provided guidance for CRD flange leakage evaluations which states that drip-type leaks of 30 drops per minute or less which show a constant or decreasing leak rate over an 8-hour period at reactor pressures greater than or equal to 1000 psig, do not require any corrective maintenance action. Corrective maintenance is not required because a decreasing leak rate will eventually seal without being internally pressurized, provided the flange bolts remain properly torqued.

In addition, the CRD bolting is monitored during the hydrostatic pressure test, performed during startup. And when the unit is operating at rated temperature and pressure the drywell drain sump monitoring system provides indication of leakage. The Monticello TS provide action levels for unidentified leakage in the drywell. During a normal operating cycle Monticello unidentified leakage is typically less than 5% of the TS limit of 5 gpm. Additionally, there is also a TS limit of a 2 gpm increase in unidentified leakage in a 24-hour period.