NMC
Committed to Nuclear Excellence

Monticello Nuclear Generating Plant
Operated by Nuclear Management Company, LLC

September 19, 2002

10 CFR Part 50
Section 50.90

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

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

License Amendment Request for Relocation of a Technical Specification Surveillance Requirement to a Licensee Controlled Program and Implementation of the Boiling Water Reactor Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP)

Reference:1. Letter from W. H. Bateman (USNRC) to C. Terry (BWRVIP Chairman) titled, "Safety Evaluation Regarding EPRI Proprietary Report 'BWR Vessel and Internals Project, BWR Integrated Surveillance Program Plan (BWRVIP-78)' and 'BWRVIP-86: BWR Vessel and Internals Project, BWR Integrated Surveillance Program Implementation Plan,'" dated February 1, 2002.

2. Regulatory Issue Summary No. 2002-05, "NRC Approval of Boiling Water Reactor Pressure Vessel Integrated Surveillance Program," dated April 8, 2002.

Pursuant to and in accordance with the requirements of 10 CFR 50.90, 50.59, and Appendix H, Nuclear Management Company, LLC (NMC) hereby requests a change to the Operating License, DPR-22, for the Monticello Nuclear Generating Plant. The proposed changes to Appendix A of the Operating License will relocate a Technical Specification (TS) Surveillance Requirement to the Monticello Nuclear Generating Plant Updated Safety Analysis Report (Monticello USAR). This proposed change relocates Surveillance Requirement 4.6.B.2 to the Monticello USAR and revises the Reactor Pressure Vessel Material Surveillance Program in accordance with References 1 and 2.

Exhibit A contains the Proposed Change, Reasons for Change, a Technical Analysis, a Determination of No Significant Hazards Consideration and an Environmental Assessment. Exhibit B contains the current Monticello TS Page and TS Bases Page marked up to show the proposed change. Exhibit C contains the revised Monticello TS Page and TS Bases Page. Exhibit D contains the current Monticello USAR page marked up to show the proposed change. Exhibit E contains the revised Monticello USAR page.

The Monticello Nuclear Generating Plant Operations Committee and the Off-Site Review Committee have reviewed the proposed change. A copy of this submittal, along with the

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evaluation of No Significant Hazards Consideration, is being forwarded to our appointed state official pursuant to the requirements of 10 CFR 50.91.

Consistent with the process established between the NRC and the BWRVIP, this change is being processed as a license amendment to facilitate NRC review and approval.

NMC plans to implement the proposed changes at Monticello in the Spring of 2003 to support deletion of the removal of a test specimen from the 21st Refueling and Inspection Outage. Therefore, we request NRC complete its review of this change by December 31, 2002 with the changes effective 60 days after approval.

Any questions regarding this request should be directed to John Fields, Sr. Licensing Engineer, at (763) 295-1663.

Jeffrey S. Forbes
Site Vice President
Monticello Nuclear Generating Plant

Subscribed and sworn before me this 19th day of September, 2002

STEPHEN R. BLEGEN
NOTARY PUBLIC-MINNESOTA
My Comm. Exp. Jan. 31, 2005

Notary

Attachments:

Attachments:	Exhibit A-	Proposed Change, Reason for Change, Technical Analysis, Determination of No significant Hazards Consideration, and Environmental Assessment
	Exhibit B-	Current Monticello TS Page and TS Bases Page Marked Up With the Proposed Change
	Exhibit C-	Revised Monticello TS Page and TS Bases Page
	Exhibit D-	Current Monticello Updated Safety Analysis Report Page Marked Up with the Proposed Change
	Exhibit E-	Revised Monticello Updated Safety Analysis Report Page

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

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Proposed Change, Reason for Change, Technical Analysis, Determination of No Significant Hazards Considerations and Environmental Assessment For Revision to the Reactor Pressure Vessel Material Surveillance Program

License Amendment Request for Relocation of a Technical Specification Surveillance Requirement to a Licensee Controlled Program and Implementation of the Boiling Water Reactor Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP)

The following provides the basis for the proposed revision to the reactor pressure vessel material surveillance program.

1.0 DESCRIPTION OF THE PROPOSED CHANGE

Nuclear Management Company, LLC (NMC) proposes to revise Appendix A, Technical Specifications, to Operating Licensing DPR-22 for the Monticello Nuclear Generating Plant. The change would relocate the current Monticello Technical Specifications (TS) Surveillance Requirement 4.6.B.2 from the TS to the Monticello Updated Safety Analysis Report (USAR). The revised Monticello USAR will implement the Boiling Water Reactor Vessel and Internals Program (BWRVIP) Integrated Surveillance Program (ISP), approved by the NRC in its Safety Evaluation (SE) dated February 1, 2002 (Reference 1). The proposed revision to the Monticello Technical Specifications (TS) Section 4.6.B.2 and the TS Bases reflecting these changes are provided in Exhibit C. The revised Monticello USAR page reflecting this change is provided in Exhibit E.

2.0 REASON FOR THE PROPOSED CHANGE

The BWRVIP ISP was developed in response to an issue raised by the NRC staff regarding the potential lack of adequate unirradiated baseline Charpy V-notch (CVN) data for one or more materials in plant-specific RPV surveillance programs at several Boiling Water Reactors (BWRs). The lack of baseline properties would inhibit a licensee's ability to effectively monitor changes in the fracture toughness properties of RPV materials in accordance with Appendix H to 10 CFR 50. The BWRVIP ISP, as approved by the NRC, resolves this issue. The relocation of TS Surveillance Requirement 4.6.B.2, from the TS to the USAR, is needed to more accurately reflect changes to the BWRVIP ISP. This will provide additional flexibility for future changes to the ISP process.

Implementation of the ISP will provide additional benefits. When the original surveillance materials were selected for plant-specific surveillance programs, the state of knowledge concerning Reactor Pressure Vessel (RPV) material response to irradiation and post-irradiation fracture toughness was not the same as it is today. As a result, many facilities did not include what would be identified today as the plant's limiting RPV materials in their surveillance programs. Hence, this effort to identify and evaluate materials from other BWRs, which may better represent a facility's limiting materials, should improve the overall evaluation of BWR RPV embrittlement. Second, the inclusion of data from the testing of BWR Owners' Group (BWROG) Supplemental Surveillance Program capsules will improve overall quality of the data being used to

evaluate BWR RPV embrittlement. Finally, implementation of the ISP is also expected to reduce the cost of surveillance testing and analysis since surveillance materials that are of little or no value (either because they lack adequate unirradiated baseline CVN data or because they are not the best representative materials) will no longer be tested.

3.0 TECHNICAL ANALYSIS

The deletion of TS Surveillance Requirement 4.6.B.2 is acceptable because it does not meet the minimum requirements of 10 CFR 50.36(c)(3) for inclusion in the TSs. This Surveillance Requirement may be deleted because the installation of test specimens in the reactor vessel and their associated material sample program monitor fluence embrittlement for long term operation of the reactor vessel and for establishing pressure-temperature curve limitations. These measurements do not assure that the necessary quality of systems and components are maintained, or that facility operation will be within safety limits, nor do they assure that a limiting condition for operation will be met. Therefore, the surveillance is not a required Technical Specification Surveillance Requirement per the provisions of 10 CFR 50.36(c)(3).

Reference 1 concludes that the proposed ISP, if implemented in accordance with the conditions in the NRC Safety Evaluation, has been determined to be an acceptable alternative to all existing BWR plant-specific RPV surveillance programs for the purpose of maintaining compliance with the requirements of Appendix H to 10 CFR Part 50 through the end of current facility 40 year operating licenses. Reference 1 requires that each licensee (1) provide information regarding what specific neutron fluence methodology will be implemented as part of participation in the ISP and (2) address the neutron fluence methodology compatibility issue as it applies to the comparison of neutron fluences calculated for its RPV versus the neutron fluences calculated for surveillance capsules in the ISP which are designated to represent its RPV. This information is provided in the following discussion:

The Monticello Technical Specifications, as discussed in Amendment No. 106 to the Monticello Operating License (DPR-22), required that new Pressure and Temperature curves be implemented based on updated fluence calculations. See Reference 2 for additional information.

NMC intends to use the BWRVIP RAMA code or other already approved NRC methodology to revise the calculations for Monticello. The RAMA code will perform a full 3D-neutron transport solution to determine fluence within the vessel. The analysis will use the BUGLE-96 data library as recommended by Regulatory Guide 1.190. It will perform a full uncertainty analysis to determine the accuracy of the calculation.

The current schedule for completion of the BWRVIP RAMA code is December 2002. The BWRVIP intends to submit a topical report on the RAMA code to the NRC for review, with the objective of receiving a safety evaluation in 2003 approving use of the methodology.

The first surveillance capsule to be tested under the ISP is the River Bend 183°capsule. The test report for the River Bend capsule is currently scheduled to be submitted to the NRC by February 2003. Coincidentally, these capsules, according to the ISP, are the substitute capsules for Monticello weld material. Thus in accordance with the ISP, the Monticello capsule will not be removed and tested during the 2003 refueling outage.

The BWRVIP-86 ISP Capsule Test Schedule is an EPRI Proprietary document that currently requires the next Monticello surveillance capsule be removed between 2005 and 2007 and tested within one year of removal. The Monticello fluence calculations will be reevaluated based on the test report from River Bend and after completion of the Monticello capsule test.

4.0 DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

The Commission has provided standards in 10 CFR 50.92(c) for determining whether a significant hazards consideration exists. A proposed amendment to an operating license for a facility involves a determination of no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

Nuclear Management Company, LLC (NMC) proposes to relocate the requirements to install and sample reactor vessel, base weld, and heat affected zone test specimens in the reactor vessel, to the Monticello Updated Safety Analysis Report (USAR) and implement the Boiling Water Reactor Vessel and Internals Program (BWRVIP) Integrated Surveillance Program (ISP). This change is acceptable because the relocation of this Surveillance Requirement to a Licensee controlled document provides an equivalent method of implementing the BWRVIP ISP, which was approved by the NRC staff as meeting the requirements of paragraph III.C of Appendix H to 10 CFR 50 for an integrated surveillance program. Additionally, the deletion of TS Surveillance Requirement 4.6.B.2 is also acceptable because it does not meet the requirements of 10 CFR 50.36(c)(3) for inclusion in the TS.

In accordance with the criteria set forth in 10 CFR 50.92, NMC has evaluated the proposed TS change for Monticello and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. ***The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.***

   The proposed change relocates the requirement of the TS Surveillance Requirement to a Licensee controlled document and implements an integrated surveillance program that has been evaluated by the NRC staff as meeting the requirements of paragraph III.C of Appendix H to 10 CFR 50. The proposed change of relocating a TS Surveillance Requirement to the Monticello USAR and implementing an integrated surveillance program is not considered a precursor or initiator of an accident previously evaluated. The proposed change does not impact current plant operations or the design function of any structure, system or component. Consequently, the proposed change does not significantly increase the probability of any accident previously evaluated.

   The proposed change provides the same assurance of Reactor Pressure Vessel integrity as has always been assured. The relocation of the TS Surveillance Requirement provides an acceptable method for implementing the integrated surveillance program which was evaluated by the NRC staff as meeting the requirements of 10 CFR 50, Appendix H, paragraph III.C. The relocation of the TS Surveillance or the implementation of an integrated surveillance program is not an input or consideration in any accident previously evaluated, thus the proposed change will not increase the probability of any such accident occurring. The proposed amendment does not involve any change to the configuration or method of operation of any plant equipment that is used to mitigate the consequences of an accident, nor does it affect any assumptions or conditions in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly increased.

   Therefore, operation of the facility in accordance with the proposed change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. ***The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated?***

   The proposed amendment does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. No equipment interfaces are modified and no changes to any equipment function or the method of operating the equipment are being made. The proposed change, to relocate the TS Surveillance and implement an integrated surveillance program, maintains an equivalent level of RPV material surveillance and does not introduce any new accident initiators. The proposed change will not change the design, configuration or operation of the plant.

   Therefore, operation of the facility in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. ***The proposed change will not involve a significant reduction in a margin of safety?***

   The proposed amendment has been evaluated as providing an acceptable alternative to the plant-specific RPV material surveillance program that meets the requirements of the regulations for RPV material surveillance. The proposed change does not exceed or alter a design basis or safety limit. The change relocates a TS Surveillance Requirement and implements an integrated surveillance program and as such does not significantly reduce the margin of safety.

   Therefore, operation of the facility in accordance with the proposed change does not involve a significant reduction in a margin of safety.

5.0 ENVIRONMENTAL CONSIDERATION

10 CFR 51.22(c)(9) identifies certain licensing and regulatory actions, which are eligible for categorical exclusion from the requirement to perform an environmental assessment. A proposed amendment to an operating license for a facility does not require an environmental assessment if operation of the facility in accordance with the proposed amendment would not (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite; or (3) result in a significant increase in individual or cumulative occupational radiation exposure. NMC has evaluated the proposed change for Monticello and has determined that the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Accordingly, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with issuance of the amendment. The basis for this determination, using the above criteria, follows:

Basis

1. As demonstrated in the Determination of No Significant Hazards Consideration, the proposed amendment does not involve a significant hazards consideration.

2. There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite. The proposed change does not involve any physical alteration of the plant (no new or different type of equipment will be installed) or change in methods governing normal plant operation.

3. There is no significant increase in individual or cumulative occupational radiation exposure. The proposed change does not involve any physical alteration of the plant (no new or different type of equipment will be installed) or change in methods governing normal plant operation.

6.0 REFERENCES:

1. Letter from W. H. Bateman (USNRC) to C. Terry (BWRVIP Chairman) titled, "Safety Evaluation Regarding EPRI Proprietary Report 'BWR Vessel and Internals Project, BWR Integrated Surveillance Program Plan (BWRVIP-78)' and 'BWRVIP-86: BWR Vessel and Internals Project, BWR Integrated Surveillance Program Implementation Plan,'" dated February 1, 2002.

2. Letter from C. F. Lyon (USNRC) to R. O. Anderson (NSP) titled, "Monticello Nuclear Generating Plant – Issuance of Amendment RE: Revision of Reactor Pressure Vessel Pressure-Temperature Limit Curves and Removal of Standby Liquid Control Relief Valve Setpoint (TAC No. MA4532)," dated October 12, 1999.

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Current Monticello TS Page and TS Bases Page Marked-Up

With Proposed Change

This Exhibit consists of current TS page and TS Bases page marked up with the proposed change. The pages included in this Exhibit are as listed below:

Pages:

122
146

B. Reactor Vessel Temperature and Pressure

1. During in-service 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.

~~2.~~ During heatup by non-nuclear means (except with the reactor vessel vented), cooldown following nuclear shutdown, or low level physics tests the reactor vessel shell and fluid temperatures specified in 4.6.A shall be at or above the higher of the temperatures of Figure 3.6.3 where the dashed curve, "RPV Core Beltline," is increased by the expected shift in RT_{NDT} from Figure 3.6.1.

3. During all operation with a critical reactor, other than for low level physics tests or at times when the reactor vessel is vented, the reactor vessel shell and fluid temperatures specified in 4.6.A shall be at or above the higher of the temperatures of Figure 3.6.4 where the dashed curve, "RPV Core Beltline," is increased by the expected shift in RT_{NDT} from Figure 3.6.1.

B. Reactor Vessel Temperature and Pressure

1. During in-service hydrostatic or leak testing when the vessel pressure is above 312 psig, the following temperatures shall be recorded at least every 15 minutes.

   a. Reactor vessel shell adjacent to shell flange.
   b. Reactor vessel bottom head.
   c. Reactor vessel shell or coolant temperature representative of the minimum temperature of the beltline region.

~~2. Test specimens representing the reactor vessel, base weld, and weld heat affected zone metal shall be installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The material sample program shall conform to ASTM E 185-66.~~

Bases 3.6/4.6 (Continued) :

The fracture toughness of all ferritic steels gradually and uniformly decreases with exposure to fast neutrons above a threshold value, and it is prudent and conservative to account for this in the operation of the reactor pressure vessel. Two types of information are needed in this analysis: 1) A relationship between the changes in fracture toughness of the reactor pressure vessel steel and the neutron fluence (integrated neutron flux), and 2) A measure of the neutron fluence at the point of interest in the reactor pressure vessel wall.

The relationship of predicted adjustment of reference temperature versus fluence and the copper and nickel content of the core beltline materials given in Regulatory Guide 1.99, Revision 2, was originally used to define the core beltline temperature adjustment versus fluence shown on Figure 3.6.1.

A relationship between full power years of operation and neutron fluence has been experimentally determined for the reactor vessel. The vessel pressurization temperatures at any time period can be determined from the thermal energy output of the plant and Figure 3.6.1 used in conjunction with Figure 3.6.2 (pressure tests), Figure 3.6.3 (mechanical heatup or cooldown with a noncritical core), or Figure 3.6.4 (operation with a critical core). During the first fuel cycle, only calculated neutron fluence values were used. At the first refueling, neutron dosimeter wires which were installed adjacent to the vessel wall were removed to experimentally determine the neutron fluence versus full power years of operation. This experimental result was updated by testing additional dosimetry removed with the first surveillance capsule.

Reactor vessel material samples are provided, however, to verify the relationship expressed by Figure 3.6.1. Three sets of mechanical test specimens representing the base metal, weld metal, and weld heat affected zone (HAZ) metal have been placed in the vessel and can be removed and tested as required. Two sets of specimens were contained in the first surveillance capsule which was removed from the vessel in 1981. One set of specimens was tested at this time. The second set was later inserted into a new capsule, and installed in the Prairie Island Nuclear Generating Plant RPV for accelerated irradiation. This capsule was removed and tested in 1996. NSP performed calculations per the requirements of Regulatory Guide 1.99, Rev. 2, Position 2.1 to develop new pressure/temperature (P-T) curves. Results of Charpy V-notch impact tests for the two sets of data and from 1997 non-irradiated material test data were used in developing the revised Figures 3.6.1, 3.6.2, 3.6.3, and 3.6.4. ~~An analysis and report will be submitted to the Commission on all such surveillance specimens removed from the reactor vessel in accordance with 10 CFR 50, Appendix H, including information obtained on the level of integrated fast neutron irradiation received by the specimens and actual vessel material.~~

License Amendment Request for Relocation of a Technical Specification Surveillance Requirement to a Licensee Controlled Program and Implementation of the Boiling Water Reactor Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP)

Revised Monticello TS Page and TS Bases Page

This Exhibit consists of revised TS page and TS Bases page that incorporate the proposed change. The pages included in this Exhibit are as listed below:

Pages:

122
146

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
B. Reactor Vessel Temperature and Pressure	B. Reactor Vessel Temperature and Pressure
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2. During heatup by non-nuclear means (except with the reactor vessel vented), cooldown following nuclear shutdown, or low level physics tests the reactor vessel shell and fluid temperatures specified in 4.6.A shall be at or above the higher of the temperatures of Figure 3.6.3 where the dashed curve, "RPV Core Beltline," is increased by the expected shift in RT_{NDT} from Figure 3.6.1.	
3. During all operation with a critical reactor, other than for low level physics tests or at times when the reactor vessel is vented, the reactor vessel shell and fluid temperatures specified in 4.6.A shall be at or above the higher of the temperatures of Figure 3.6.4 where the dashed curve, "RPV Core Beltline," is increased by the expected shift in RT_{NDT} from Figure 3.6.1.	

Bases 3.6/4.6 (Continued):

The fracture toughness of all ferritic steels gradually and uniformly decreases with exposure to fast neutrons above a threshold value, and it is prudent and conservative to account for this in the operation of the reactor pressure vessel. Two types of information are needed in this analysis: 1) A relationship between the changes in fracture toughness of the reactor pressure vessel steel and the neutron fluence (integrated neutron flux), and 2) A measure of the neutron fluence at the point of interest in the reactor pressure vessel wall.

The relationship of predicted adjustment of reference temperature versus fluence and the copper and nickel content of the core beltline materials given in Regulatory Guide 1.99, Revision 2, was originally used to define the core beltline temperature adjustment versus fluence shown on Figure 3.6.1.

A relationship between full power years of operation and neutron fluence has been experimentally determined for the reactor vessel. The vessel pressurization temperatures at any time period can be determined from the thermal energy output of the plant and Figure 3.6.1 used in conjunction with Figure 3.6.2 (pressure tests), Figure 3.6.3 (mechanical heatup or cooldown with a noncritical core), or Figure 3.6.4 (operation with a critical core). During the first fuel cycle, only calculated neutron fluence values were used. At the first refueling, neutron dosimeter wires which were installed adjacent to the vessel wall were removed to experimentally determine the neutron fluence versus full power years of operation. This experimental result was updated by testing additional dosimetry removed with the first surveillance capsule.

Reactor vessel material samples are provided, however, to verify the relationship expressed by Figure 3.6.1. Three sets of mechanical test specimens representing the base metal, weld metal, and weld heat affected zone (HAZ) metal have been placed in the vessel and can be removed and tested as required. Two sets of specimens were contained in the first surveillance capsule which was removed from the vessel in 1981. One set of specimens was tested at this time. The second set was later inserted into a new capsule, and installed in the Prairie Island Nuclear Generating Plant RPV for accelerated irradiation. This capsule was removed and tested in 1996. NSP performed calculations per the requirements of Regulatory Guide 1.99, Rev. 2, Position 2.1 to develop new pressure/temperature (P-T) curves. Results of Charpy V-notch impact tests for the two sets of data and from 1997 non-irradiated material test data were used in developing the revised Figures 3.6.1, 3.6.2, 3.6.3, and 3.6.4.