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Monticello Nuclear Generating Plant
Operated by Nuclear Management Company, LLC

December 22, 2004

L-MT-04-068
10 CFR 50.71(e)
10 CFR 50.59(d)(2)

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

Monticello Nuclear Generating Plant
Docket 50-263
License No. DPR-22

Submittal of Revision 21 to the Updated Safety Analysis Report

Pursuant to 10 CFR Part 50, Section 50.71(e), Revision 21 to the Updated Safety Analysis Report (USAR) for the Monticello Nuclear Generating Plant is hereby submitted. This revision provides an update of the information in the USAR for the period from May 31, 2003 through September 30, 2004.

The changes in this revision reflect the incorporation of design changes, 10 CFR 50.59 Evaluations, License Amendments, and some editorial corrections and clarifications. These changes are made in accordance with the guidance provided in Nuclear Energy Institute (NEI) 98-03, "Guidelines for Updated Final Safety Analysis Reports," Revision 1, and Regulatory Guide 1.181, "Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e)."

Included as part of this submittal is the periodic report of changes, tests and experiments required by 10 CFR Part 50, Section 50.59(d)(2). The summary report of changes, tests and experiments requiring evaluation under the provisions of 10 CFR 50.59 is provided as Enclosure 1.

Enclosure 2, "Report of Changes to Licensee Docketed Commitments," provides a brief description and summary of changes to NRC commitments identified to be reported to the Commission in accordance with guidance provided in NEI 99-04, "Guidelines for Managing NRC Commitment Changes."

Enclosure 3, "Report of Information Removed from the USAR," provides a summary of information removed from the USAR in this revision cycle. This information is provided in accordance with NEI 98-03, "Guidelines for Updating Final Safety Analysis Reports," Revision 1.

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Enclosure 4 contains Revision 21 of the Monticello USAR and instructions for posting the document. Enclosure 4 is being submitted on a page replacement basis in accordance with 10 CFR 50.71(e)(3)(i).

Enclosure 5, "Report of Changes to Monticello Fire Protection Program," provides a summary of changes to the Monticello Fire Protection Program. Changes to the Fire Protection Program are provided in accordance with 10 CFR 50.71(e), 10 CFR 50.59 and the guidance in Generic Letter 86-10.

This letter makes no new commitments or changes to any existing commitments.

I hereby certify that I am a duly authorized officer of Nuclear Management Company, LLC, and that to the best of my knowledge, information, and belief, the information provided in the attached Revision 21 to the Monticello USAR meets the requirements of 10 CFR 50.71(e) to update the USAR through September 30, 2004.

John T. Conway for TJP

Thomas J. Palmisano
Site Vice President, Monticello Nuclear Generating Plant
Nuclear Management Company, LLC

Enclosures (5)

cc: Administrator, Region III, USNRC
Project Manager, Monticello, USNRC
Resident Inspector, Monticello, USNRC

ENCLOSURE 1

MONTICELLO NUCLEAR GENERATING PLANT
REPORT OF CHANGES, TESTS AND EXPERIMENTS

The following includes a brief description and summary of the 10CFR50.59 evaluations for those changes, tests and experiments that were carried out without prior NRC approval, pursuant to the requirements of 10 CFR Part 50, Section 50.59(d)(2).

1. Installation of RPS Test Fixture to Preclude Half SCRAMS During Surveillance Testing (50.59 Evaluation 02Q065)

Activity Description: Banana jack adaptors were added to the reactor protection system (RPS) relays to facilitate testing. Use of an associated test fixture allowed testing by observing the status of the trip channel instead of the trip logic and associated half scram. An evaluation was performed because of concerns over the possible adverse effects associated with inadvertently leaving a test fixture installed after a test, and the possible decrease in scram contactor reliability caused by the reduced number of cycles on the scram contactors due to use of the test fixture.

Summary: General Electric Topic Report NEDC-30851P, Technical Specification Improvement Analysis for BWR Reactor Protection System," identified common cause failure in the scram contactors as the major cause of RPS unavailability. Testing the scram contactors at seven day intervals was found to be optimal for detecting common cause failures. Changing the test method as described in the USAR from verifying trip logic trips to verifying trip channel trips reduced the number of cycles on the scram contactors. Therefore, weekly testing of the scram contactors using the subchannel test handswitches was implemented to ensure that RPS reliability was not decreased.

Administrative controls and independent verification prevent the test fixture from inadvertently being left installed after the test. This is the same level of control as that used in other procedures that could affect the likelihood of malfunction of the trip logic.

Conclusion: Any increase in the likelihood of malfunction of the RPS was less than minimal. Prior NRC approval per 10 CFR 50.59 was not required.

2. Procedure Change to Limit Time RHR Division I Torus Cooling Return Line is Used for Pipe Wall Thinning Downstream of MO-2008 (50.59 Evaluation 03-006)

Activity Description: Procedure changes were made to limit the time the Residual Heat Removal (RHR) Division I torus cooling return line is used during normal and post accident operation. The use of RHR Division I torus cooling is being minimized due to thinning of the pipe wall downstream of MO-2008. Procedures were changed to identify the associated operational limitations. These procedural limitations will be in place until the degraded piping down stream of MO-2008 is replaced, currently scheduled for the 2005 refueling outage

Summary: The procedure changes will assure that the piping section downstream of MO-2008 will continue to meet the Code operability limits until the pipe section is replaced. The limitations imposed are consistent with RHR suppression pool cooling completing its required mission time. The procedure changes will not affect the ability of primary containment to perform its design functions. The procedure change to limit the time the RHR Division I torus cooling return line is used during normal operation will not affect the design function or limit required testing of the RHR system. The procedure change to limit the time the RHR Division I torus cooling return line is used during post accident operation will not affect the design function or mission time for that function. Therefore, since the design function and perceived mission time of suppression pool cooling are unaffected, there was no adverse affect on the containment safety analysis.

Conclusion: This activity did not require prior NRC approval per 10 CFR 50.59, and no Technical Specification change is involved. The procedure change to limit the time the RHR Division I torus cooling return line is used during normal and post accident operation was implemented without prior NRC approval per 10 CFR 50.59.

3. **Relationship Between Gardel and Safety Limit Minimum Critical Power Ratio (50.59 Evaluation 04-002)**

Activity Description: The Studsvik/Scandpower Gardel computer program will be used as the reactor core thermal limits monitor as it relates to the Safety Limit Minimum Critical Power Ratio (SLMCPR).

Summary: The current licensing basis for the SLMCPR is defined in Global Nuclear Fuels report NEDC-32601 using uncertainties defined in the GETAB methodology. There were no changes to either of these items as a result of the use of Gardel as the core monitor.

Conclusion: Neither a change to the Technical Specifications nor a change to the USAR was required. Prior NRC approval per 10 CFR 50.59 was not required.

4. **Procedural Change for Switching Evolutions for Fire Pumps Resulting in Bypassing the Pumps Automatic Initiation Logic (50.59 Evaluation 04-003)**

Activity Description: The activity involves a procedure change to allow a dedicated operator, in contact with the control room, to momentarily switch a fire pump out of its normal automatic initiation mode. This switching activity occurs when the screen wash/fire pump control switch is moved through the STOP position when switching between AUTO and START or when the diesel fire pump is manually started. During these occasions, the automatic start feature of the initiation logic is momentarily bypassed. This is contrary to how the function is described in USAR Section 10.3.

ENCLOSURE 1

The revised procedure included the following precautions and prerequisites:

- The Fire Protection System is in standby readiness during all phases of plant operation.
- A Fire Impairment is not required to be entered as long as Operations Personnel are in attendance at the affected fire pump panel when the automatic initiation logic is momentarily bypassed for pump starts or mode changes (e.g., hand switch moved from AUTO through STOP to START).
- The attending Operator is in radio or telephone contact with the Control Room when the automatic initiation logic is bypassed.
- If fire suppression water is required, then the affected fire pump will be returned to service, unless otherwise directed by Shift Management.
- A seven-day impairment per Fire Protection Program procedures shall be entered if a fire pump automatic initiation logic is bypassed without an operator in attendance.

<u>Summary</u>: USAR Section 10.3 describes the automatic actions/initiation of the three fire pumps when fire header pressure decreases to various fire header pressure setpoints. During switching for manual starting of the screen wash/fire pump and the diesel fire pump, operator action momentarily bypasses this automatic initiation feature. When these manual switching operations occur, an operator is available locally to restore the pump should the need arise, as allowed by the performance requirements of the USAR and fire protection program procedures.

Only two of the three normally available fire pumps are required to meet design fire header flow. If there are not two pumps available, both the USAR and the fire protection plan/procedures allow the use of qualified dedicated operators for fire watches.

<u>Conclusion</u>: The procedure change had no impact on the availability or reliability of the fire protection system pumps and introduces no additional risk of any accident or transient, provided the precautions and prerequisites detailed above are met. Prior NRC approval per 10 CFR 50.59 was not required.

ENCLOSURE 2

MONTICELLO NUCLEAR GENERATING PLANT
REPORT OF CHANGES TO LICENSEE DOCKETED COMMITMENTS

The purpose of this exhibit is to provide a brief description and a summary of changes to formally tracked commitments established with the NRC by Nuclear Management Company, LLC (NMC). These commitments are being identified and reported to the Commission in accordance with guidance provided in Nuclear Energy Institute (NEI) Technical Report 99-04 Revision 0, "Guidelines for Managing NRC Commitment Changes."

1. Monticello Commitment M70001A

   Source Document: Significant Operating Event 70-004, "Refueling Interlocks regarding core Alterations," dated September 23, 1970.

   Commitment: The Operations Jumper Log Book entry must, record the Sections of the Tech Specs applicable and reviewed. If Tech Specs are not involved record it as "Tech Specs not applicable."

   Change: Shift Supervision shall verify that the plant is in a suitable condition for the bypass installation and that the installation will not result in a Technical Specification violation or impact pending testing or operations.

2. Monticello Commitment M75019A

   Source Document: Response to NRC document, "Personnel Hand Exposure," dated August 15, 1975.

   Commitment: When it is determined that the extremity exposure will be >5 times the whole body exposure, the working time limits will be based on the extremity exposure.

   Change: This commitment has been deleted. Station procedures and processes provide sufficient barriers to prevent an over-exposure of an extremity.

ENCLOSURE 3

MONTICELLO NUCLEAR GENERATING PLANT
SUMMARY OF INFORMATION REMOVED FROM THE USAR

Consistent with the guidance in Nuclear Energy Institute (NEI) Technical Report 98-03, "Guidelines for Updating Final Safety Analysis Reports," Revision 1, and Regulatory Guide 1.181, information removed from the Monticello USAR is summarized below.

- In section 6.2.3.2.4.e, second paragraph, a sentence was removed that stated that the Residual Heat Removal system piping is carbon steel. The piping is predominantly carbon steel. But a portion of the system was found to be susceptible to erosion corrosion. The affected carbon steel piping will be replaced by an erosion resistant material during the next refueling outage. The description continues to state that the system is designed in accordance with applicable codes for reactor auxiliary systems. (USAR Change 04-213)

- In section 7.2.1.2.3.b, in the description of the reactor manual control system control rod select pushbuttons, the reference to the pushbuttons as "DPST" [double pole single throw] was removed. This corrects an error in the description. The contacts are actually double pole double throw (DPDT). Describing the contacts in this manner is considered unnecessarily detailed for a USAR description. The description now states that pushbuttons activate two sets of contacts. (USAR Change 04-203)

- In section 7.3.5.2.2 (Average Power Rate Monitor description), second paragraph, the reference to section 2.3.A.1 of the Technical Specifications was deleted. This section of the Technical Specifications was removed in License Amendment 128. This change corrects an error made during implementation of License Amendment 128, when this reference to the Technical Specifications in the USAR was missed. (USAR Change 04-908)

- In section 7.6.3.2.4.10, second paragraph, a sentence was deleted describing the arrangement of setpoints in the High Pressure Coolant Injection turbine high steam flow trip logic. This change reflects a modification approved by NRC (License Amendment 117). This sentence should have been deleted in USAR Revision 19, but was overlooked. Making this change in Revision 21 corrects an error made during a previous USAR revision. (USAR Change 04-216)

- At the end of section 8.9.1, a list of the types of documents contained in the Equipment Qualification (EQ) central file included, "Component Evaluation Worksheets." This item was deleted from the list. These worksheets have been discontinued and the information incorporated into each of the EQ Calculation Files. (USAR Change 04-412)

ENCLOSURE 3

- In section 10.3.4.3, third to last paragraph, a description of an air compressor and 30 gallon air receiver was deleted. This equipment has been removed. Two Chapter 15 drawings, NH-36881 and NH-36049-14, were revised accordingly. (USAR Change 04-202)

- In section 10.3.10.1, first paragraph, a portion of the description of the design basis of the Post-Loss of Coolant Accident Sampling System (PASS) was removed. This change reflects removal of the PASS from Technical Specifications per License Amendment 136, as approved by NRC in a letter dated June 17, 2003 (TAC No. MB8063). A description of the associated commitments made to NRC in the License Amendment was added to this section. (USAR Change 04-002)

- Reference 20 in Appendix I has been deleted. Reference 20 was drawing NH-11145, "ESW-EFT ESW Crosstie Division A ESW SW30A-3"-HF Routing (Issued for Construction), Revision 0, 10/14/87." These cross ties are shown in USAR Appendix I, References 18 and 19. (USAR Change 04-909)

ENCLOSURE 4

MONTICELLO NUCLEAR GENERATING PLANT
USAR REVISION 21

Replace the following sections with the attached revised sections.

REMOVE	INSERT
VOLUME 1	
List of Effective Pages	List of Effective Pages
Section 1	
1 TOC	1 TOC
1.3	1.3
1.FIGURES	1.FIGURES
Section 2	
2 TOC	2 TOC
2.3	2.3
2.4	2.4
2.9	2.9
Section 3	
3 TOC	3 TOC
3.5	3.5
Section 4	
4 TOC	4 TOC
4.3	4.3
VOLUME 2	
Section 5	
5 TOC	5 TOC
5.2	5.2
5.4	5.4
5.FIGURES	5.FIGURES
Section 6	
6 TOC	6 TOC
6.1	6.1
6.2	6.2
6.6	6.6
6.FIGURES	6.FIGURES

ENCLOSURE 4

REMOVE	INSERT
Section 7	
7 TOC	7 TOC
7.2	7.2
7.3	7.3
7.5	7.5
7.6	7.6
7.9	7.9
7.14	7.14
7.FIGURES	7.FIGURES
VOLUME 3	
Section 8	
8 TOC	8 TOC
8.3	8.3
8.4	8.4
8.9	8.9
Section 10	
10 TOC	10 TOC
10.2	10.2
10.3	10.3
10.4	10.4
10.5	10.5
VOLUME 4	
Section 12	
12 TOC	12 TOC
12.2	12.2
Section 14	
14 TOC	14 TOC
14.7	14.7
14.9	14.9
14.11	14.11
14.FIGURES	14.FIGURES
VOLUME 5	
15 TOC	15 TOC
Remove all drawings	Replace with new drawings
VOLUME 8	
Appendix I TOC	Appendix I TOC
I.2	I.2
I.4	I.4
I.5	I.5
I.6	I.6

ENCLOSURE 5

MONTICELLO NUCLEAR GENERATING PLANT
REPORT OF CHANGES TO MONTICELLO FIRE PROTECTION PROGRAM

This section contains a report of changes to the Monticello Fire Protection Program (FPP) in accordance with the provisions of 10 CFR 50.71(e), 10 CFR 50.59, and Generic Letter (GL) 86-10.

PART 1

In conformance with GL 86-10, the following FPP documents were revised since the previous submittal.

1. No changes were made to the Updated Fire Hazards Analysis during this reporting period.

2. The Safe Shutdown Analysis (SSDA) was revised to incorporate various administrative changes. These changes included:

   - Revised technical evaluations to clarify the use of the Alternate Shutdown Method for the Cable Spreading Room within Fire Area VI. The previous version of the SSDA did not clearly identify that the safe shutdown method for the Cable Spreading Room differed from the remainder of Fire Area VI. No technical change was made to the post-fire shutdown method or compliance strategy by this revision.
   - Cable C101-C91-01 was removed from the SSDA. The analysis of this cable determined it is not associated with any safe shutdown components. No technical change was made to the post-fire shutdown method or compliance strategy by this revision.

3. The Fire Protection Program Plan has not been changed during this reporting period.

4. The Operations Manual, B.08.05-05, Tables, A.2-1, A.2-2, A.2-3 and A.2-4 are incorporated in the USAR by reference. Revision 29 and 30 to B.08.05-05 changed the number of required operable smoke detectors in various fire zones to reflect USAR and National Fire Protection Association requirements. Revision 30 also allows a dedicated operator to briefly switch a fire pump out of its normal automatic initiation mode (see Item 4 in Enclosure 1 to this letter). A number of editorial and administrative changes were also made.

ENCLOSURE 5

PART 2

In accordance with License Amendment 119 to Monticello Facility Operating License, Monticello committed to provide a summary of the Out of Service (OOS) reports that were submitted to the Operations Committee for fire protection system impairments.

Consistent with the requirements of the Monticello FPP, a summary of OOS reports regarding fire protection equipment declared inoperable is to be provided to the NRC:

Impairment	Basis for Reporting	Description of Impairment & Action Taken
Detection	Minimum number of detectors not operable for > 14 days	A detection system was declared inoperable as a result of a failure in the fire detection panel. A compensatory measure was in place until the system was returned to service.
Detection	Minimum number of detectors not operable for > 14 days	A detection system was declared inoperable to accommodate implementation of a modification to perform improvements to the detection system. Compensatory measures were in place until completion of the detection system improvements.
Detection	Minimum number of detectors not operable for > 14 days	A detection system was declared inoperable as a result of the minimum number of detectors not available for a fire area. A compensatory measure was in place until the system was returned to service. An evaluation determined that the number of detectors for the fire area was incorrect based on plant configuration. This permitted the detection system to be returned to service.
Fire Barrier	Fire barrier not operable for > 14 days	An Appendix R barrier compliance issue was identified. A compensatory measure was put in place until the barrier was returned to service. Further evaluation determined that the barrier was operable but degraded and the barrier was returned to service. Subsequently, repairs were completed which returned the barrier to full operable status.