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Monticello Nuclear Generating Plant
Operated by Nuclear Management Company, LLC

L-MT-03-083
10 CFR 50.71(e)
10 CFR 50.59(d)(2)

November 20, 2003

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 20 to the Updated Safety Analysis Report

Pursuant to 10 CFR Part 50, Section 50.71(e), Revision 20 to the Updated Safety Analysis Report (USAR) for the Monticello Nuclear Generating Plant is hereby submitted. This revision completes an update of the information in the USAR for the period from April 1, 2002 through May 31, 2003.

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.

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 Exhibit A.

Exhibit B, "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." This letter contains no new NRC commitments.

Exhibit C, "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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Exhibit D contains Revision 20 of the Monticello USAR and instructions for posting the document. The USAR is being submitted electronically on CD-Rom according to the instructions in RIS 2001-005, "Guidance on Submitting Documents to the NRC by Electronic Information Exchange or on CD-ROM."

Exhibit E, "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.

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 20 to the Monticello USAR meets the requirements of 10 CFR 50.71(e) to update the USAR through May 31, 2003.

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

Enclosures:

Exhibit A	Monticello Nuclear Generating Plant Report of Changes, Tests and Experiments
Exhibit B	Monticello Nuclear Generating Plant Report of Changes to Licensee Docketed Commitments
Exhibit C	Monticello Nuclear Generating Plant Summary of Information Removed from the USAR
Exhibit D	USAR Revision 20 Changes (Enclosed separately)
Exhibit E	Report of Changes to Monticello Fire Protection Program

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

EXHIBIT A

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).

Note: In the last USAR Update submittal (USAR Rev. 19) the 10CFR50.59 summary submittal (Exhibit A, Item 25) referenced an incorrect modification number. The correct modification number is 00Q027, not 00Q270.

1. **GL 96-06, Containment Penetration Overpressurization X-12 (Modification 98Q010, Part B, Revision 2)**

   Activity Description: A check valve was added as a thermal pressure relieving device in the shutdown cooling primary containment penetration so that the penetration could withstand drywell heat up during a LOCA. Appropriate test valves were also installed to permit functional and leakage testing of the check valve.

   Note: This modification was screened and evaluated under the old 10CFR50.59 rule. Under the old 10CFR50.59 rule, an evaluation was required because a USAR drawing was affected. Under the new rule, a 10CFR50.59 evaluation would not have been required.

   Summary: The modification satisfied commitment M-99004-B MNGP made to the NRC on 4-12-2000 in response to Generic Letter 96-06.

   Conclusion: Prior NRC approval was not required.

2. **Offgas Dilution Fan Damper Control (Modification 99Q025)**

   Activity Description: The air operated dampers for the stack dilution fans V-EF-18A(B) were replaced with manually positioned dampers that are secured in position when proper flow has been established. The removal of the pneumatic system decreased the number of potential fan control failures that could directly impact the Standby Gas Treatment System (i.e., the dilution fans provide a boost to the Standby Gas Treatment System to establish required flow. The dampers control the flow.) USAR drawing NH-36159 shows the air supply and was revised to show the new configuration.

   Note: This modification was screened and evaluated under the old 10CFR50.59 rule. Under the old 10CFR50.59 rule, an evaluation was required because a USAR figure (Chapter 15, drawing NH-36159) was affected. Under the new rule, a 10CFR50.59 evaluation would not have been required.

Summary: The air operated dampers for fans V-EF-18A(B) were less reliable than the new manual dampers because there were more failure modes that could have affected the Standby Gas Treatment System. The new manual dampers are positioned to provide the correct flow requirements for the Standby Gas Treatment System. Thus system reliability was enhanced. Drawing NH-36159 was changed to show the new damper control configuration.

Conclusion: Prior NRC approval was not required.

3. **Modification to Replace Cooling Tower Fan Stacks, Upper Level Decking and Decking Support Joists (Modification 00Q280)**

Activity Description: This modification removed all existing redwood fan stacks (18 total) from both cooling towers and all existing redwood decking and support joists at the fan stack level on both cooling towers. It removed any structural members that appeared to be defective that were found during construction or identified during previous inspections. The fan stacks were replaced with prefabricated fiberglass fan stacks; the joists and other structural members were replaced with treated Douglas fir, and decking was replaced with treated plywood.

Note: This modification was screened and evaluated under the old 10CFR50.59 rule. Under the old 10CFR50.59 rule, an evaluation was required because a USAR description of the cooling towers was affected. Under the new rule, a 10CFR50.59 evaluation would not have been required.

Summary: This design change did not affect the safe operation of the plant. The Cooling Tower system is not safety related.

Conclusion: Prior NRC approval was not required.

4. **11 RHR Heat Exchanger Tube Plugging (Modification 01Q125)**

Activity Description: Up to five tubes of either Residual Heat Removal (RHR) heat exchanger may be plugged as a result of this design change.

Summary: The RHR heat exchangers' ability to transfer heat is an input for USAR Figure 5.2-15 "Containment Pressure Response to the Design Basis Accident." Plugging five tubes slightly decreases the RHR heat exchangers' ability to transfer heat. This reduction in heat transfer is bounded by measured values. The K value of 143.1 used in NEDO-30485 is unchanged.

Conclusion: Prior NRC approval was not required.

5. FT-2943 Replacement (Modification 02Q200)

Activity Description: Flow transmitter FT-2943 is associated with Standby Gas Treatment (SBGT) Loop A flow control. The modification replaced the transmitter with a different design and added fuses in the control loop. Installation of the fuses allowed the transmitter to be downgraded from safety related to standard quality and non-EQ.

Summary: The only reason FT-2943 was classified as safety related was because a short in the transmitter could blow a fuse to a power supply that also supplied the safety related portions of the instrument loop. The addition of properly coordinated fuses protected the safety related portion of the instrument loop and allowed FT-2943 to be downgraded.

Conclusion: The change increased the reliability of the SBGT flow control instrument loop. Prior NRC approval was not required.

6. RHR Service Water Pump Emergency Core Cooling System Load Shed Bypass Procedure and Bypass Switches (SRI-02-003)

Activity Description: During an NRC inspection of plant modifications and 10CFR50.59 evaluations (NRC Inspection report 50-263/00-17(DRS)), a finding involving the failure to follow plant procedures when preparing a 10CFR50.59 evaluation was identified. The finding was associated with Design Change 98Q140 which installed permanent bypass switches for the ECCS load shed lockout signals to the Residual Heat Removal Service Water (RHRSW) pumps. The bypass switches would be used following a design basis LOCA coincident with a loss of normal offsite AC power, to start an RHRSW pump for containment cooling.

Prior to the installation of Design Change 98Q140, bypass of the ECCS load shed lockout signals had been procedurally controlled using the temporary modification process and relied upon the installation of jumpers and contact boots. The bypass switches were intended to reduce the time required to start an RHRSW pump and establish containment cooling. The NRC found that the 10CFR50.59 evaluation supporting Design Change 98Q140 relied on the acceptability of the previous use of jumpers and contact boots to justify the control logic. The acceptability of the previous use of jumpers and contact boots to bypass the interlock had not been previously evaluated in accordance with 10CFR50.59.

The purpose and scope of this 10CCFR50.59 evaluation addressed the deficiencies identified by the NRC and provided an amended evaluation supporting procedural and design changes for the use of installed bypass switches to bypass the RHRSW pump ECCS load shed interlock.

Summary: Additional contacts and wiring added to two existing switches on control panel C-03 are used to accomplish the RHRSW load shed/lockout bypass function. The design and construction methods used to modify the switches conformed to, or exceeded, the original plant safety-related standards applicable to control board installations of this type. As such, the design change installation is robust. Original plant design divisional separation criteria are met. Use of the bypass switches is procedurally controlled. Therefore, the installation and use of these components cannot significantly contribute to an accident or malfunction of any SSC important to safety.

The new RHRSW load shed/lockout bypass capability ensures that long term containment cooling can be initiated in a timely manner, consistent with the plant safety analysis. They will be used in the event of a design basis LOCA in conjunction with loss of offsite AC power. The original design criteria for the Monticello plant require this capability.

Conclusion: Installation and use, under procedural control, of these switches can be accomplished under the provisions of 10CFR50.59 without prior NRC approval.

7. Alternative Mark I Containment Analysis Procedure (SRI-02-005)

Activity Description: An alternate analytical procedure (method of evaluation) was proposed for qualifying the torus attached piping (TAP) for Mark I containment related (LOCA and SRV) loading conditions. Previously, a "pseudo" coupled approach had been used (i.e., tours and piping were analyzed separately with the CMDOF program forcing a coupled response of the system). The new approach used a fully coupled approach employing one model, which included the torus and the piping.

Summary: The new approach using a fully coupled analysis method, in which the torus and associated piping are combined in a single model, has been described in the USAR by reference. Although the Mark I analyses performed to date have used a "pseudo" coupled approach, the fully coupled approach (i.e., torus and attached piping included in the same model) is also specified as an acceptable analysis procedure in the MNGP USAR and the associated NRC SER. Accordingly, the use of the fully coupled approach was not a departure from a method of evaluation described in the USAR.

Conclusion: Prior NRC approval was not required.

8. **<u>Operation of 3 Service Water Pumps (SRI-02-001)</u>**

<u>Activity Description</u>: This activity involved a procedure change to allow operation of all three service water pumps during the summer months.

<u>Summary</u>: Operation of all three service water pumps was acceptable vs. relying on a service water pump to auto start. There were no physical changes needed to the service water system to accomplish this. Operation of all three pumps was operationally desireable to provide additional margin for system flow and resulted in a more reliable service water system.

Evaluation showed that components cooled by the service water system were not adversely affected by the trip of a service water pump. Operation with three service water pumps does not affect any USAR analysis.

<u>Conclusion</u>: Prior NRC approval was not required to operate with three service water pumps.

9. **<u>Time Delay for CRD Pump Suction Trip (Modification 02Q290, Evaluation 03-001)</u>**

<u>Activity Description</u>: A modification added a time delay (up to five seconds) to the CRD pump suction trip logic. Time delay relays were installed in series with pressure switches PS-3-210A and PS-3-201B. The time delay did remove a small degree of pump protection. But the addition of the time delay in the low suction pressure trip logic prevents the operating CRD from tripping during a short pressure transient such as an unplanned HPCI start.

<u>Summary</u>: Installing a time delay in the low suction pressure trip logic for the CRD pumps introduced a manageable risk for pump damage and failure modes. Adding the time delay removed a small degree of equipment protection for the in-service pump. This is a less than minimal risk as the likelihood of pump failure is very low. The addition of the time delay also adds a new failure mode in that a failure in either the pressure switch or the relay can prevent a train from operating. This is a less than minimal risk because of the duplication in trains and the reliability of the components installed.

<u>Conclusion</u>: Considering the beneficial effects of the time delay, the net effect of the change on pump reliability and availability is positive. The positive and negative effects of the change may be combined and considered together because they result from the same element of the facility change (i.e., adding the time delay relays.) It was determined that increasing the CRD pump reliability and availability, even with the adverse effects discussed above, was acceptable. NRC approval was not required.

10. **Diesel Exhaust Missile Protection Design Consideration (Modification 03Q035, Evaluation 03-004)**

Activity Description: The capability of the diesel exhaust system to withstand tornadic winds and tornado generated missiles was questioned by the NRC. As a result, Modification 03Q035 was prepared to increase the load carrying capability of the diesel exhaust piping supports. MNGP's 10CFR50.59 screening process was applied to the modification. The screening concluded that there were two aspects of the modification that needed to be evaluated against the eight criteria of 10CFR50.59. The two aspects were:

- A design decision was made to leave the diesel exhaust piping unprotected from tornado-generated missiles. (Questions one through seven were applicable.)
- The probabilistic risk assessment technique used to show that the risk from tornado-generated missiles was sufficiently small had not been used for MNGP before. (Question eight was applicable.)

Summary: In support of a modification to reinforce the emergency diesel exhaust piping to withstand tornadic wind loads, a probabilistic risk assessment was performed to show that the risk of damage to the diesel exhaust system from tornado-generated missiles was sufficiently small, i.e., less than the guidance presented in the Standard Review Plan (SRP) Sections 2.2.3 and 3.5.1.4. This was the basis for satisfying questions one through seven.

The probabilistic risk assessment was performed in accordance with the method contained in EPRI NP-2-5 (TORMIS). This was a new methodology for MNGP that had been previously approved by the NRC for use by other licensees and in the general SRP for all plants. This methodology was applicable for the activities described. This was the basis for satisfying question eight.

Conclusion: The risk of damage to the standby diesel exhaust system from tornado-generated missiles is sufficiently small. The methodology used (TORMIS) to quantify the risk had been previously approved by the NRC. Therefore, the activities described do not require prior NRC approval and were implemented by Modification 03Q035.

11. **Justification for Maintaining LPCI Outboard Isolation Valves, MO-2012 and MO-2013 in Closed Position During Plant Operation (SRI-92-004, Rev. 1, Evaluation 03-005)**

Activity Description: SRI 92-004, Rev. 1 provides justification for maintaining LPCI outboard containment isolation valves MO-2012 and MO-2013 in the closed position during plant operation. Primary containment is not adversely affected by maintaining the inboard and outboard isolation valves in a closed position. However, the proposed change in valve alignment was a change to the method of performing the LPCI function. The normal valve alignment required one valve to operate (open) in the selected loop and one valve to operate (close) in the non-selected loop. The proposed change required two valves in the selected loop to operate (open).

Summary: The main issue was the impact of closing both the inboard and outboard LPCI injection isolation valves. The USAR contains no language regarding the initial position of the LPCI injection valves. The evaluation focused on ensuring that the required functions of LPCI and primary containment were not compromised by maintaining both isolation valves in the closed position.

Primary containment was not adversely affected by maintaining both LPCI injection valves in the closed position. Both valves in each loop are considered primary containment isolation valves and receive identical isolation signals. The piping between the valves is rated for reactor vessel pressure and temperature. Maintaining both valves in the closed position had a positive impact on the primary containment design function.

Evaluation for impact on the LPCI function focused on LPCI flow timing and the potential for an increase in core damage frequency (CDF) because two valves had to stroke open instead of one. LPCI flow timing was not affected because the outboard valves stroke quicker than the inboard valves. The normally closed inboard valve limiting stroke time (LST) bounded the stroke time of the outboard valve and provided margin to the 69 seconds assumed in the safety analysis.

Malfunction of a normally closed outboard LPCI isolation valve was addressed with respect to the potential adverse impact on LPCI injection performance/reliability. The selected flow path would require two normally closed injection valves to open rather than one. However, the PRA fault tree model for LPCI incorporates potential failures of all significant LPCI components; LPCI is not assumed to be single failure tolerant. In the context of potential malfunction of the LPCI design function, the proposed change in valve alignment did not significantly degrade the level of LPCI performance and reliability assumed in safety evaluations.

Conclusion: The proposed change to maintain the inboard and outboard LPCI injection valves in the closed position had no adverse effect on the primary containment function and a less than minimal effect on the LPCI function. Prior NRC approval was not required.

EXHIBIT B

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 the Monticello Nuclear Generating Plant. These commitments are being identified and reported to the Commission in accordance with guidance provided in NEI Technical Report 99-04 Revision 0, "Guidelines for Managing NRC Commitment Changes."

1. <u>Monticello Commitment M82082A</u>

Source Document:	Response to NRC URI Review of Special Lifting Devices 2.1.3d
Commitment:	The following special lifting devices shall be examined by NDE methods prior to each maintenance/refueling outage when usage is being considered; dryer and steam separator sling lifting device, reactor vessel head lifting device; and turbine high/low pressure rotor lifting device.
Change:	Perform NDE examination of the major load-bearing welds on these special lifting devices (dryer and steam separator sling lifting device, reactor vessel head lifting device; and turbine high/low pressure rotor lifting device) every 5 years. NDE examination of the hook pins on these lifting devices and thorough visual examination for damage and deformation will continue to be done within the last 12 months prior to using these devices. The thorough visual examination will include looking at the paint on the welds to verify that there are no cracks. If cracks in the paint are present, the paint will be removed and NDE on the weld will be performed.

2. <u>Monticello Commitment M97036A</u>

Source Document:	Response to NRC NOV dated July 18, 1997
Commitment:	After "as found evaluations" are completed, time should be allowed on the simulator for the crew to train on and validate C-4 procedures. The SM or SS should notify the training department on which C-4's their crew would like to train on. The simulator instructor should assist the crew and ensure all steps, both immediate and subsequent, are completed.
Change:	When developing a simulator lesson plan to train one or more tasks during a scenario, the goal of training and learning objectives ***SHALL*** only reflect those C-4 procedure tasks that are sufficiently completed under the assumed task performance conditions.

EXHIBIT C

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 4.4.4, fourth paragraph, a statement was deleted that said that equipment exists to remotely test the SRV bellows pressure switches. The statement was removed because modification 01Q050 (SRV Bellows Leak Detection System Changes) removed this function.

- In section 8.4.1.2, last full paragraph, a statement was removed that said that no operator action is required to reduce EDG room temperature for at least six hours following an EDG start with EDG room ventilation failure. No documented technical basis could be established for this statement. Removal is acceptable because the question was rendered moot by a previous modification that caused the ventilation dampers to fail open on loss of instrument air.

- In section 6.2.2.2.1, fourth paragraph, a statement was removed that identified that the maximum ambient conditions in the core spray corner rooms is 140° F at a relative humidity of 100%. The statement was removed because temperature and humidity conditions are not listed for other systems (e.g., HPCI, RCIC, RHR, etc.). Deleting the information from the core spray section maintains a consistent level of detail throughout the USAR.

- In section 8.3, ninth paragraph, a statement was deleted that stated that since the supplies to the redundant off-site power sources to the plant safeguards are not part of the Xcel interconnected grid, it is not likely that remote switching of the switchyard sources to these loads will be performed from the System Control Center. This statement was inaccurate and of little value. The pertinent information is in the first sentence of the paragraph, which expresses that the Xcel System Operator must obtain prior approval from the site before isolating the redundant off-site power sources. This would be done only under extreme circumstances.

- In section 10.3.9.2, three items were deleted. A statement was deleted in the middle of the first paragraph that referred to the appendix R requirements for emergency lighting. The third paragraph was deleted, which also referred to a requirement for 8-hour battery powered lights in safe shutdown areas. And the fourth paragraph was deleted, which generally described the fixtures used for emergency lights. To improve clarity and accuracy, these three items were replaced by a new paragraph that describes the emergency lights. These changes were made for the following reasons:

- o The Appendix R rule does not specifically state whether fixed or portable lights may be used to meet the emergency lighting requirement. The change gives MNGP the flexibility to use portable lights when appropriate.
- o The use of portable lights for fire fighting activities was established by the original August 1979 Safety Evaluation Report. Section 4.6 of the SER identifies that portable lights will be provided to support fire brigade activities.
- o Use of portable lights is also consistent with Regulatory Guide 1.189, section 4.1.6.2.
- o The description of light fixtures was deleted to prevent confusion. Portable lights are not housed in fixtures. Portable lights are stored in lockers.

- In Table 14.7-5 (Initial Conditions for Monticello ECCS Performance Analysis) the entry identifying an initial water level of 510.0 inches above vessel zero was removed. Also, Table 14.7-6 (Other Monticello Parameters for ECCS Performance Evaluation), which listed water level alarm setpoints, was deleted in its entirety. Deleting this information eliminates any inconsistency between the USAR and the information appearing in the currently governing ECCS analyses for MNGP. These analyses are:

  - o (1) NEDC-32514P, Rev. 1, Monticello SAFER/GESTER-LOCA Loss-of-Coolant Accident Analysis," dated October 1997, and
  - o (2) GE-NE-J1103878-09-02P, "Monticello ECCS-LOCA Evaluation for GE14," dated August 2001.

  The information deleted from Table 14.7-5 is consistent with the information listed in Table 4-1 of Item (1) and in Table 1 (Section 4) of item (2), above. Neither analysis supports the 510.0 inch initial water level. The 510.0 inch water level most likely originated from Table 4.2 of GE-NE-187-02-0392, Rev. 1, "Monticello Nuclear Generating Plant SAFER/GESTER-LOCA Analysis Basis Documentation," dated July 1993, which was a non-standard companion report to the governing ECCS analysis report at that time, NEDC-31786P, "Monticello SAFER/GESTER-LOCA Loss-of-Coolant Accident Analysis," dated December 1990.

  Similarly, neither item (1) nor item (2), above, reports any of the data shown in Table 14.7-6. The data in Table 14.7-6 appeared in Table 4-3 of GE-NE-187-02-0392, Rev. 1, but appears in no other documentation. Many of the setpoint values shown in Table 14.7-6 are irrelevant to SAFER/GESTER analysis.

  In summary, because these input values are no longer used in the ECCS analysis reports of record, it is appropriate that this information be deleted from the USAR.

- There were several deletions in Section 14 that involved descriptions of analytical methods (computer codes, etc.) that had been used by the NMC Nuclear Analysis and Design group (NAD) to perform accident and transient analysis. These analyses are now performed by Global Nuclear Fuels (GNF). The applicable GNF topical reports, which describe the GNF calculational methods, are referenced by the USAR.

- Other deletions in Section 14 described GE12 fuel. This type of fuel is no longer used at MNGP.

EXHIBIT D

USAR REVISION 20 CHANGES

ENCLOSED SEPARATELY

EXHIBIT E

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 Updated Fire Hazards Analysis (UFHA) and the Safe Shutdown Analysis (SSDA) are incorporated by reference in the Update Safety Analysis Report (USAR). The following documents were revised since the previous submittal.

1. The UFHA was revised to incorporate various administrative changes. Specifically, these changes:

    - Changed Licensee name from Northern States Power to Nuclear Management Company, LLC.
    - Revised technical evaluations to more accurately document the existing plant configuration and raise evaluation quality to industry standards. This resulted in an enhanced Fire Hazard Analysis.
    - Updated the technical qualification requirements for Fire Protection personnel.
    - Corrected various typographical errors and other inaccuracies.

2. No Changes were made to the SSDA during this reporting period.

3. The Fire Protection Program Plan has been revised since the previous submittal. The revision included correction of various administrative issues that were identified during the NRC triennial inspection and subsequent follow-up reviews. Specifically, these changes:

    - Clearly identified the Fire Protection Program owner to address the revised site organization.
    - Removed reference to the Technical Specifications in accordance with License Amendment 119 and the Monticello Operating License.
    - Updated 10CFR 50.71(e) reporting requirements.
    - Added reporting requirements for fire protection impairments

4. The Operations Manual, B.08.05-05, Tables, A.2-1, A.2-2, A.2-3 and A.2-4 have been incorporated in the USAR by reference. The last revision to this document clarified reporting impairments to the Operations Committee.

EXHIBIT E

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 (OC) for fire protection system impairments.

Consistent with the requirements of the Monticello FPP, a summary of OOS reports submitted by May 31, 2003, regarding fire protection equipment declared inoperable is provided:

Impairment	Basis for Reporting	Description of Impairment & Action Taken
Detection and App R barrier	Minimum number of detectors not operable for > 14 days & fire barrier not operable for > 14 days	Detection code compliance issues resulted in the need for modifications to various detection systems. Compensatory measures are in place until systems are returned to service. An Appendix R barrier compliance issue was identified. A compensatory measure is in place until the barrier is returned to service.
Detection	Minimum number of detectors not operable for > 14 days	A detection system was declared inoperable as a result of a failure in the control panel. A Compensatory measure is in place until the system is returned to service.
Fire Pumps	Multiple Fire Pumps OOS	The Electric and Screenwash fire pumps were removed from service to replace their respective relief valves, perform flow testing and pump packing replacement. Provisions were in place to return the fire pumps to service if the pumps were required.
Fire Pumps	Multiple Fire Pumps OOS	The fire pumps were considered inoperable since the test valve was opened and could divert suppression water. The electric fire pump was declared inoperable as a result of a failure in the control panel. The test valve was closed and the impairment was exited.

PART 3

During the NRC triennial Fire Protection inspection conducted in June 2002, the NRC identified that NMC had not properly documented past changes to the MNGP Fire Protection Program. As a result of this finding NMC reviewed all of the past revisions to the Safe Shutdown Analysis (SSDA) and Updated Fire Hazards Analysis (UFHA). The 10CFR50.59 screening criteria were applied to the changes, as necessary. These reviews indicated that no 10CFR50.59 evaluations were required.