NMC
Committed to Nuclear Excellence

Monticello Nuclear Generating Plant
Operated by Nuclear Management Company, LLC

September 24, 2003

L-MT-03-067
10 CFR Part 50
Section 50.90

U.S. 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 RELATED TO REVISED LONG-TERM CONTAINMENT RESPONSE AND NET-POSITIVE SUCTION HEAD ANALYSES (TAC NO. MB7185)

Reference 1: NMC Letter to NRC, "License Amendment Request, Dated December 6, 2002, Revised Analyses of Long-Term Containment Response and Overpressure Required for Adequate NPSH for Low Pressure ECCS Pumps"

Reference 2: NRC Letter to NMC, "Monticello Nuclear Generating Plant – Request for Additional Information Related to Revised Long-Term Containment Response and Net-Positive Suction Head Analyses," (TAC No. MB7185) dated August 19, 2003

Pursuant to 10 CFR 50.90, the Nuclear Management Company, LLC (NMC), requested U.S. Nuclear Regulatory Commission (NRC) approval of updates to the design basis loss of coolant accident containment response and containment overpressure assumed for adequate net positive suction head (NPSH) in the low pressure emergency core cooling analyses as described in the Monticello Updated Safety Analysis Report (Reference 1). The NRC staff reviewed and requested additional information (RAI) on August 19, 2003 (Reference 2). Attachment 1 provides the response to the RAI.

Attachment 2 provides several editorial corrections to previously submitted information related to the long-term containment response and NPSH analyses. Attachment 3 provides a revised non-proprietary version of the General Electric (GE) report (Revision 2) that serves as the primary basis for the proposed analysis changes. Attachment 4 provides an affidavit for the revised proprietary version of this GE report included as Attachment 5. Pursuant to 10 CFR 2.790, it is requested that Attachment 5 be withheld from public disclosure. Upon separation of Attachment 5 from this letter, the remainder of this letter may be decontrolled.

A001

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This letter makes no new commitments. If you have any questions regarding this submittal, please contact Rick Loeffler, Senior Regulatory Affairs Engineer at (763) 295-1247.

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

CC: Regional Administrator, Region III, USNRC (w/o Attachments 4 and 5)
Senior Project Manager, Monticello, USNRC
Senior Resident Inspector, Monticello, USNRC (w/o Attachments 4 and 5)
Minnesota Department of Commerce (w/o Attachments 4 and 5)

Attachment 1: Response to a Request for Additional Information Related to Revised Long-Term Containment Response and Net-Positive Suction Head Analyses

Attachment 2: Corrections to Previously Submitted Information Related to Revised Long-Term Containment Response and Net-Positive Suction Head Analyses

Attachment 3: GE Nuclear Energy Report, GE-NE-0000-0002-8817-01-R2, "Monticello Nuclear Generating Plant Long-Term Containment Analysis," Revision 2, Dated August 2003, Non-Proprietary Version

Attachment 4: Affidavit for GE Nuclear Energy Report, GE-NE-0000-0002-8817-01-R2, "Monticello Nuclear Generating Plant Long-Term Containment Analysis," Revision 2, Dated August 2003, Proprietary Version

Attachment 5: GE Nuclear Energy Report, GE-NE-0000-0002-8817-01-R2, "Monticello Nuclear Generating Plant Long-Term Containment Analysis," Revision 2, Dated August 2003, Proprietary Version

Attachment 1

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

RESPONSE TO A REQUEST FOR ADDITIONAL INFORMATION RELATED TO REVISED LONG-TERM CONTAINMENT RESPONSE AND NET-POSITIVE SUCTION HEAD ANALYSES

9 pages follow

RESPONSE TO A REQUEST FOR ADDITIONAL INFORMATION RELATED TO REVISED LONG-TERM CONTAINMENT RESPONSE AND NET-POSITIVE SUCTION HEAD ANALYSES

On December 6, 2002, the Nuclear Management Company, LLC (NMC) requested U.S. Nuclear Regulatory Commission (NRC) approval of updates to the design basis loss of coolant accident (LOCA) containment response and containment overpressure assumed for adequate net positive suction head (NPSH) in the low pressure emergency core cooling analyses described in the Monticello Nuclear Generating Plant (MNGP) Updated Safety Analysis Report (USAR). The NRC staff requested additional information (RAI) pertaining to this submittal on August 19, 2003. The NRC questions or requests are shown in 'bold' text below and the NMC response is provided in 'standard' text immediately after.

1. (a) **What assurance is there that the K value will remain at 147 or above?**

"RHR [Residual Heat Removal] Heat Exchanger Efficiency Test" 1136 periodically checks the efficiency of the RHR Heat Exchangers and provides indication whether any deterioration has occurred in the heat removal capabilities. The results of the RHR Heat Exchanger Efficiency test are trended and based on those trends the RHR heat exchangers are cleaned to maintain the K value at or above it's design bases value.

(b) **How often does NMC verify this?**

The RHR Heat Exchanger Efficiency Test 1136 is an annual surveillance.

(c) **Has NMC made a measurement to verify that the K value is currently greater than 147?**

NMC periodically measures the K value. The most recent RHR Heat Exchanger Efficiency test, 1136, for the # 11 RHR heat exchanger was conducted on November 19, 2002, and the corrected K value accounting for uncertainty was 170.97 BTU/sec-°F. The most recent RHR Heat Exchanger Efficiency test, 1136, for the # 12 RHR heat exchanger was conducted on February 16, 2003, and the corrected K value accounting for uncertainty was 159.9 BTU/sec-°F.

2. **If NMC has revised the calculation of residual heat removal room temperature from the analysis provided in NMC's March 4, 1997, letter to the NRC, briefly describe the changes and the conclusions.**

The March 4, 1997, submittal (Reference 1) updated the design basis accident (DBA) containment temperature and pressure response utilized in the evaluation of the RHR Room temperatures during a DBA LOCA. This submittal supplemented the original License Amendment Request (LAR) dated January 23, 1997.

By letter dated June 19, 1997, a complete revision and replacement of the January 23, 1997, LAR was provided (Reference 2). The scope of the original LAR was expanded to address the use of containment pressure for assuring adequate NPSH for emergency core cooling system (ECCS) pumps under worst-case conditions. Calculation CA-97-157, Revision 0, dated June 13, 1997, summarized in Exhibit G of Reference 2, evaluated the RHR Room temperatures based upon revised suppression pool (torus) water temperature profiles provided by General Electric (GE) in conjunction with the re-rate analysis for several different cases. On July 25, 1997, the NRC issued a Safety Evaluation (SE) authorizing reliance on containment pressure to support NPSH requirements for the ECCS pumps following a DBA (Reference 3).

In conjunction with the above, Northern States Power (NSP) pursued a power uprate for the MNGP. Section 2.8.c, "Cooling Water Systems," Item (1), "Emergency Service Water System," in the SE for the power uprate (Reference 4) indicated that NSP had "performed a bounding evaluation of the RHR room cooler performance during suppression pool heatup conditions for power uprate" and that the results were approved in an NRC SE dated July 25, 1997, i.e., Reference 3 above.

Calculation CA-97-157, Revision 1, dated May 16, 2002, expanded on CA-97-157, Revision 0 as follows:

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1. Northern States Power letter to the U.S. Nuclear Regulatory Commission, "Supplement No. 1 to License Amendment Request Dated January 23,1997; Update of Design Basis Accident Containment Temperature and Pressure Response," dated March 4, 1997.
2. Northern States Power letter to the U.S. Nuclear Regulatory Commission, "Revision 2 to License Amendment Request Dated January 23,1997; Update of Design Basis Accident Containment Temperature and Pressure Response," dated June 19, 1997.
3 U.S. Nuclear Regulatory Commission to Northern States Power Company, "Monticello Nuclear Generating Plant – Issuance of Amendment Re: Updated Analysis of DBA Containment Temperature and Pressure Response and Reliance on Containment Pressure to Compensate for Potential Deficiency in NPSH for ECCS Pumps During DBA," (TAC No. M97781) dated July 25, 1997.
4. U.S. Nuclear Regulatory Commission to Northern States Power Company, "Monticello Nuclear Generating Plant – Issuance of Amendment Re: Power Uprate Program," (TAC No. M96238) dated September 16, 1998.

- Calculation CA-97-157 was revised to extend the modeling of the RHR room temperature to 198 days post-accident. The Monticello Nuclear Generating Plant (MNGP) equipment qualification (EQ) program uses 198 days for the equipment qualification in the RHR rooms. Revision 0 of the calculation had assumed an RHR pump motor was turned-off to maintain the room temperature below 140°F after an accident.
- A case was run in which no insulation was assumed on the pumps and piping components except for the RHR Heat Exchanger.
- The RHR pump minimum flow lines were assumed to be hot to bound the possibility of the minimum flow valves failing open.

The results were:

RHR Room	
Case	Maximum RHR Room Temperature (°F)
2 pump/2 pump, insulated	142.9
2 pump/2 pump, uninsulated*	143.8
1 pump/1 pump, uninsulated*	135.6

* All piping and pumps are uninsulated. Heat exchanger is insulated.

Calculation CA-97-157, Revision 2, dated October 14, 2002, assumed that the RHR Room Coolers started 13 minutes into an accident. This increased the maximum temperature in the RHR "A" and "B" Rooms by less then 0.01 °F, a negligible increase.

3. **Briefly describe the analysis that concludes that the piping temperature limit can be increased to 196.7 degrees F.**

The piping and supports affected by the postulated long-term LOCA temperature of 196.7°F were evaluated by direct ratio of the original rerate thermal stress results at 195°F (References 5 and 6). Re-calculation of the stresses and Code compliance checks for the governing load combinations followed. Accordingly, consistency with the original licensing basis analysis methodology was preserved.

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5. Northern States Power letter to the U.S. Nuclear Regulatory Commission, "License Amendment Request Dated July 26,1996; Supporting the Monticello Nuclear Generating Plant Power Rerate Program," dated July 26, 1996.
6. Northern States Power letter to the U.S. Nuclear Regulatory Commission, "Revision 1 to License Amendment Request Dated July 26,1996; Supporting the Monticello Nuclear Generating Plant Power Rerate Program," dated December 4, 1997.

4. **Describe the SAFER/GESTR models and the assumptions used to calculate Monticello's response to a vessel isolation with high-pressure coolant injection (HPCI) unavailable. Include a nodalization diagram. Describe any conservatism in this analysis.**

The GESTR-LOCA model provides the parameters to initialize the fuel stored energy and fuel rod fission gas inventory at the onset of a postulated LOCA for input to SAFER. GESTR-LOCA also establishes the initial transient pellet-cladding gap conductance for input to SAFER.

The SAFER evaluation model calculates the long-term system response of the reactor over a complete spectrum of hypothetical break sizes and locations. SAFER is compatible with the GESTR-LOCA fuel rod model for gap conductance and fission gas release. SAFER calculates the core and vessel water levels, system pressure response, Emergency Core Cooling System (ECCS) performance, and other primary thermal-hydraulic phenomena occurring in the reactor as a function of time. SAFER realistically models all regimes of heat transfer that occur inside the core, and provides the peak cladding temperature (PCT) and the heat transfer coefficients (which determine the severity of the temperature change) as a function of time.

The SAFER/GESTR models are documented in Volume II of a proprietary General Electric (GE) Report NEDE-23785-1-PA (Reference 7). Figure 3-1 in this report provides a nodalization diagram. The analysis assumptions are consistent with the nominal calculational assumptions described in Volume III of this report, with the exception of the overall plant power level, which is initiated at 102 percent of rated. This power assumption provides the only additional conservatism in the analysis which overall is intended to produce realistic system responses.

5. **Regarding Section 4.5 of General Electric's report GE-NE-0000-0002-8817-01, R1, dated September 2002, "Monticello Nuclear Generating Plant Long-term Containment Analysis," explain how it is physically possible to have a service water temperature of 94 degrees F and a suppression pool temperature of 90 degrees F under steady state conditions.**

The ultimate heat sink and the source of the Residual Heat Removal Service Water (RHRSW) coolant is the Mississippi river. The highest river temperature experienced during recent MNGP operation was approximately 85.9°F. This temperature was experienced in August of 1987 and in August of 2001. The assumed RHRSW service water temperature of 94°F was chosen as a conservative bounding value. The initial suppression pool (SP) temperature of 90°F was chosen since this is the maximum SP water temperature allowed

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7. GE Nuclear Energy Report, NEDE-23785-1-PA, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident," Volume II and III, Revision 1, October 1984.

during normal operation by Technical Specification 3.7.A.1.a, "Primary Containment, Suppression Pool Volume and Temperature." In the summer, the average hourly river temperature cycles from a high during the day to a low during the night. During times of high river temperature, it is possible to maintain the SP temperature below the river temperature during normal operation by suspending testing that adds heat to the SP, and if necessary, only running SP cooling at night when the RHRSW intake water is cooler. High river temperatures are only experienced during short periods of time.

6. **Section 4.4 of GE-NE-0000-0002-8817-01, R1, page 4-12, begins by discussing the design basis loss-of-coolant accident (LOCA) analysis with the updated heat exchanger K value and "updated data." What are these updated data?**

The original equipment manufacturer (OEM) of the RHR heat exchangers was contacted to provide expected performance analysis at the original design conditions, at expected peak suppression pool temperature conditions and at expected RHR Heat Exchanger Efficiency Test 1136 temperature conditions. The "Updated Data" was the OEM expected performance analysis at the original design conditions converted to a "K" value and verified by NMC.

7. **Verify that the information in the table below is correct.**

Break Size	Residual Heat Removal (RHR) Heat Exchanger K	Service Water Temp °F	Peak Suppression Pool (SP) Temp [°F]	Comment
Large break*	143.1**	90	195.6	Direct SP cooling
Large break	147	90	194.1	Direct SP cooling
Large break	147	90	194.2	Containment spray cooling
Large break	147	94***	~~195.8~~ 196.5	Direct SP cooling
Large break	147	94	~~196.5~~ 196.2	Containment spray cooling
Reactor isolation	143.1	90	194.0	One RHR loop, HPCI unavailable, direct SP cooling
Reactor isolation	143.1	90	167.0	Two RHR loops with HPCI unavailable, direct SP cooling
.01 ft^2	143.1	90	190.0	One RHR loop with HPCI unavailable, direct SP cooling
.1 ft^2	143.1	90	191.2	One RHR loop with HPCI unavailable, direct SP cooling

* **A study of single failures in the June 19, 1997, NMC application showed the failure of one emergency diesel generator with loss of offsite power to be most limiting.**

** **As stated above, the original K value for the RHR heat exchanger is 143.1 BTU/sec-°F while the updated value is 147 BTU/sec-°F.**

*** **SP water temperature remains at 90 degrees F.**

The information provided in the table above is correct, except for the value of the peak SP temperature, for the large break cases with a Service Water (SW) temperature of 94°F. The values for the peak SP temperature (large break, SW temperature of 94°F) with direct SP cooling is 196.5°F, and with containment spray cooling is 196.2°F, as indicated above.

Report GE-NE-0000-0002-8817-01-R1 presents the results for the above two cases in Sections 2.1 and 4.5. The correct values are presented in Section 2.1, whereas Section 4.5 shows incorrect values as listed in the table for Question 7. These errors have been corrected in a revised version of the report, provided in Attachment 3 (Reference 8), with a proprietary version provided in Attachment 5.

8. **Describe how heat transfer to structures is modeled for the net positive suction head calculations.**

For the heat transfer to structures, the methodology described in Appendix B of NUREG-0588, "Interim Staff Position on Equipment Qualification of Safety-Related Electrical Equipment," was used. Specifically, the Uchida heat transfer correlation was used while in the condensing mode. A natural convection heat transfer coefficient was used when not in the condensing mode.

9. **In Exhibit F, Figures 8, 9, 10, and 11, NMC showed required and available overpressure for the isolation event and the Appendix R event. What is the source of the pressure for these events since the steam from the safety/relief valves is condensed in the suppression pool?**

During both the isolation and Appendix R events, core decay heat is transferred to the suppression chamber pool by the safety relief valves. As the suppression chamber (pool) and air space temperatures increase, the confined non-condensable within the suppression chamber will cause the pressure inside primary containment to increase.

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8. GE Nuclear Energy Report, GE-NE-0000-0002-8817-01-R2, Class III, "Monticello Nuclear Generating Plant Long-term Containment Analysis," Revision 2, August 2003.

10. **Verify that the table below is correct.**

Accident Scenario	Current Licensing Basis	Value	Proposed Change to Licensing Basis	Value
Peak containment pressure ((short-term large-break (LB) LOCA))	**Date:** ~~7/26/96~~ 12/04/97 **Power:** ~~1880~~ 102% of 1880 **megawatts thermal (Mwt)**	~~40~~ 39.5 psig	**Unchanged**	
Peak containment temperature (short-term LB LOCA)	**Date:** ~~7/26/96~~ 12/04/97 **Power:** ~~1880~~ 102% of 1880 Mwt	**331 degrees F**	**Unchanged**	
Peak bulk pool temp (long-term LB LOCA)	**Date:** ~~6/19/97~~ 12/04/97 **Power:** 102% of **1880 Mwt**		**Date: 12/6/02 Power:** 102% of **1775 Mwt** Decay Heat: Nominal plus 2σ	~~195.6 degrees F~~
Case 1 SW Temp 90°F	"K"= 143.1 SW Temp = 90°F Initial SP Temp = 90°F	**194.2 degrees F**	"K"= 147 SW Temp = 90°F Initial SP Temp = 90°F	194.2 degrees F
Case 2 SW Temp 94°F	None	None	"K"= 147 SW Temp = 94°F Initial SP Temp = 90°F	196.5 degrees F
Max local pool temperature (short-term LB LOCA)	**Date:** ~~7/26/96~~ Date: January 18, 2002 Eliminated ~~Power: 1880 Mwt~~	~~194 degrees F~~ N/A	**Unchanged**	
Drywell wall temperature (small steam line break)	**Date:** ~~7/26/96~~ 12/04/97 **Power:** ~~1880~~ 102% of 1880 Mwt	**273 degrees F**	**Unchanged**	
Reactor isolation peak pool temperature	**None**		**Date: 12/6/02 Power:** 102% of **1775 Mwt** Decay Heat: Nominal plus 2σ	**194 degrees F**

The information provided in the above table is correct, except for the following information that should be modified as marked-up above, and is discussed below. Items are either crossed-out or added in 'standard' text above.

The date and power level listed in the Current Licensing Basis column should be "12/04/97" and "102% of 1880 megawatts thermal" for the peak Containment pressure and temperature (short-term large-break LOCA event), and for the Drywell wall temperature (small steam line break). The peak Containment pressure (short-term large-break LOCA event) value should be "39.5 psig" rather than "40.0 psig".

Under the peak bulk pool temperature (long-term large-break LOCA) table entry, it is suggested that a distinction be made between the peak bulk pool temperatures achieved with the two sets of initial conditions; i.e., with a K value of either 143.1 or 147, and an initial SP temperature of either 90°F or 94°F. To do this two additional rows have been added under this entry. Note, that the power level for which the proposed analysis was performed was actually "102% of 1775 Mwt" and the decay heat profile used was nominal plus 2σ.

The maximum local suppression pool temperature requirement (short-term large-break LOCA) was eliminated as part of the MNGP licensing basis by a letter dated May 31, 2001, and approved by an NRC SE dated January 18, 2002.

For the reactor isolation peak pool temperature event, in the Proposed Change to Licensing Basis column, the power level for which the analysis was performed was actually "102% of 1775 Mwt" and the decay heat profile used was nominal plus 2σ.

A final revised version of the table is provided below for clarity.

Accident Scenario	Current Licensing Basis	Value	Proposed Change to Licensing Basis	Value
Peak containment pressure (short-term large-break (LB) LOCA)	Date: 12/04/97 Power: 102% of 1880 Megawatts thermal (Mwt)	39.5 psig	Unchanged	
Peak containment temperature (short-term LB LOCA)	Date: 12/04/97 Power: 102% of 1880 Mwt	331 degrees F	Unchanged	
Peak bulk pool temp (long-term LB LOCA)	Date: 12/04/97 Power: 102% of 1880 Mwt		Date: 12/6/02 Power: 102% of 1775 Mwt Decay Heat: Nominal plus 2σ	
Case 1 SW Temp 90°F	"K"= 143.1 SW Temp = 90°F Initial SP Temp = 90°F	194.2 degrees F	"K"= 147 SW Temp = 90°F Initial SP Temp = 90°F	194.2 degrees F
Case 2 SW Temp 94°F	None	None	"K"= 147 SW Temp = 94°F Initial SP Temp = 90°F	196.5 degrees F
Max local pool temperature (short-term LB LOCA)	Date: January 18, 2002 Eliminated	N/A	Unchanged	
Drywell wall temperature (small steam line break)	Date: 12/04/97 Power: 102% of 1880 Mwt	273 degrees F	Unchanged	
Reactor isolation peak pool temperature	None		Date: 12/6/02 Power: 102% of 1775 Mwt Decay Heat: Nominal plus 2σ	194 degrees F

11. **Verify that there has been no change in Monticello's licensing basis for calculating the debris loading on the emergency core cooling system suction strainers.**

There has been no change in the licensing basis for calculating the debris loading on the ECCS suction strainers. NPSH calculations consider the head loss due to debris loading on the ECCS suction strainers.

12. **What value of required net position suction head used for the calculation of required containment overpressure?**

The value of the required net position suction head used for the calculation of the required containment overpressure was:

- Short term Cases RHR – 27 feet Core Spray – 27 feet
- Long term Cases RHR – 26 feet Core Spray – 28.5 feet

13. **Regarding Exhibit F, describe, or reference, how the effects of pipe friction are accounted for, including the increase to account for aging?**

To model the effects that aging has on the friction factor of a pipe, the lengths of each pipe in the FLO-SERIES model were increased by 15 percent. The increase in pipe length by 15 percent will increase the head loss of the particular pipe by 15 percent, thus simulating a 15 percent increase of the pipe friction factor.

Attachment 2

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

CORRECTIONS TO PREVIOUSLY SUBMITTED INFORMATION RELATED TO REVISED LONG-TERM CONTAINMENT RESPONSE AND NET-POSITIVE SUCTION HEAD ANALYSES

4 pages follow

CORRECTIONS TO PREVIOUSLY SUBMITTED INFORMATION RELATED TO REVISED LONG-TERM CONTAINMENT RESPONSE AND NET-POSITIVE SUCTION HEAD ANALYSES

During development of the response to this request for additional information three corrections were identified to the information previously submitted. These corrections are:

1) The last sentence on page A-5 of Exhibit A incorrectly states the current differential pressure assumed between the wetwell and drywell vacuum breakers for them to fully open was 0.5 psid. The value listed, the then current assumed pressure differential, should have been 0.25 psid.

   This correction should have no effect on review because this parameter was modified as part of these analyses, i.e., reduced to 0.0 psid as an assumption in the net positive suction head analyses. A corrected mark-up page is enclosed.

2) In Exhibit B, on a page describing changes to USAR Table 5.2-7, "Assumptions for the LOCA Containment Evaluation," an incorrect month is listed for the GE report specified as the basis for number 3, which refers to item (6) in the table. The correct month for GE report GE-NE-0000-0002-8817-01, Rev 1, is September rather than June 2002.

   This correction should have no effect on review because this same report is specified throughout as the source of this information. A corrected mark-up page is enclosed.

3) In the first paragraph on page F-7 of Exhibit F a reference was made to a "(Ref. 7)" implying that there was a Reference 7 in the exhibit and by implication that other references were missing. The paragraph was cut and pasted into the calculation summary (Exhibit F) from the calculation. Reference 7 should have been deleted. Reference 7 in the calculation refers to the Steam Tables, Keenan, Keene, Hills and Moore, 1969. This correction should have no effect on review. A corrected mark-up page is enclosed.

The marked-up pages discussed above are enclosed immediately following.

A corrected Revision 2 of General Electric report GE-NE-0000-0002-8817-01 (Reference 1) is attached which supercedes Revision 1 of this report provided in our submittal dated December 16, 2002 (Reference 2). Revision 2 contains a listing of changes made to the document. These changes do not alter the statements made in the previous submittal based on the above discussions.

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1. GE Nuclear Energy Report, GE-NE-0000-0002-8817-01-R2, Class III, "Monticello Nuclear Generating Plant Long-term Containment Analysis," Revision 2, August 2003.
2. Nuclear Management Company letter to the U.S. Nuclear Regulatory Commission, "License Amendment Request Dated December 6, 2002, Revised Analyses of Long-Term Containment Response and Overpressure Required for Adequate NPSH for Low Pressure ECCS Pumps," dated December 6, 2002.

A small change in the shape of the plotted line representing the NRC approved containment pressure for NPSH purposes is being proposed in this License Amendment Request.

Bounding containment response cases for NPSH purposes are:

Short Term (< 600 sec)	-	DBA LOCA with LPCI Loop Select Logic Failure
Long Term (≥ 600 sec)	-	DBA LOCA with containment spray cooling and loss of offsite power and failure of one emergency diesel generator

Wetwell pressure and water temperature were recalculated by GE using the updated decay heat, updated RHR heat exchanger K values, and other updated inputs described above. Results are presented in Appendices C and D of Exhibit C.

NPSH requirements for RHR and core spray pumps were recalculated using the new containment temperature responses calculated by GE. The USAR design bases assumptions for head loss due to LOCA generated debris collecting on the ECCS suction strainers were used in these calculations. The methodology and results of these calculations are presented in Exhibit F, Cases 1, 2, and 3.

Additional cases presented in Exhibit F confirm the adequacy of available pump NPSH for a medium LOCA, shutdown from the Alternate Shutdown Panel (Appendix R requirement), and reactor isolation conditions.

Refer to Exhibit F for a summary of the calculations and results associated with the NPSH analyses.

The new long term DBA LOCA wetwell pressure response with containment spray cooling is reduced by a small amount in the new GE analysis during the period from 2000 to 4000 seconds. During this interval wetwell pressure falls below the previously approved NRC limit graph showing containment pressure NPSH credit. To accommodate this change, a reduction in the approved NPSH credit from 18.26 to 17.51 psia is requested in this time interval.

A conservative change in assumptions related to thermal equilibrium between wetwell air and water volumes in the GE model is the primary reason for the pressure reduction in the interval from 2000 to 4000 seconds. Additionally, the pressure difference required between the wetwell and drywell for vacuum breakers to fully open was conservatively reduced from ~~0.5~~ to 0.0 psid in the current NPSH analyses.

0.25

Necessary changes to: Table 5.2-7 <u>Assumptions for the LOCA Containment Evaluation</u>, RHR 5.2, page 72 of 73

1. Change item (1) to Table to read:

   Reactor is at 102% OF 1775 MWt

   Change BASIS to "GE-NE-0000-0002-8817-01, Rev 1, September 2002"

2. Change item (4) to read:

   Drywell temperature and humidity are assumed to be 135 °F and 20% RH, respectively, for DBA-LOCA, isolation, and intermediate break cases. For NPSH cases, drywell temperature and humidty are assumed to be 150 °F and 100% RH, respectively.

   Change BASIS to "GE-NE-0000-0002-8817-01, Rev 1, September 2002."

3. Change item (6) to read:

   The wetwell air space is in thermal equilibrium with the suppression pool, except for NPSH cases.. In the NPSH cases, the wetwell air space is in thermal equilibrium with the wetwell water during the early blow down period, then a mechanistic heat and mass transfer is assumed.

   Change BASIS to "GE-NE-0000-0002-8817-01, Rev 1, ~~June~~ September 2002."

4. Change item (7) to read:

   Initial drywell pressure and wetwell pressure assumed to be 16.7 psia, except for NPSH cases, where 14.26 psia is assumed.

   Change BASIS to "GE-NE-0000-0002-8817-01, Rev 1, September 2002"

5. Change item (12) to read:

   Decay heat for long-term response calculated using ANSI/ANS 5.1-1979 standard consistent with GE SIL 636. A 2-sigma uncertainty was added to nominal decay heat values.

   Change BASIS to "GE-NE-0000-0002-8817-01, Rev 1, September 2002"

Case 2 – Long Term, DBA LOCA with #11 diesel generator failure.
Case 3 – Long Term, DBA LOCA with #12 diesel generator failure.
Case 4 – Long Term, DBA LOCA with #12 diesel generator failure and medium sized LOCA, fiber debris being generated.
Case 5 – Long Term, Appendix R event with #12 diesel generator available.
Case 6 – Long Term, Isolation event with #12 diesel generator failure.

NPSH available (NPSHA) is calculated for each operating pump, at the specified time interval and associated temperature, for each case. The pump suction pressure and velocity that is determined from the FLO-SERIES results is used, along with the specific weight and vapor pressure for that specified temperature ~~(Ref. 7)~~ to calculate the NPSHA. The NPSHA is then subtracted from the NPSH required (NPSHR), and that result is the containment pressure that is needed to satisfy NPSHR. The atmospheric pressure is added to this value to provide absolute pressure values. Plots of the required containment pressure versus the approved containment pressure for the limiting pumps are provided in the Results section to demonstrate that the acceptance criteria have been satisfied.

Results / Conclusion

The following eleven figures graphically demonstrate that the containment pressure required for adequate NPSH for the limiting low pressure ECCS pumps is less than the approved containment pressure and the requested containment pressure. This demonstrates that the acceptance criterion has been satisfied.

Attachment 4

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

AFFIDAVIT FOR GE NUCLEAR ENERGY REPORT, GE-NE-0000-0002-8817-01-R2
MONTICELLO NUCLEAR GENERATING PLANT LONG-TERM
CONTAINMENT ANALYSIS, REVISION 2,
DATED AUGUST 2003, PROPRIETARY VERSION

Affidavit follows

General Electric Company

AFFIDAVIT

I, George B. Stramback, state as follows:

(1) I am Manager, Regulatory Services, General Electric Company ("GE") and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in the GE proprietary report, GE-NE-0000-0002-8817-01, *Monticello Nuclear Generating Plant Long-term Containment Analysis,* Revision 2, Class III (GE Proprietary Information), dated August 2003. The proprietary information is delineated by bars marked in the margin adjacent to the specific material, as was done in Revision 1 of the report. The basis for the proprietary determination for all material that is so marked in this report is provided in paragraph (3) of this affidavit. Specific information that is not so marked is not GE proprietary. There are no changes to the proprietary information relative to the Revision 1 report.

(3) In making this application for withholding of proprietary information of which it is the owner, GE relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), and 2.790(a)(4) for "trade secrets" (Exemption 4). The material for which exemption from disclosure is here sought also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by General Electric's competitors without license from General Electric constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information which reveals aspects of past, present, or future General Electric customer-funded development plans and programs, resulting in potential products to General Electric;

d. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in both paragraphs (4)a., and (4)b, above.

(5) To address 10 CFR 2.790 (b) (4), the information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GE, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GE, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within GE is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his delegate), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GE are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information identified in paragraph (2), above, is classified as proprietary because it contains detailed results of analytical models, methods and processes, including computer codes, which GE has developed, discussed with the NRC, and applies in the Containment analyses for the BWR.

The development of the containment computer code was achieved at a significant cost, on the order of several million dollars, to GE.

The development of the evaluation process along with the interpretation and application of the analytical results is derived from the extensive experience database that constitutes a major GE asset.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GE's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GE's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical and NRC review costs comprise a substantial investment of time and money by GE.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

GE's competitive advantage will be lost if its competitors are able to use the results of the GE experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GE would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GE of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information, and belief.

Executed on this 28th day of August 2003.

George B. Stramback

George B. Stramback
General Electric Company