NMC
Committed to Nuclear Excellence

Monticello Nuclear Generating Plant
2807 West County Road 75
Monticello, MN 55362-9637

Operated by Nuclear Management Company LLC

August 16, 2000

10CFR Part 50
Section 50.46(a)(3)

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

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

2000 Report of Changes and Errors in ECCS Evaluation Models

References:

1.) GE Report NEDC-32514P, Revision 1, "Monticello SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," dated October 1997. This report is Exhibit G of "Revision 1 to License Amendment Request dated July 26, 1996 supporting the Monticello Nuclear Generating Plant Power Rerate Program," from Michael F. Hammer (NSP) to US NRC Document Control Desk, December 4, 1997.

2.) Letter TGO:00-058 entitled "Summary of Changes and Errors in ECCS Evaluation Models," from TG Orr (GNF) to DL Orrock (NSP), dated July 29, 2000. (Attached)

3.) Letter entitled "1999 Report of Changes and Errors in ECCS Evaluation Models," from MF Hammer (NSP) to US NRC Document Control Desk, September 9, 1999.

Pursuant to 10 CFR 50.46(a)(3) the following is the required annual report of any change or error identified in ECCS analytical models or their application. Monticello's LOCA analysis of record (AOR) is contained in the License Amendment Request for Plant Rerate (Reference 1). Correspondence, from Global Nuclear Fuel to Northern States Power Company (NSP), reporting errors and changes to the ECCS Analysis is denoted as Reference 2. No changes or errors in the calculated Peak Clad Temperature (PCT) of fuel used in the Monticello reactor were reported during the period from July 1999 through July 2000.

A001

The table below summarizes the calculated Peak Clad Temperature (PCT) from the AOR.

Monticello Licensing Basis Peak Clad Temperatures

Fuel Type	*PCT (°F)*	*Reference*
GE10	1927	3
GE11	2087	1
GE12	(See note below)	(See note below)

Note

The GE12 lead use assemblies are bounded by the GE11 LOCA analysis due to the following:

(1) GE12 design has a greater number of fuel rods, resulting in initial temperatures and stored energy lower than GE11 assemblies.
(2) GE12 fuel has a greater heat transfer area than GE11 fuel, which improves heat transfer characteristics during a LOCA.
(3) The GE12 bundles are specifically designed to have lower linear heat generation rates than the coresident GE11 fuel.

This letter contains no new commitments nor does it modify any existing commitments. Please contact Marcus Voth, Project Manager of Licensing at (763) 271-5116 if you require further information.

Michael F. Hammer
Site General Manager
Monticello Nuclear Generating Plant

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

Tammy Orr
Fuel Project Manager

Global Nuclear Fuel

A Joint Venture of GE, Toshiba, & Hitachi

Global Nuclear Fuel – Americas, LLC
Castle Hayne Road, Wilmington, NC 28401
(910) 675-5752, Fax (910) 675-5684
Tammy.Orr@gnf.com

IC.ZB.00.004

June 29, 2000
TGO:00-058

cc: C. A. Bonneau
J. E. Fawks
M. A. Higar
D. G. Horneck
H. H. Paustian
R. A. Rossi
D. G. Wegener

Mr. D. L. Orrock
Senior Fuel Buyer, Nuclear Energy Engineering
Northern States Power Company
414 Nicollet Mall - Ren Sq 10
Minneapolis, MN 55401-1927

Subject: **Summary of Changes and Errors in ECCS Evaluation Models**

Reference: 1. NRC Information Notice 9715, dated April 4, 1997.

Dear Don:

Enclosed is a copy of the annual letter recently sent to the NRC summarizing the impact of changes and errors in the methodology used by GE/GNF-A to demonstrate compliance with the Emergency Core Cooling System (ECCS) requirements of 10CFR50.46.

This letter is for information only.

These changes and errors have previously been reported to affected utilities. Plant specific errors (e.g., input errors) are not reported to the NRC in the annual letter. These errors are separately communicated to affected utilities and are now incorporated into the cycle specific Supplemental Reload Licensing Report.

The enclosed notification letter to the NRC does not satisfy the licensee reporting requirements per 10 CFR 50.46(a)(3)(ii), however, per Reference 1, licensees may reference this notification in their annual report.

Best regards,

T. G. Orr

GNF

Global Nuclear Fuel

Glen A. Watford
Manager, Nuclear Fuel Engineering

A Joint Venture of GE, Toshiba, & Hitachi

Global Nuclear Fuel – Americas, LLC
Castle Hayne Road, Wilmington, NC 28401
(910) 675-5446, Fax (910) 675-5764
Glen.Watford@gnf.com

June 30, 2000

FLN-2000-06

Document Control Desk
US Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention: J. L. Wermiel

Subject: **Summary of Changes and Errors in ECCS Evaluation Models**

Reference: Letter, G. A. Watford to the Document Control Desk (J. L. Wermiel), *Reporting of Changes and Errors in ECCS Evaluation Models*, dated June 30, 1999 (MFN–004–99).

The purpose of this letter is to summarize the impact of changes and errors in the methodology used by GE/GNF-A to demonstrate compliance with the Emergency Core Cooling System (ECCS) requirements of 10 CFR 50.46. This report covers the period from the last report (Reference) to the present. It is noted that Peak Cladding Temperature (PCT) variations resulting from system or fuel changes are not addressed in this letter. These should be treated, as appropriate, on a plant specific basis in accordance with other sections of 10CFR50.

A summary of the changes and errors is provided in the attached table. The table describes the approved methodology affected, the range of applicability of the change/error, a brief description of the change/error and the estimated impact.

All utilities using these evaluation models have been notified of these changes/errors to assist them in reporting, in accordance with 10CFR50.46 (a) (3) (ii).

If you have any questions, please call me at (910) 675-5446.

Sincerely,

Glen A. Watford, Manager
Nuclear Fuel Engineering

Summary of Changes and Errors in ECCS Evaluation Models
July 1999 through June 2000

Error/ Change	Approved Methodology	Applicability	Description	Impact
Error	NEDC-32950P, Compilation of Improvements to GENE's SAFER ECCS-LOCA Evaluation Model," January 2000.	BWR/6 plants	The reactor pressure vessel thermal response is simulated in the SAFER code as several heat slabs for which the one-dimensional radial conduction equation is solved (Reference). A logic error was discovered in an automated SAFER/GESTR basedeck generation procedure that calculated the heat transfer areas for the vessel heat slabs. As a result of this logic error, the heat transfer areas for the vessel heat slabs in the downcomer region were incorrectly specified for BWR/6 plants. This error affects the steam generation in the vessel during the reflooding stage of the event once the lower plenum fills and water spilling over from the jet pumps comes into contact with the vessel wall in the downcomer region.	0 to -45°F