Xcel Energy®

Monticello Nuclear Generating Plant
2807 W County Rd 75
Monticello, MN 55362

December 21, 2010

L-MT-10-072
10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

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

Subject: Monticello Extended Power Uprate: Updates to Docketed Information (TAC MD9990)

References:

1) Letter from T J O'Connor (NSPM), to Document Control Desk (NRC), "License Amendment Request: Extended Power Uprate (TAC MD9990)," L-MT-08-052, dated November 5, 2008. (ADAMS Accession No. ML083230111)
2) Letter from T J O'Connor (NSPM), to Document Control Desk (NRC), "Monticello Extended Power Uprate: Response to NRC Reactor Inspection Branch Request for Additional Information (RAI) dated March 20, 2009 (TAC No. MD9990)," L-MT-09-042, dated June 16, 2009. (ADAMS Accession No. ML091671787)
3) Letter from T J O'Connor (NSPM), to Document Control Desk (NRC), "Monticello Extended Power Uprate: Response to NRC Balance of Plant Review Branch (SBPB) Request for Additional Information (RAI) dated March 23, 2009 (TAC No. MD9990), "L-MT-09-046, dated June 12, 2009. (ADAMS Accession No. ML091670410)
4) Letter from T J O'Connor (NSPM), to Document Control Desk (NRC), "Monticello Extended Power Uprate: Replacement Steam Dryer Supplement (TAC MD9990)," L-MT-10-046, June 30, 2010. (ADAMS Accession No. ML102010462)

Pursuant to 10 CFR 50.90, the Northern States Power Company, a Minnesota corporation (NSPM), doing business as Xcel Energy, requested in Reference 1 an amendment to the Monticello Nuclear Generating Plant (MNGP) Renewed Operating

License (OL) and Technical Specifications (TS) to increase the maximum authorized power level from 1775 megawatts thermal (MWt) to 2004 MWt.

The purpose of this letter is to provide the NRC with updates to certain information previously provided to the NRC for the MNGP extended power uprate (EPU). The updates modify information provided in References 1, 2, 3 and 4. The following updates are being provided along with the enclosure where the updated information can be found:

Enclosure 1 - Revision of Commitment regarding Emergency Heat Load in the Spent Fuel Pool

Enclosure 2 - Revisions to pages from the MNGP EPU Safety Analysis Report

Enclosure 3 - Modification of the Reactor Internal Pressure Differential discussion

Enclosure 4 - Revision to the Technical Support Center Radiological Dose during a Postulated Accident

These updates provide clarifications and corrections to documents that form the licensing bases for the EPU application. The intent herein is to supplement the proposed amendment with corrected information to facilitate the NRC review.

The changes made in Enclosure 3 revise the no significant hazards consideration (NSHC) provided in Reference 1. No other changes impact the NSHC. NSPM has evaluated the proposed changes in accordance with the requirements of 10 CFR 50.91 against the standards of 10 CFR 50.92 and has determined the conclusions of the NSHC have not changed, that is, this request involves no significant hazards.

In accordance with 10 CFR 50.91(b), a copy of this application supplement, without enclosures is being provided to the designated Minnesota Official.

<u>Summary of Commitments</u>

This letter makes no new commitments. This letter makes a revision to an existing commitment made in Reference 3. See Enclosure 1 for details concerning the justification for the revised commitment. The revised commitment is as follows:

Prior to implementation of EPU, the USAR will be revised to indicate that the emergency heat load of 24.7 MBTU/hr occurs approximately 192 hours after shutdown.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on: December 21, 2010

Timothy J. O'Connor
Site Vice-President
Monticello Nuclear Generating Plant
Northern States Power Company-Minnesota

Enclosures (4)

cc: Administrator, Region III, USNRC (w/o enclosures)
Project Manager, Monticello Nuclear Generating Plant, USNRC (w/o enclosures)
Resident Inspector, Monticello Nuclear Generating Plant, USNRC (w/o enclosures)
Minnesota Department of Commerce (w/o enclosures)

ENCLOSURE 1

REVISION OF COMMITMENT REGARDING EMERGENCY HEAT LOAD
IN THE SPENT FUEL POOL

2 pages follow

REVISION OF COMMITMENT REGARDING EMERGENCY HEAT LOAD IN THE SPENT FUEL POOL

Background

In Reference E1-1, Northern States Power Company, a Minnesota corporation (NSPM), doing business as Xcel Energy, committed to the following:

> *Prior to RF025, the USAR will be revised to indicate that the emergency heat load of 24.7 MBTU/hr occurs approximately 168 hours after shutdown.*

This commitment was made in June 2009. Subsequently the Nuclear Regulatory Commission (NRC) delayed the review of the Monticello Nuclear Generating Plant (MNGP) Extended Power Uprate (EPU) application due to concerns regarding the application of containment accident pressure (CAP) credit (Reference E1-2). Refueling Outage (RFO) 25 is scheduled for the spring of 2011. This commitment should not be implemented prior to the NRC's approval of EPU for MNGP.

In addition, due to further evaluation, NSPM has determined that the value for emergency heat load requires revision.

Discussion

NSPM made the subject commitment prior to the NRC delaying the review of the MNGP EPU application (Reference E1-2). At the time that the commitment was made it appeared to NSPM that completion of this activity would be required prior to RFO25 since this was within the anticipated timeframe for NRC review and approval of the MNGP EPU application.

Now, based on the delay in the NRC review and to ensure that this EPU implementation activity is incorporated into the MNGP design bases after NRC approval, the commitment is being modified to require the USAR change to be developed prior to implementation of the EPU, that is, prior to increasing plant thermal power above the current licensed thermal power limit.

NSPM recognizes that the emergency heat load is increased for EPU conditions. This increase will be managed, as described in reference E1-3, by performing the required cycle-specific heat load calculation prior to moving fuel to the spent fuel pool. MNGP will continue to meet and maintain the emergency heat load within the heat removal capabilities of the Spent Fuel Pool Cooling and Residual Heat Removal systems using cycle specific calculations and procedural controls described in Reference E1-3.

Current calculations continue to use the emergency heat load value of 24.7 MBtu/hr as the limiting heat load for performing a full core discharge (FCD) to the spent fuel pool when the EPU is implemented for MNGP. However, the calculations demonstrate that after using EPU fuel, a further delay in performing a FCD to the spent fuel pool is required. The calculation determined that when EPU fuel is used, a FCD to the spent fuel pool should not be completed until after approximately 192 hours.

Conclusion

Based on the above the subject commitment should be revised. NSPM proposes to revise the commitment as follows:

> *Prior to implementation of EPU, the USAR will be revised to indicate that the emergency heat load of 24.7 MBTU/hr occurs approximately 192 hours after shutdown.*

References:

E1-1 Letter from T J O'Connor (NSPM), to Document Control Desk (NRC), "Monticello Extended Power Uprate: Response to NRC Balance of Plant Review Branch (SBPB) Request for Additional Information (RAI) dated March 23, 2009 (TAC No. MD9990), "L-MT-09-046, dated June 12, 2009, (ADAMS Accession No. ML091670410)

E1-2 Letter from Eric J Leeds (NRC) to T J O'Connor (NSPM), "Subject: Monticello Nuclear Generating Plant - Revised Schedule for Review of Extended Power Uprate Amendment Application (TAC No. MD9990)," dated October 1, 2009, (ADAMS Accession No. ML092600850)

E1-3 Letter from John T Conway to Document Control Desk (NRC), "Response to Request for Additional Thermal Hydraulic Information for a License Amendment Request for Contingent Installation of a Temporary Fuel Storage Rack in the Spent Fuel Pool (TAC No. MD0302)," L-MT-06-070, dated December 15, 2006, (ADAMS Accession No. ML063610073)

ENCLOSURE 2

REVISIONS TO PAGES FROM THE MONTICELLO NUCLEAR GENERATING PLANT EXTENDED POWER UPRATE SAFETY ANALYSIS REPORT

9 pages follow

Revisions to Pages from the EPU Safety Analysis Report

The table below identifies the changes to Monticello Nuclear Generating Plant (MNGP) documentation associated with the Extended Power Uprate (EPU) project. General Electric – Hitachi (GEH) supplied the revised pages to correct typographical errors and update computer code changes that have taken place during the course of evaluating the MNGP for EPU. Revised pages are replaced in their entirety. The replacement pages are provided after this table. Changes to the pages are indicated with a sidebar notation.

Table of Changes to MNGP EPU documentation

Remove - Document – Page Number	**Insert - Document – Page Number**
L-MT-08-052, Enclosure 5, NEDC-33322P, Revision 3, page 1-10	L-MT-08-052, Enclosure 5, NEDC-33322P, Revision 3 – Corrected Page, page 1-10
L-MT-08-052, Enclosure 5, NEDC-33322P, Revision 3, page 1-12	L-MT-08-052, Enclosure 5, NEDC-33322P, Revision 3 – Corrected Page, page 1-12
L-MT-08-052, Enclosure 5, NEDC-33322P, Revision 3, page 2-244	L-MT-08-052, Enclosure 5, NEDC-33322P, Revision 3 – Corrected Page, page 2-244
L-MT-08-052, Enclosure 5, NEDC-33322P, Revision 3, page A-9	L-MT-08-052, Enclosure 5, NEDC-33322P, Revision 3 – Corrected Page, page A-9
L-MT-08-052, Enclosure 7, NEDO-33322, Revision 3, page 1-10	L-MT-08-052, Enclosure 7, NEDO-33322, Revision 3 – Corrected Page, page 1-10
L-MT-08-052, Enclosure 7, NEDO-33322, Revision 3, page 1-12	L-MT-08-052, Enclosure 7, NEDO-33322, Revision 3 – Corrected Page, page 1-12
L-MT-08-052, Enclosure 7, NEDO-33322, Revision 3, page 2-244	L-MT-08-052, Enclosure 7, NEDO-33322, Revision 3 – Corrected Page, page 2-244
L-MT-08-052, Enclosure 7, NEDO-33322, Revision 3, page A-9	L-MT-08-052, Enclosure 7, NEDO-33322, Revision 3 – Corrected Page, page A-9

Each of the NEDC-33322P corrected pages is marked, "Proprietary Information." However, GEH has indicated to NSPM that the information provided is not of a proprietary nature. Therefore, this information is considered non-proprietary and may be released to the public. GEH is not requesting this information be withheld from public disclosure. No affidavit for withholding this information is required.

Table 1-1 Computer Codes Used For EPU

Task	Computer Code*	Version or Revision	NRC Approved	Comments
Nominal Reactor Heat Balance	ISCOR	09	(3)	NEDE-24011P Rev. 0 SER
Reactor Core and Fuel Performance	TGBLA PANACEA ISCOR	06 11 09	Y(2) Y(2) (3)	NEDE-30130-P-A NEDE-30130-P-A NEDE-24011P Rev. 0 SER
Thermal Hydraulic Stability	ODYSY TRACG OPRM	05 04 01	Y N(16) Y	NEDC-32992P-A, Class III, July 2001 NEDO-32465-A, Class I, August 1996
RPV Fluence	TGBLA DORTG	06 01	Y(2) Y	 See notes 13 and 14
Reactor Internal Pressure Differences	ISCOR LAMB TRACG	09 07 02	(3) (4) (15)	NEDE-24011P Rev. 0 SER NEDE-20566-P-A NEDE-32176P, Rev. 2, Dec. 1999 NEDC-32177P, Rev. 2, Jan 2000 NRC TAC No M90270, Sep 1994
Transient Analysis	PANACEA ISCOR ODYN SAFER	11 09 10 04	Y (3) Y (6)	NEDE-30130-P-A (5) NEDE-24011P Rev. 0 SER NEDO-24154-A NEDC-32424P-A, NEDC-32523P-A, (9), (10) (11)
Anticipated Transient Without Scram	ODYN STEMP PANACEA	10 04 11	Y (7) Y	NEDE-24154P-A Supp. 1, Vol. 4 NEDE-30130-P-A
Containment System Response	SHEX M3CPT LAMB	06 05 08	Y Y (4)	(8) NEDO-10320, Apr. 1971 NEDE-20566-P-A September 1986
Appendix R Fire Protection	GESTR SAFER SHEX	08 04 06	(6) (6) Y	NEDE-23785-1-PA, Rev. 1 (9) (10) (11) (8)
Reactor Recirculation System	BILBO	04V	NA	(1) NEDE-23504, February 1977
ECCS-LOCA	LAMB GESTR SAFER ISCOR TASC	08 08 04 09 03A	Y Y Y (3) Y	NEDO-20566A NEDE-23785-1-PA, Rev. 1 (6) (9) (10) (11) NEDE-24011P Rev. 0 SER NEDC-32084P (12)
Station Blackout (SBO)	SHEX	06	Y	(8)
Fission Product Inventory	ORIGEN	2.1	N	Isotope Generation and Depletion Code
Plant Life	CHECWORKS™	2.1	N	Industry Standard

approved LTRs for power uprate: "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate," NEDC-32424P-A, February 1999 and "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate," NEDC-32523P-A, February 2000. The Appendix R events are similar to the loss of FW and small break LOCA events.

(7) The STEMP code uses fundamental mass and energy conservation laws to calculate the suppression pool heatup. The use of STEMP was noted in NEDE-24222, "Assessment of BWR Mitigation of ATWS, Volume I & II (NUREG-0460 Alternate No. 3) December 1, 1979." The code has been used in ATWS applications since that time. There is no formal NRC review and approval of STEMP or the ATWS topical report.

(8) The NRC approved the application of the methodology in the SHEX code to containment response applications in the CLTR, (Reference 1, Section 4.1). The NRC approval of SHEX for containment analysis applications at Monticello is described in USNRC, Issuance of Amendment responding to Monticello Nuclear Generating Plant, License Amendment Request dated June 2, 2004, Revised Analysis of Long-Term Containment Response and Overpressure Required for Adequate NPSH for Low Pressure ECCS Pumps, (TAC No. MB7185), Amendment 139 to DPR-22.

(9) Letter, J.F. Klapproth (GEH) to USNRC, Transmittal of GE Proprietary Report NEDC-32950P "Compilation of Improvements to GENE's SAFER ECCS-LOCA Evaluation Model," dated January 2000 by letter dated January 27, 2000.

(10) Letter, S.A. Richards (NRC) to J.F. Klapproth, "General Electric Nuclear Energy (GENE) Topical Reports GENE (NEDC)-32950P and GENE (NEDC)-32084P Acceptability Review," May 24, 2000.

(11) "SAFER Model for Evaluation of Loss-of-Coolant Accidents for Jet Pump and Non-Jet Pump Plants," NEDE-30996P-A, General Electric Company, October 1987.

(12) The NRC approved the TASC-03A code by letter from S. A. Richards, NRC, to J. F. Klapproth, GE Nuclear Energy, Subject: "Review of NEDC-32084P, TASC-03A, A Computer Code for Transient Analysis of a Single Fuel Channel," TAC NO. MB0564, March 13, 2002. The acceptance version has not yet been published.

(13) CCC-543, "TORT-DORT Two-and Three-Dimensional Discrete Ordinates Transport Version 2.8.14," Radiation Shielding Information Center (RSIC), January 1994.

(14) Letter, H. N. Berkow (USNRC) to G. B. Stramback (GEH), "Final Safety Evaluation Regarding Removal of Methodology Limitations for NEDC-32983P-A, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations (TAC No. MC3788)," November 17, 2005.

(15) NRC has reviewed and accepted the TRACG application for the flow-induced loads on the core shroud as stated in NRC SER TAC No. M90270.

(16) TRACG02 has been approved in NEDO-32465-A by the US NRC for the stability DIVOM analysis. The CLTP stability analysis is based on TRACG04, which has been shown to provide essentially the same or more conservative results in DIVOM applications as the previous version, TRACG02.

The Thermal and Hydraulic Design is described in Monticello USAR Section 3.2, "Thermal and Hydraulic Characteristics." Power oscillations are addressed in USAR Section 14.6, "Plant Stability Analysis."

Technical Evaluation

Section 3.2 of ELTR2 documents interim corrective actions and four long-term stability options. Monticello has adopted Option III (Reference 29). Option III evaluations are core reload dependent and are performed for each reload fuel cycle. The Monticello Option III hardware will be installed and connected to the Reactor Protection System. In the event that the OPRM system is declared inoperable, Monticello will operate under the BWROG Guidelines for Backup Stability Protection (BSP) as described in Reference 31. When the EPU is implemented, cycle specific setpoints will be determined and documented in the same Supplemental Reload Licensing Report (SRLR).

2.8.3.1 Option III

The Option III solution combines closely spaced LPRM detectors into "cells" to effectively detect either core-wide or regional (local) modes of reactor instability. These cells are termed Oscillation Power Range Monitor (OPRM) cells and are configured to provide local area coverage with multiple channels. Plants implementing Option III have hardware to combine the LPRM signals and to evaluate the cell signals with instability detection algorithms. The Period Based Detection Algorithm (PBDA) is the only algorithm credited in the Option III licensing basis. Two defense-in-depth algorithms, referred to as the Amplitude Based Algorithm (ABA) and the Growth Rate Based Algorithm (GRBA), offer a high degree of assurance that fuel failure will not occur as a consequence of stability related oscillations.

The Option III trip is armed only when plant operation is within the Option III trip-enabled region. The Option III trip-enabled region is generically defined as the region on the power/flow map with power ≥ 30% of OLTP and core flow ≤ 60% of rated core flow. For EPU, the Option III trip-enabled region is rescaled to maintain the same absolute power/flow region boundaries. Because the rated core flow is not changed, the 60% core flow boundary is not rescaled. The 30% of OLTP boundary changes by the following equation:

EPU Region Boundary = 30% OLTP * (100% ÷ EPU (% OLTP))

Thus, for a 120% of OLTP EPU:

EPU Region Boundary = 30% OLTP * (100% ÷ 120%) = 25% EPU

The Monticello OPRM trip-enabled region is shown in Figure 2.8-19. The Backup Stability Protection (BSP) evaluation described in Section 2.8.3.2 shows that the generic Option III Trip Enabled Region is adequate. The adequacy of the OPRM armed Region will be confirmed for each reload.

Number	Title	Limitation Description	Disposition
19	Void-Quality Correlation 1	For applications involving PANCEA/ODYN/ISCOR/TASC for operation at EPU and MELLLA+, an additional 0.01 will be added to the OLMCPR, until such time that GE expands the experimental database supporting the Findlay-Dix void-quality correlation to demonstrate the accuracy and performance of the void-quality correlation based on experimental data representative of the current fuel designs and operating conditions during steady-state, transient, and accident conditions.	The Monticello reload core analysis will include a 0.01 adder to the OLMCPR consistent with Reference A-5, pending GEH's resolution of the void quality correlation.
20	Void-Quality Correlation 2	The NRC staff is currently reviewing Supplement 3 to NEDE-32906P, "Migration to TRACG04/PANAC11 from TRACG02/PANAC10," dated May 2006 (Reference A-4). The adequacy of the TRACG interfacial shear model qualification for application to EPU and MELLLA+ will be addressed under this review. Any conclusions specified in the NRC staff SE approving Supplement 3 to LTR NEDC-32906P (Reference A-4) will be applicable as approved.	The Limitation is not applicable to the current Monticello EPU license application because the application is not based on Supplement 3 to NEDC-32906P.
21	Mixed Core Method 1	Plants implementing EPU or MELLLA+ with mixed fuel vendor cores will provide plant-specific justification for extension of GE's analytical methods or codes. The content of the plant-specific application will cover the topics addressed in this SE as well as subjects relevant to application of GE's methods to legacy fuel. Alternatively, GE may supplement or revise LTR NEDC-33173P (Reference A-1) for mixed core application.	This limitation is not applicable to Monticello because the EPU core will consist of GE14 fuel exclusively and is not a mixed vendor core.

Table 1-1 Computer Codes Used For EPU

Task	Computer Code*	Version or Revision	NRC Approved	Comments
Nominal Reactor Heat Balance	ISCOR	09	(3)	NEDE-24011P Rev. 0 SER
Reactor Core and Fuel Performance	TGBLA PANACEA ISCOR	06 11 09	Y(2) Y(2) (3)	NEDE-30130-P-A NEDE-30130-P-A NEDE-24011P Rev. 0 SER
Thermal Hydraulic Stability	ODYSY TRACG OPRM	05 04 01	Y N(16) Y	NEDC-32992P-A, Class III, July 2001 NEDO-32465-A, Class I, August 1996
RPV Fluence	TGBLA DORTG	06 01	Y(2) Y	 See notes 13 and 14
Reactor Internal Pressure Differences	ISCOR LAMB TRACG	09 07 02	(3) (4) (15)	NEDE-24011P Rev. 0 SER NEDE-20566-P-A NEDE-32176P, Rev. 2, Dec. 1999 NEDC-32177P, Rev. 2, Jan 2000 NRC TAC No M90270, Sep 1994
Transient Analysis	PANACEA ISCOR ODYN SAFER	11 09 10 04	Y (3) Y (6)	NEDE-30130-P-A (5) NEDE-24011P Rev. 0 SER NEDO-24154-A NEDC-32424P-A, NEDC-32523P-A, (9), (10) (11)
Anticipated Transient Without Scram	ODYN STEMP PANACEA	10 04 11	Y (7) Y	NEDE-24154P-A Supp. 1, Vol. 4 NEDE-30130-P-A
Containment System Response	SHEX M3CPT LAMB	06 05 08	Y Y (4)	(8) NEDO-10320, Apr. 1971 NEDE-20566-P-A September 1986
Appendix R Fire Protection	GESTR SAFER SHEX	08 04 06	(6) (6) Y	NEDE-23785-1-PA, Rev. 1 (9) (10) (11) (8)
Reactor Recirculation System	BILBO	04V	NA	(1) NEDE-23504, February 1977
ECCS-LOCA	LAMB GESTR SAFER ISCOR TASC	08 08 04 09 03A	Y Y Y (3) Y	NEDO-20566A NEDE-23785-1-PA, Rev. 1 (6) (9) (10) (11) NEDE-24011P Rev. 0 SER NEDC-32084P (12)
Station Blackout (SBO)	SHEX	06	Y	(8)
Fission Product Inventory	ORIGEN	2.1	N	Isotope Generation and Depletion Code
Plant Life	CHECWORKS™	2.1	N	Industry Standard

approved LTRs for power uprate: "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate," NEDC-32424P-A, February 1999 and "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate," NEDC-32523P-A, February 2000. The Appendix R events are similar to the loss of FW and small break LOCA events.

(7) The STEMP code uses fundamental mass and energy conservation laws to calculate the suppression pool heatup. The use of STEMP was noted in NEDE-24222, "Assessment of BWR Mitigation of ATWS, Volume I & II (NUREG-0460 Alternate No. 3) December 1, 1979." The code has been used in ATWS applications since that time. There is no formal NRC review and approval of STEMP or the ATWS topical report.

(8) The NRC approved the application of the methodology in the SHEX code to containment response applications in the CLTR, (Reference 1, Section 4.1). The NRC approval of SHEX for containment analysis applications at Monticello is described in USNRC, Issuance of Amendment responding to Monticello Nuclear Generating Plant, License Amendment Request dated June 2, 2004, Revised Analysis of Long-Term Containment Response and Overpressure Required for Adequate NPSH for Low Pressure ECCS Pumps, (TAC No. MB7185), Amendment 139 to DPR-22.

(9) Letter, J.F. Klapproth (GEH) to USNRC, Transmittal of GE Proprietary Report NEDC-32950P "Compilation of Improvements to GENE's SAFER ECCS-LOCA Evaluation Model," dated January 2000 by letter dated January 27, 2000.

(10) Letter, S.A. Richards (NRC) to J.F. Klapproth, "General Electric Nuclear Energy (GENE) Topical Reports GENE (NEDC)-32950P and GENE (NEDC)-32084P Acceptability Review," May 24, 2000.

(11) "SAFER Model for Evaluation of Loss-of-Coolant Accidents for Jet Pump and Non-Jet Pump Plants," NEDE-30996P-A, General Electric Company, October 1987.

(12) The NRC approved the TASC-03A code by letter from S. A. Richards, NRC, to J. F. Klapproth, GE Nuclear Energy, Subject: "Review of NEDC-32084P, TASC-03A, A Computer Code for Transient Analysis of a Single Fuel Channel," TAC NO. MB0564, March 13, 2002. The acceptance version has not yet been published.

(13) CCC-543, "TORT-DORT Two-and Three-Dimensional Discrete Ordinates Transport Version 2.8.14," Radiation Shielding Information Center (RSIC), January 1994.

(14) Letter, H. N. Berkow (USNRC) to G. B. Stramback (GEH), "Final Safety Evaluation Regarding Removal of Methodology Limitations for NEDC-32983P-A, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations (TAC No. MC3788)," November 17, 2005.

(15) NRC has reviewed and accepted the TRACG application for the flow-induced loads on the core shroud as stated in NRC SER TAC No. M90270.

(16) TRACG02 has been approved in NEDO-32465-A by the US NRC for the stability DIVOM analysis. The CLTP stability analysis is based on TRACG04, which has been shown to provide essentially the same or more conservative results in DIVOM applications as the previous version, TRACG02.

The Thermal and Hydraulic Design is described in Monticello USAR Section 3.2, "Thermal and Hydraulic Characteristics." Power oscillations are addressed in USAR Section 14.6, "Plant Stability Analysis."

Technical Evaluation

Section 3.2 of ELTR2 documents interim corrective actions and four long-term stability options. Monticello has adopted Option III (Reference 29). Option III evaluations are core reload dependent and are performed for each reload fuel cycle. The Monticello Option III hardware will be installed and connected to the Reactor Protection System. In the event that the OPRM system is declared inoperable, Monticello will operate under the BWROG Guidelines for Backup Stability Protection (BSP) as described in Reference 31. When the EPU is implemented, cycle specific setpoints will be determined and documented in the same Supplemental Reload Licensing Report (SRLR).

2.8.3.1 Option III

The Option III solution combines closely spaced LPRM detectors into "cells" to effectively detect either core-wide or regional (local) modes of reactor instability. These cells are termed Oscillation Power Range Monitor (OPRM) cells and are configured to provide local area coverage with multiple channels. Plants implementing Option III have hardware to combine the LPRM signals and to evaluate the cell signals with instability detection algorithms. The Period Based Detection Algorithm (PBDA) is the only algorithm credited in the Option III licensing basis. Two defense-in-depth algorithms, referred to as the Amplitude Based Algorithm (ABA) and the Growth Rate Based Algorithm (GRBA), offer a high degree of assurance that fuel failure will not occur as a consequence of stability related oscillations.

The Option III trip is armed only when plant operation is within the Option III trip-enabled region. The Option III trip-enabled region is generically defined as the region on the power/flow map with power $\geq$ 30% of OLTP and core flow $\leq$ 60% of rated core flow. For EPU, the Option III trip-enabled region is rescaled to maintain the same absolute power/flow region boundaries. Because the rated core flow is not changed, the 60% core flow boundary is not rescaled. The 30% of OLTP boundary changes by the following equation:

EPU Region Boundary = 30% OLTP * (100% ÷ EPU (% OLTP))

Thus, for a 120% of OLTP EPU:

EPU Region Boundary = 30% OLTP * (100% ÷ 120%) = 25% EPU

The Monticello OPRM trip-enabled region is shown in Figure 2.8-19. The Backup Stability Protection (BSP) evaluation described in Section 2.8.3.2 shows that the generic Option III Trip Enabled Region is adequate. The adequacy of the OPRM armed Region will be confirmed for each reload.

Number	Title	Limitation Description	Disposition
19	Void-Quality Correlation 1	For applications involving PANCEA/ODYN/ISCOR/TASC for operation at EPU and MELLLA+, an additional 0.01 will be added to the OLMCPR, until such time that GE expands the experimental database supporting the Findlay-Dix void-quality correlation to demonstrate the accuracy and performance of the void-quality correlation based on experimental data representative of the current fuel designs and operating conditions during steady-state, transient, and accident conditions.	The Monticello reload core analysis will include a 0.01 adder to the OLMCPR consistent with Reference A-5, pending GEH's resolution of the void quality correlation.
20	Void-Quality Correlation 2	The NRC staff is currently reviewing Supplement 3 to NEDE-32906P, "Migration to TRACG04/PANAC11 from TRACG02/PANAC10," dated May 2006 (Reference A-4). The adequacy of the TRACG interfacial shear model qualification for application to EPU and MELLLA+ will be addressed under this review. Any conclusions specified in the NRC staff SE approving Supplement 3 to LTR NEDC-32906P (Reference A-4) will be applicable as approved.	The Limitation is not applicable to the current Monticello EPU license application because the application is not based on Supplement 3 to NEDC-32906P.
21	Mixed Core Method 1	Plants implementing EPU or MELLLA+ with mixed fuel vendor cores will provide plant-specific justification for extension of GE's analytical methods or codes. The content of the plant-specific application will cover the topics addressed in this SE as well as subjects relevant to application of GE's methods to legacy fuel. Alternatively, GE may supplement or revise LTR NEDC-33173P (Reference A-1) for mixed core application.	This limitation is not applicable to Monticello because the EPU core will consist of GE14 fuel exclusively and is not a mixed vendor core.

ENCLOSURE 3

MODIFICATION OF THE REACTOR INTERNAL PRESSURE DIFFERENTIAL DISCUSSION

20 pages follow

Modification of the Reactor Internal Pressure Differential Discussion

Background
In the Extended Power Uprate (EPU) resubmittal (Reference E3-1) Northern States Power Company, a Minnesota corporation (NSPM), doing business as Xcel Energy, called the reactor internal pressure differential (RIPD) value change, a change in method of evaluation. Reference E3-1, Enclosure 1, sections 2 and 4 described the current approach of determining RIPD as the General Electric (GE) air test data for BWR4-6 steam dryers. The proposed change in approach was to utilize a more realistic correlation for a BWR3 steam dryer. This was discussed further in Enclosure 1 and Enclosure 5 of Reference E3-1. This included discussion in the No Significant Hazards Consideration (NSHC) evaluation.

The replacement steam dryer (RSD) supplement for EPU (Reference E3-2) changed the RIPD value and approach again. Reference E3-2 changed the approach from GE BWR3 realistic correlation to a Westinghouse Electric Corp. LLC (WEC) (RSD provider) Computational Fluid Dynamics (CFD) approach. In the RSD supplement NSPM evaluated this and other changes to the EPU resubmittal.

The MNGP Updated Safety Analysis Report (USAR) states that the steam dryer design requirement is to preclude failure which would result in any part being discharged through the main steam line, in the event of a steam line break, which might prevent closure of a main steam line isolation valve. USAR section 3.6.3 states that an analytical method is used to determine reactor internal pressure, but the method for determining reactor internal pressure is not described. The description of the analysis provided in the USAR is only an identification of two inputs to the analysis; the bounding event (steam line severance) considered and a description of the nodes or major chambers (boundaries) in the reactor vessel where flow resistance (differential pressure) is determined (USAR figure 3.6-2). The RIPD values for the steam dryer and other reactor internals are provided in USAR Table 3.6-1.

Upon further analysis NSPM has determined that the RIPD approach described in Reference E3-2 does not meet the requirements of a method of evaluation change that requires prior NRC approval to implement.

Discussion
The RIPD approach should more properly be characterized as an input change not a methodology change. Below is the NSPM reasoning for re-characterizing the change from a method of evaluation change to an input change:

The Nuclear Regulatory Commission (NRC) Inspection Manual (Reference E3-3, section D.4.h) describes an evaluation method as follows:

> *"...this criterion* [10 CFR 50.59(c)(2)viii] *is specifically directed at changes to evaluation methods. The implementation guidance discusses the meaning of 'evaluation method,' and notes that the FSAR (as updated) (or documents incorporated by reference), must describe the method, and the change must affect this description, to require evaluation. Then, in accordance with criterion (viii), if the method is used in establishing the design bases, or in the safety analyses, prior NRC approval is required if there is a departure from the method as described in the FSAR (as updated)."*

Method of evaluation as described in the FSAR (as updated) is described in NEI 96-07, "Guidelines for 10 CFR 50.59 Implementation," (Reference E3-4). NEI 96-07 was recommended for adoption by the NRC staff in SECY-00-0203 and approved by the NRC Commissioners on November 14, 2000. NEI 96-07, section 3.10 states:

> *"Methods of evaluation means the calculational framework used for evaluating behavior or response of the facility or an SSC. ...*
>
> *Changes to such methods of evaluation require evaluation under 10 CFR 50.59(c)(2)(viii) only for evaluations used either in UFSAR safety analyses or in establishing the design bases, and only if the methods are described, outlined or summarized in the UFSAR. Methodology changes that are subject to 10 CFR 50.59 include changes to elements of existing methods described in the USAR and to changes that involve replacement of existing methods of evaluation with alternative methodologies."*

As described above, only the bounding event considered in the analysis and a description of the nodes or major chambers in the reactor vessel are described in the MNGP USAR, no calculational framework is described, outlined or summarized. Instead RIPD meets the definition of an "Input Parameter" as defined in NEI 96-07.

> *"Input parameters are those values derived directly from the physical characteristics of SSC or processes in the plant, including flow rates, temperatures, pressures, dimensions or measurements (e.g., volume, weight, size, etc.), and system response times. ...*
>
> *The principal intent of this definition is to distinguish methods of evaluation from evaluation input parameters. Changes to methods of evaluation described in the UFSAR ...are evaluated under criterion 10 CFR 50.59(c)(2)(viii), whereas changes to input parameters described in the FSAR are considered changes to the facility that would be evaluated under the other seven criteria of 10 CFR 50.59(c)(2) but not criterion (c)(2)(viii).*
>
> *If a methodology permits the licensee to establish the value of an input parameter on the basis of plant-specific considerations, then that value is an input to the methodology, not part of the methodology."*

In this case, the load on the steam dryer due to pressure differentials during accident conditions is considered. This is a plant specific value that is derived directly from the physical characteristics of the plant (i.e. a plant component). In addition, as described above, the MNGP USAR does not describe the method by which RIPDs were determined for the currently installed steam dryer. Therefore, NSPM has concluded that the change in RIPD for the replacement steam dryer does not require prior NRC approval to implement.

Conclusion

Based upon the discussion provided above, NSPM has determined that the RIPD approach described in Reference E3-2 does not meet the requirements of a "method of evaluation change" as described in the USAR and thus does not require prior NRC approval to implement. The RIPD approach should more properly be characterized as an input change not a methodology change.

To implement this conclusion changes are required to documentation previously provided for the MNGP EPU. The table below identifies the changes to MNGP documentation associated with the EPU project. No proprietary information was changed. Pages with changed information are indicated below.

Table of Changes to MNGP EPU documentation	
Remove - Document – Page Number	**Insert - Document – Page Number**
L-MT-08-052, Enclosure 1, page 1 of 36	L-MT-08-052, Enclosure 1, page 1 of 36 – Corrected Page
L-MT-08-052, Enclosure 1, page 4 of 36	L-MT-08-052, Enclosure 1, page 4 of 36 – Corrected Page
L-MT-08-052, Enclosure 1, pages 26 - 34 of 36	L-MT-08-052, Enclosure 1, pages 26 - 34 of 36 – Corrected Page
L-MT-10-046, Enclosure 1, pages 19 - 20 of 21	L-MT-10-046, Enclosure 1, pages 19 - 20 of 21 – Corrected Page
L-MT-10-046, Enclosure 1, Appendix 3 pages 3 - 5 of 25	L-MT-10-046, Enclosure 1, Appendix 3 pages 3 - 5 of 25 – Corrected Page

The listed replacement pages are provided immediately following this page. Changes to the pages are indicated with a sidebar notation.

References:

E3-1 Letter from T J O'Connor (NSPM), to Document Control Desk (NRC), "License Amendment Request: Extended Power Uprate (TAC MD9990)," L-MT-08-052, dated November 5, 2008. (ADAMS Accession No. ML083230111)

E3-2 Letter from T J O'Connor (NSPM), to Document Control Desk (NRC), "Monticello Extended Power Uprate: Replacement Steam Dryer Supplement (TAC MD9990)," L-MT-10-046, June 30, 2010. (ADAMS Accession No. ML102010462)

E3-3 NRC Inspection Manual, Part 9900, "10 CFR Guidance, 10 CFR 50.59, Changes, Test and Experiments," dated 3/13/01.

E3-4 NEI 96-07, Revision 1, "Guidelines for 10 CFR 50.59 Implementation," September 22, 2000.

ENCLOSURE 1

EVALUATION OF PROPOSED CHANGES

DESCRIPTION OF CHANGE
LICENSE AMENDMENT REQUEST
EXTENDED POWER UPRATE

1.0 SUMMARY DESCRIPTION

This evaluation supports a request to amend Renewed Operating License (OL) DPR-22 for Monticello Nuclear Generating Plant (MNGP). The proposed amendment includes supporting changes to the Operating License and Technical Specifications (TSs) necessary to implement the increased power level.

The proposed changes would change the TS definition of the term "Rated Thermal Power (RTP)." The proposed changes also revise the OL to increase the MNGP authorized steady state reactor core power level to 2,004 megawatts thermal (MWt), which is approximately 20 percent above the original rated thermal power (RTP) of 1,670 MWt, and approximately thirteen percent above the current RTP of 1,775 MWt.

Nuclear Regulatory Commission (NRC) approval of the requested increase in licensed thermal power level will allow Northern States Power Company, a Minnesota corporation (NSPM) to implement operational changes to generate and supply a higher steam flow to the turbine-generator. Higher steam flow is accomplished by increasing the reactor power along specified control rod and core flow lines of the power to flow map. This increase in steam flow will enable increasing the electrical output of the plant.

Enclosure 5 contains the power uprate safety analysis report (PUSAR) formatted in accordance with RS-001, "Review Standard for Extended Power Uprates." The PUSAR follows the guidelines contained in General Electric (GE) Licensing Topical Reports (LTR) NEDC-33004P-A, "Constant Pressure Power Uprate" (CLTR) (Reference 1). The PUSAR provides the technical bases for this request and contains an integrated summary of the results of the underlying safety analyses and evaluations performed specifically for the MNGP extended power uprate (EPU). The PUSAR analyses were completed to support an EPU to 2,004 MWt.

As part of the MNGP EPU request, NSPM is also proposing changes to the licensing basis for methodology used for containment analysis and credit for use of containment overpressure for net positive suction head (NPSH) for low pressure Emergency Core Cooling System (ECCS) pumps.

NSPM plans to implement the first phase of the extended power uprate following the spring 2009 refueling outage (RF024). Therefore, to support the NSPM changes to the current licensing basis for containment analysis. Approval is also requested for use of the SHEX code for containment analysis performed for station blackout.

Credit for Containment Overpressure for Low Head ECCS Pumps

The NRC, by its safety evaluation report (SER) dated June 2, 2004 (Reference 8), approved use of containment overpressure for the low head ECCS pumps for DBA-LOCA and Appendix R for MNGP. EPU operation increases the reactor decay heat, which increases the heat addition to the suppression pool following an event. As a result, both the suppression pool water temperature and containment pressure increase. Changes in vapor pressures corresponding to the increases in suppression pool temperatures affect the NPSH margins. NRC approval is requested for a change to the current licensing basis to credit containment pressure for the low head ECCS pumps to bound all design and licensing basis events.

Based on the above, Monticello is requesting continued approval of maximum overpressure credit of 20.36 psia to bound NPSH requirements for any analyzed design basis or license basis event.

For each event the analyses indicates sufficient containment overpressure is available to satisfy the NPSHR for the associated low pressure ECCS pumps using conservative methodology that maximized the suppression pool temperature and minimized the available containment pressure.

5.0 REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

In accordance with the requirements of 10 CFR 50.90, Northern States Power Company, a Minnesota corporation (NSPM) requests an amendment for an extended power uprate. NSPM has evaluated the proposed amendment in accordance with 10 CFR 50.91 against the standards in 10 CFR 50.92 and has determined that the operation of the facility in accordance with the proposed amendment presents no significant hazards. NSPM's evaluation against each of the criteria in 10 CFR 50.92 follows.

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

<u>Extended Power Uprate</u>

Response: No.

The probability (frequency of occurrence) of Design Basis Accidents occurring is not affected by the increased power level, because Monticello Nuclear Generating Plant (MNGP) continues to comply with the regulatory and design basis criteria established for plant equipment. A probabilistic risk assessment demonstrates that the calculated core damage frequencies do not significantly change due to Extended Power Uprate (EPU). Scram setpoints (equipment settings that initiate automatic plant shutdowns) are established such that there is no significant increase in scram frequency due to EPU. No new challenges to safety-related equipment result from EPU.

The changes in consequences of postulated accidents, which would occur from 102 percent of the EPU (rated thermal power) RTP compared to those previously evaluated, are acceptable. The results of EPU accident evaluations do not exceed the NRC approved acceptance limits. The spectrum of postulated

accidents and transients has been investigated, and are shown to meet the plant's currently licensed regulatory criteria. In the area of fuel and core design, for example, the Safety Limit Minimum Critical Power Ratio (SLMCPR) and other applicable Specified Acceptable Fuel Design Limits (SAFDL) are still met. Continued compliance with the SLMCPR and other SAFDLs will be confirmed on a cycle specific basis consistent with the criteria accepted by the NRC.

Challenges to the Reactor Coolant Pressure Boundary were evaluated at EPU conditions (pressure, temperature, flow, and radiation) and were found to meet their acceptance criteria for allowable stresses and overpressure margin.

Challenges to the containment have been evaluated, and the containment and its associated cooling systems continue to meet the current licensing basis. The increase in the calculated post LOCA suppression pool temperature above the currently assumed peak temperature was evaluated and determined to be acceptable. Radiological release events (accidents) have been evaluated, and have been shown to meet the guidelines of 10 CFR 50.67.

Containment Analysis Methods Change

Response: No.

The use of passive heat sinks, variable RHR heat exchanger capability K-value, and mechanistic heat and mass transfer from the suppression pool surface to the wetwell airspace after 30 seconds for the long term design basis accident loss of coolant accident (DBA-LOCA) containment analysis are not relevant to accident initiation, but rather, pertain to the method used to accurately evaluate postulated accidents. The use of these elements does not, in any way, alter existing fission product boundaries, and provides a conservative prediction of the containment response to DBA-LOCAs. Therefore, the containment analysis method change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Credit for Containment Overpressure for Low Head Emergency Core Cooling System (ECCS) Pumps

Response: No.

These changes update parameters used in the MNGP safety analyses and expand the range and scope of the analyses. This will result in a more realistic analysis of available containment overpressure under design basis accident conditions. The updated analyses affect only the evaluation of previously reviewed accidents. No plant structure, system, or component (SSC) is physically affected by the updated and expanded analyses. No method of operation of any plant SSC is affected. Therefore, there is no significant increase in the probability or consequence of a previously evaluated accident.

The analyses supporting the above evaluations were performed at the EPU power level of 2,004 MWt.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Extended Power Uprate

Response: No.

Equipment that could be affected by EPU has been evaluated. No new operating mode, safety-related equipment lineup, accident scenario, or equipment failure mode was identified. The full spectrum of accident considerations has been evaluated and no new or different kind of accident has been identified. EPU uses developed technology and applies it within capabilities of existing or modified plant safety related equipment in accordance with the regulatory criteria (including NRC approved codes, standards and methods). No new accidents or event precursors have been identified.

The MNGP TS require revision to implement EPU. The revisions have been assessed and it was determined that the proposed change will not introduce a different accident than that previously evaluated. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

Containment Analysis Methods Change

Response: No.

The use of passive heat sinks, variable RHR heat exchanger capability K-value, and mechanistic heat transfer from the suppression pool surface to the wetwell airspace after 30 seconds for the long term DBA-LOCA containment analysis are not relevant to accident initiation, but pertain to the method used to evaluate currently postulated accidents. The use of these analytical tools does not involve any physical changes to plant structures or systems, and does not create a new initiating event for the spectrum of events currently postulated. Further, they do not result in the need to postulate any new accident scenarios. Therefore, the containment analysis method change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Credit for Containment Overpressure for Low Head ECCS Pumps

Response: No.

The proposed change involves the updating and expansion in scope of the existing design bases analysis with respect to the available containment overpressure to cover additional events. No new failure mode or mechanisms have been created for any plant SSC important to safety nor has any new limiting single failure been identified as a result of the proposed analytical changes. Therefore, the change to containment overpressure credited for low pressure ECCS pumps does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The analyses supporting the above evaluations were performed at the EPU power level of 2,004 MWt.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Extended Power Uprate

Response: No.

The EPU affects only design and operational margins. Challenges to the fuel, reactor coolant pressure boundary, and containment were evaluated for EPU conditions. Fuel integrity is maintained by meeting existing design and regulatory limits. The calculated loads on affected structures, systems and components, including the reactor coolant pressure boundary, will remain within their design allowables for design basis event categories. No NRC acceptance criterion is exceeded. Because the MNGP configuration and responses to transients and postulated accidents do not result in exceeding the presently approved NRC acceptance limits, the proposed changes do not involve a significant reduction in a margin of safety.

Containment Analysis Methods Change

Response: No.

The use of passive heat sinks, variable RHR heat exchanger capability K-value, and mechanistic heat transfer from the suppression pool surface to the wetwell airspace after 30 seconds for the long term DBA-LOCA containment analysis are realistic phenomena and provide a conservative prediction of the plant response to DBA-LOCAs. The increase in pressure and temperature are relatively small and are within design limits. Therefore, the containment analysis methods change does not involve a significant reduction in the margin of safety.

Credit for Containment Overpressure for Low Head ECCS Pumps

Response: No.

The proposed changes revise containment response analytical methods and scope for containment pressure to assist in ECCS pump net positive suction head (NPSH). The changes are still based on conservative but more realistic analysis of available containment overpressure determined using analysis methods that minimize containment pressure and maximize suppression pool temperature. These changes do not constitute a significant reduction in the margin of safety.

The analyses supporting the above evaluations were performed at the EPU power level of 2,004 MWt.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the considerations above, the NSPM has determined that operation of the facility in accordance with the proposed change does not involve a significant hazards consideration as defined in 10 CFR 50.92(c), in that it does not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

5.2 Applicable Regulatory Requirements

5.2.1 Analysis

Extended Power Uprate

10 CFR 50.36 (d)(2)(ii) Criterion 2, requires that TS LCOs include process variables, design features, and operating restrictions that are initial conditions of design basis accident analysis. The Technical Specifications ensure that the MNGP system performance parameters are maintained within the values assumed in the safety analyses. The Technical Specification changes are supported by the safety analyses and continue to provide a level of protection comparable to the current Technical Specifications. Applicable regulatory requirements and significant safety evaluations performed in support of the proposed changes are described in Enclosure 5.

Containment Analysis Methods Change

The MNGP principal design criteria with respect to containment are specified in USAR section 1.2.4. The applicable criteria in this section are specified in USAR sections 1.2.4.a and 1.2.4.b.

USAR Section 1.2.4.a requires that a primary containment system be provided that is designed, fabricated and erected to accommodate, without failure, the pressures and temperatures resulting from or subsequent to the double-ended rupture, or equivalent failure of any coolant pipe within the primary containment. The evaluations described in Enclosure 5, Section 2.6 demonstrate that containment parameters stay within their design limits.

Section 1.2 of the Monticello USAR contains principal design criteria specific to MNGP. Section 1.2.4.b of the USAR states, "Provision is made both for the removal of energy from within the primary containment and/or such other measures as may be necessary to maintain integrity of the primary containment system as long as necessary following the various postulated design-basis loss-of-coolant accidents." The evaluations described in Enclosure 5, Section 2.6 demonstrate that containment parameters stay within their design limits.

Credit for Containment Overpressure for Low Head ECCS Pumps

Section 1.2 of the Monticello USAR contains principal design criteria specific to MNGP. Section 1.2.4.b of the USAR states, "Provision is made both for the removal of energy from within the primary containment and/or such other measures as may be necessary to maintain integrity of the primary containment system as long as necessary following the various postulated design-basis loss-of-coolant accidents."

Regulatory Guide (RG) 1.82, Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident, Revision 3 (Reference 11) is not part of MNGP's licensing basis. However its provisions may be useful as guidance. This RG recognizes that it may not be practicable to alter the design of an operating reactor. Therefore, some overpressure may be needed to assure adequate available NPSH. RG 1.82 indicates that containment accident pressure should be conservatively calculated and the amount of credit given for containment overpressure should be minimized.

The proposed credit for containment overpressure bounds analyzed design and licensing basis events. The containment response used for NPSH evaluations was calculated using MNGP specific inputs to maximize suppression pool temperature and minimize containment pressure for the DBA LOCA analysis. The containment responses used for NPSH evaluations for Special Events (such as ATWS, SBO, and Appendix R) used MNGP specific nominal inputs to provide realistic maximized suppression pool temperatures and corresponding realistic minimized wetwell pressures.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

Proposed Changes for Extended Power Uprate

The proposed TS changes required for implementation of EPU meet the requirements for an environmental review as set forth in 10 CFR 51.20, "Criteria for and Identification of Licensing and Regulatory Actions Requiring Environmental Impact Statements." The Environmental Assessment in Enclosure 4 concludes that, "Extended power uprate does not involve any significant impacts to the environment. There are no new significant environmental hazards in addition to those previously evaluated. The environmental impacts and adverse effects identified by the NRC Staff for MNGP operation at 1,670 MWt in the Summary and Conclusions Section of the Final Environmental Statement continue to bound plant operation at extended power uprate conditions. The proposed changes do not, individually or cumulatively, affect the human environment. There is no significant change in the types or amounts of plant effluents. Extended power uprate does not involve significant increases in individual or cumulative occupational radiation exposure." The evaluation described in the Environmental Assessment, Enclosure 4, supports increases in the licensed power level up to 2,004 MWt.

Other Proposed Changes

Containment Analysis Methods Change and Containment Overpressure for NPSH for Low Pressure ECCS Pumps

These proposed changes do not involve (i) a significant hazards consideration, (ii), a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, these proposed changes meet the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with these proposed changes.

normal plant operations as described in Reference 1, Enclosure 9, section 4.2 and table 2.

Data collected will be compared to the limit curves provided in Enclosure 7. If a level 1 acceptance criterion is exceeded, operations will place the unit in a previously acceptable plant condition. If this requires the plant to return to a lower power level then the plant will be placed in that lower power level condition until the level 1 criteria is re-evaluated and new limit curves are generated.

NSPM is making a new commitment to complete the RSD - PATP. The commitment is as follows:

> *As part of MNGP restart following installation of the replacement steam dryer, NSPM will implement the Power Ascension Test Plan found in Enclosure 1, Appendix 5 of this letter.*

The commitment is required to satisfy the requirements of Regulatory Guide 1.20. The RSD - PATP contains requirements from RG 1.20 concerning testing and reporting requirements for the RSD. See Enclosure 11 of this letter for details concerning this commitment. See Appendix 1 to this enclosure for further details on compliance with RG 1.20.

4.2.2 Data reduction

Data reduction and comparisons to design data will be transmitted to the NRC after appropriate plant management review. If new limit curves are generated, they will be included in the data package transmitted to the MNGP NRC PM. Power ascension will continue when operations is satisfied that all test conditions have been successfully met. This is documented in the RSD - PATP which is provided in Appendix 5 to this enclosure.

4.3 Evaluation of the No Significant Hazards Consideration

With this letter NSPM is essentially describing two changes to the EPU analyses provided in the original amendment request (Reference 1). These changes are:

Moisture Carryover (MCO) analysis results
Replacement Steam Dryer Power Ascension Testing Plan (RSD-PATP)

MCO

The MCO analysis value provided in section 4.1.2 of this enclosure is less than the MCO value previously provided value found in the analysis in Reference 1. Therefore, the bounding MCO value provided will still be used. In addition, the MCO changes in reference 1 did not require evaluation under the NSHC. Therefore, no changes to the NSHC are required.

RSD-PATP

Finally, the RSD-PATP documented in section 4.2 of this enclosure provides documentation of the revised testing required to support the RSD installation. The RSD-PATP provides assurance that the installed component has the analyzed margin of safety and confirms the results of the vibration analysis. The RSD-PATP does not require evaluation under the NSHC and therefore, no changes to the NSHC are required.

Letter No. ADAMS No.	Locations Evaluated	Applicable Contents/Issues	Required Actions or Resolution
L-MT-08-052, EPU LAR ML083230111	Enclosure 1, pg 1	Page 1 states: *"As part of the MNGP EPU request, NSPM is also proposing changes to the licensing basis for methodology used for containment analysis, credit for use of containment overpressure for net positive suction head (NPSH) for low pressure Emergency Core Cooling System (ECCS) pumps, and reactor internal pressure differentials for the steam dryer."*	This statement is modified in L-MT-10-072 to remove any reference to reactor internal pressure differentials (RIPDs). The change in determination of RIPD is not a method of evaluation change described in the USAR and therefore, is not required for evaluation under 10CFR50.91.
L-MT-08-052, EPU LAR ML083230111	Enclosure 1, pg 4	Page 4 states "*<u>Reactor Internal Pressure Differentials (RIPDs) for the Steam Dryer</u> The effects on reactor internal loads as a result of EPU were evaluated. The increase in core power generally results in increased RIPDs for reactor internals due to the higher core exit steam quality. The RIPDs for the steam dryer in the EPU analysis are reduced from those used in the current analyses. NRC approval is requested for this change since it is a change to the current licensing basis for analytical methods used for evaluation of the loads for the reactor internals. The change methodology for determining steam dryer RIPDs is described in Enclosure 5, Section 2.2.3."*	This text is deleted in its entirety as it is linked to changes that require evaluation under 10CFR50.91. The change in determination of RIPD is not a method of evaluation change described in the USAR and therefore, is not required for evaluation under 10CFR50.91. Subsequent statements in Enclosure 5, section 2.2.3 are still applicable. Section 2.2.3 was previously modified in L-MT-10-046. (see L-MT-10-046, Enclosure 1, Appendix 4 for previously changed pages)
L-MT-08-052, EPU LAR ML083230111	Enclosure 1, pg 26	Page 26 is part of the *Evaluation of Proposed Changes*, which states: *"<u>Reactor Internal Pressure Differentials for the Steam Dryer</u> The technical bases for the change in steam dryer RIPDs used in the reactor vessel internal load evaluation includes information proprietary to Westinghouse Electric Corporation (WEC) and are discussed in Enclosure 5, Section 2.2.3."*	This text is deleted in its entirety as it is linked to changes that require evaluation under 10CFR50.91. The change in determination of RIPD is not a method of evaluation change described in the USAR and therefore, is not required for evaluation under 10CFR50.91. This section was previously modified in L-MT-10-046. (see L-MT-10-046, Enclosure 1, Appendix 4 for previously changed pages)

Letter No. ADAMS No.	Locations Evaluated	Applicable Contents/Issues	Required Actions or Resolution
L-MT-08-052, EPU LAR ML083230111	Enclosure 1, pg 28	Page 28 is part of the *No Significant Hazards Consideration* (NSHC) evaluation for question 1, where it addresses *"Reactor Internal Pressure Differentials (RIPDs) for the Steam Dryer."*	The statements in question 1 of the NSHC regarding the evaluation of change in RIPD methodology are eliminated as the change in determination of RIPD is not a method of evaluation change described in the USAR and therefore, is not required for evaluation under 10CFR50.91. The conclusion of the NSHC section is unaffected by the change.
L-MT-08-052, EPU LAR ML083230111	Enclosure 1, pg 29	Page 29 is part of the NSHC evaluation for question 2, where it addresses *"Reactor Internal Pressure Differentials (RIPDs) for the Steam Dryer."*	The statements in question 2 of the NSHC regarding the evaluation of change in RIPD methodology are eliminated as the change in determination of RIPD is not a method of evaluation change described in the USAR and therefore, is not required for evaluation under 10CFR50.91. The conclusion of the NSHC section is unaffected by the change.
L-MT-08-052, EPU LAR ML083230111	Enclosure 1, pg 31	Page 31 is part of the NSHC evaluation for question 3, where it addresses *"Reactor Internal Pressure Differentials (RIPDs) for the Steam Dryer."*	The statements in question 3 of the NSHC regarding the evaluation of change in RIPD methodology are eliminated as the change in determination of RIPD is not a method of evaluation change described in the USAR and therefore, is not required for evaluation under 10CFR50.91. The conclusion of the NSHC section is unaffected by the change.

Letter No. ADAMS No.	Locations Evaluated	Applicable Contents/Issues	Required Actions or Resolution
L-MT-08-052, EPU LAR ML083230111	Enclosure 1, pg 33	Page 33 is part of the *Applicable Regulatory Requirements* evaluation where it addresses *"Reactor Internal Pressure Differentials (RIPDs) for the Steam Dryer"* which states: *"Section 1.2 of the Monticello USAR contains principal design criteria specific to Monticello. Section 1.2.1 .a of the USAR states, "The plant is designed, fabricated, erected, and operated to produce electrical power in a safe, reliable, and efficient manner and in accordance with applicable codes and regulations."* *Section 1.2.2.i of the USAR states, "The reactor core and associated systems are designed to accommodate plant operational transients or maneuvers which might be expected without compromising safety and without fuel damage."* *The change methodology for determining steam dryer RIPDs is described in Enclosure 5, Section 2.2.3. The evaluation indicates that the reactor internals and core supports will continue to meet the requirements of 10 CFR 50.55a and MNGP's current licensing basis following implementation of the proposed EPU."*	This text is deleted in its entirety as the Applicable Regulatory Requirements section is performed for changes that require evaluation under the NSHC section. The change in determination of RIPD is not a method of evaluation change described in the USAR and therefore, is not required for evaluation under NSHC (10CFR50.91). This section was previously modified in L-MT-10-046. (see L-MT-10-046, Enclosure 1, Appendix 4 for previously changed pages) The conclusion of the Applicable Regulatory Requirements section is unaffected by the change.
L-MT-08-052, EPU LAR ML083230111	Enclosure 1, pg 34	Page 34 is part of the *Environment Consideration*, which has a part labeled *"Containment Analysis Methods Change, Containment Overpressure for NPSH for Low Pressure ECCS Pumps, and Steam Dryer RIPDs"*	This title is modified in L-MT-10-072 to remove any reference to RIPD for the Steam Dryer. The RIPD was included in this section based on its inclusion in the NSHC. Since RIPD is being removed from the NSHC it is appropriate to remove from the Environmental Consideration section.
L-MT-08-052, EPU LAR ML083230111	Enclosure 2, all	TS markup for EPU conditions	L-MT-08-052, Enclosure 2 is not affected by the RSD. Therefore, no further actions are required.
L-MT-08-052, EPU LAR ML083230111	Enclosure 3, all	TS Bases markup for EPU conditions	L-MT-08-052, Enclosure 3 is not affected by the RSD. Therefore, no further actions are required.

ENCLOSURE 4

REVISION TO THE TECHNICAL SUPPORT CENTER RADIOLOGICAL DOSE DURING A POSTULATED ACCIDENT

7 pages follow

Revision to the Technical Support Center Radiological Dose during a Postulated Accident

Northern States Power Company, a Minnesota corporation (NSPM), doing business as Xcel Energy, erred in the reporting of values for dose levels associated with the Monticello Nuclear Generating Plant (MNGP) Technical Support Center (TSC) post accident. In References E4-1 and E4-2 NSPM discovered the following errors:

- Under Current Licensed Thermal Power (CLTP) conditions, the value reported in Reference E4-1, Enclosures 4, 5 and 7 for post-Loss of Coolant Accident (LOCA) TSC dose was under reported as 0.77 Rem. The actual dose should have been reported as 0.854 Rem.

- Under Extended Power Uprate (EPU) conditions, the value reported in Reference E4-1, Enclosures 4, 5 and 7 and in Reference E4-2, RAI No. 4 for post-LOCA TSC dose was under reported as 0.83 Rem. The actual dose should have been reported as 0.92 Rem.

In the first instance it is presumed that an administrative error resulted in the incorrect value being retrieved from the approved calculation for CLTP post-LOCA TSC dose and reported in Reference E4-1. This value had been reported correctly to the NRC previously in Reference E4-3. Discovery of this error has been entered into the MNGP corrective action program for disposition.

In the second instance NSPM recently completed a revision to the EPU calculations that determine dose levels associated with the MNGP TSC post accident. The results of the calculation indicate that the TSC dose increased by approximately 85 mrem. The total dose becomes 916 mrem or 0.92 Rem.

In Reference E4-1 NSPM reported that the MNGP TSC dose assuming EPU conditions and a LOCA would be 0.83 Rem. This value was reported in Enclosures 4, 5 and 7 of Reference E4-1. In addition, TSC dose was reported in Reference E4-2, RAI No. 4, in terms of the contribution of direct radiation exposure from plant systems containing the accident source term (called shine). The TSC dose was reported as a shine contribution of 0.0939 Rem of the total 0.83 Rem. The shine portion of the TSC dose has not changed, only the total TSC dose has changed.

The revision in calculation and subsequent change in dose resulted from a non-limiting case being used in the calculation of TSC dose. MNGP surveillance testing provides an acceptance criterion of 715 - 1100 cfm for the TSC emergency ventilation fan. A review of surveillance results indicated that the system does not operate above about 870 cfm. However, the calculation performed for the MNGP EPU assumed flow rates of 900 - 1100 scfm which was incorrectly considered to include the bounding case.

Subsequently, to properly bound anticipated operating conditions, the TSC dose analysis was reperformed using a flow rate case of 710 scfm.

The dose for the 710 scfm case is approximately 85 mrem higher than the dose previously calculated. Total TSC dose (internal plus direct dose) remains less than 1 Rem, well within the 5 Rem accident limit for TSC occupants as required by 10 CFR 50.67 and NUREG-0696.

Conclusions

The errors require revision to the EPU documentation. The table below identifies changes to MNGP documentation as a result of the incorrect reporting of TSC dose. Revised pages are replaced in their entirety. The replacement pages are provided after this table. Changes to the pages are indicated with a sidebar notation.

Table of Changes to MNGP EPU documentation	
Remove - Document – Page Number	**Insert - Document – Page Number**
L-MT-08-052, Enclosure 4, pg 61 of 69	L-MT-08-052, Enclosure 4, page 61 of 69 – Corrected Page
L-MT-08-052, Enclosure 5, NEDC-33322P, Revision 3, page 2-340	L-MT-08-052, Enclosure 5, NEDC-33322P, Revision 3 – Corrected Page, page 2-340
L-MT-08-052, Enclosure 7, NEDO-33322, Revision 3, page 2-340	L-MT-08-052, Enclosure 7, NEDO-33322, Revision 3 – Corrected Page, page 2-340
L-MT-09-042, Enclosure 1, page 11 of 11	L-MT-09-042, Enclosure 1, page 11 of 11 – Corrected Page

The NEDC-33322P corrected page is marked, "Proprietary Information." However, the information provided on this page is not of a proprietary nature. Therefore, this information is considered non-proprietary and may be released to the public. GEH is not requesting this information be withheld from public disclosure. No affidavit for withholding this information is required.

References

E4-1 Letter from T J O'Connor (NSPM), to Document Control Desk (NRC), "License Amendment Request: Extended Power Uprate (TAC MD9990)," L-MT-08-052, dated November 5, 2008. (ADAMS Accession No. ML083230111)

E4-2 Letter from T J O'Connor (NSPM), to Document Control Desk (NRC), "Monticello Extended Power Uprate: Response to NRC Reactor Inspection Branch Request for Additional Information (RAI) dated March 20, 2009 (TAC No. MD9990)," L-MT-09-042, dated June 16, 2009. (ADAMS Accession No. ML091671787)

E4-3 Letter from B J Sawatzke (NSPM), to Document Control Desk (NRC), "Commitment Closure: Technical Support Center Dose Assessment in Conjunction with the Full-Scope Alternative Source Term License Amendment," L-MT-08-021, dated April 18, 2008. (ADAMS Accession No. ML081090593)

Table 7.3.8-1 Accident Analysis Results

Parameter	1880 MWth (Current Licensed Thermal Power Level Design Assumption Value)	2004 MWth (Extended Power Uprate Value)	Regulatory Limit (10 CFR 50.67) & (10 CFR 50, App. A, GDC 19)
Post-LOCA Accident Dose:			
EAB	1.31 Rem TEDE	1.46 Rem TEDE	25 Rem TEDE
LPZ	1.72 Rem TEDE	1.99 Rem TEDE	25 Rem TEDE
CR Operator	3.40 Rem TEDE	3.80 Rem TEDE	5 Rem TEDE
TSC Operator	0.854 Rem TEDE	0.92 Rem TEDE	5 Rem TEDE
FHA Accident Dose:			
EAB	1.61 Rem TEDE	1.74 Rem TEDE	6.25 Rem TEDE
LPZ	0.31 Rem TEDE	0.34 Rem TEDE	6.25 Rem TEDE
CR Operator	4.29 Rem TEDE	4.67 Rem TEDE	5 Rem TEDE
CRDA Accident Dose:			
EAB	1.73 Rem TEDE	1.96 Rem TEDE	6.25 Rem TEDE
LPZ	0.79 Rem TEDE	0.90 Rem TEDE	6.25 Rem TEDE
CR Operator	1.70 Rem TEDE	1.86 Rem TEDE	5 Rem TEDE
MSLBA Accident Dose: Pre-Incident Iodine Spike			
EAB	1.05 Rem TEDE	1.05 Rem TEDE	25 Rem TEDE
LPZ	0.20 Rem TEDE	0.20 Rem TEDE	25 Rem TEDE
CR Operator	3.25 Rem TEDE	3.25 Rem TEDE	5 Rem TEDE
MSLBA Accident Dose: Equilibrium Iodine Conc.			
EAB	0.11 Rem TEDE	0.11 Rem TEDE	2.5 Rem TEDE
LPZ	0.02 Rem TEDE	0.02 Rem TEDE	2.5 Rem TEDE
CR Operator	0.33 Rem TEDE	0.33 Rem TEDE	5 Rem TEDE

Table 2.9-1 LOCA Radiological Consequences

	TEDE Dose (REM)			
	Receptor Location			
	CR	EAB	LPZ	TSC
Calculated Dose CLTP[1]	3.40	1.31	1.72	0.854
Calculated Dose EPU[2]	3.80	1.46	1.99	0.92
Allowable TEDE Limit[3]	5	25	25	5

Table 2.9-2 FHA Radiological Consequences

	TEDE Dose (REM)		
	Receptor Location		
	CR	EAB	LPZ
Calculated Dose CLTP[1]	4.29	1.61	0.31
Calculated Dose EPU[2]	4.67	1.74	0.34
Allowable TEDE Limit[3]	5	6.3	6.3

Table 2.9-3 CRDA Radiological Consequences

	TEDE Dose (REM)		
	Receptor Location		
	CR	EAB	LPZ
Calculated Dose CLTP[1]	1.70	1.73	0.79
Calculated Dose EPU[2]	1.86	1.96	0.90
Allowable TEDE Limit[3]	5	6.3	6.3

Tables 2.9-1, 2.9-2 and 2.9-3 notes:

1. CLTP Power Level Assumption = 1880 MWt x 1.02 = 1918 MWt
2. EPU power Level Assumption = 2004 MWt x 1.02 = 2044 MWt
3. RG 1.183 Table 6

Table 2.9-1 LOCA Radiological Consequences

	TEDE Dose (REM)			
	Receptor Location			
	CR	EAB	LPZ	TSC
Calculated Dose CLTP[1]	3.40	1.31	1.72	0.854
Calculated Dose EPU[2]	3.80	1.46	1.99	0.92
Allowable TEDE Limit[3]	5	25	25	5

Table 2.9-2 FHA Radiological Consequences

	TEDE Dose (REM)		
	Receptor Location		
	CR	EAB	LPZ
Calculated Dose CLTP[1]	4.29	1.61	0.31
Calculated Dose EPU[2]	4.67	1.74	0.34
Allowable TEDE Limit[3]	5	6.3	6.3

Table 2.9-3 CRDA Radiological Consequences

	TEDE Dose (REM)		
	Receptor Location		
	CR	EAB	LPZ
Calculated Dose CLTP[1]	1.70	1.73	0.79
Calculated Dose EPU[2]	1.86	1.96	0.90
Allowable TEDE Limit[3]	5	6.3	6.3

Tables 2.9-1, 2.9-2 and 2.9-3 notes:

1. CLTP Power Level Assumption = 1880 MWt x 1.02 = 1918 MWt
2. EPU power Level Assumption = 2004 MWt x 1.02 = 2044 MWt
3. RG 1.183 Table 6

NRC RAI No. 4

The Safety Analysis Report for the Monticello Constant Power Uprate, dated October 2008, on page 2-340, within Table 2.9-1, indicates the dose consequences in the Control Room and the Technical Support Center, from a design-basis loss-of-coolant accident under EPU conditions, as 3.80 rem and 0.83 rem, respectively. Verify that these results include direct radiation exposure from plant systems containing the accident source term, consistent with the assumptions in NUREG-0737, item II.B.2. If not, demonstrate that the direct radiation dose rates for these two vital areas meet the GDC-19 dose criteria, as specified in NUREG-0737, item II.B.2.

NSPM RESPONSE

The Control Room and Technical Support Center (TSC) total calculated doses include a component due to direct shine dose from plant systems and the reactor building as required by NUREG-0737 Item II.B.2. The shine contribution for the Control Room is 0.771 Rem of the total 3.8 Rem and the TSC is 0.0939 Rem of the total 0.92 Rem.