NSP

Northern States Power Company

Monticello Nuclear Generating Plant
2807 West Hwy 75
Monticello, Minnesota 55362-9637

March 22, 2000

10 CFR Part 50
Section 50.73

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

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

**LER 2000-004**

**Re-analysis of the High Energy Line Break Resulted in Potential Loss of Division I 4kV Switchgear**

The Licensee Event Report for this occurrence is attached. This report contains no new NRC commitments.

Please contact Mohamad Marashi at (763) 295-1425 if you require further information.

Byron Day
Plant Manager
Monticello Nuclear Generating Plant

c: Regional Administrator - III NRC
NRR Project Manager, NRC
Sr Resident Inspector, NRC
Minnesota Department of Commerce
Attachment

IE22

| NRC FORM 366 (6-1998) U.S. NUCLEAR REGULATORY COMMISSION | APPROVED BY OMB NO. 3150-0104 EXPIRES 06/30/2001 |
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| **LICENSEE EVENT REPORT (LER)**<br>(See reverse for required number of digits/characters for each block) | Estimated burden per response to comply with this mandatory information collection request: 50 hrs. Reported lessons learned are incorporated into the licensing process and fed back to the industry. Forward comments regarding burden estimate to the Records Management Branch(T-6 F33), U.S. Nuclear Regulatory Commission, Washington, DC 205555-0001, and to the Paperwork Reduction Project (3150-0104), Office of Management and Budget, Washington, DC 20503. If an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to the information collection. |

| FACILITY NAME (1) | DOCKET NUMBER (2) | PAGE (3) |
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TITLE (4) Re-analysis of the High Energy Line Break Resulted in Potential Loss of Division I 4kV Switchgear

| EVENT DATE (5) | | | LER NUMBER (6) | | | REPORT DATE (7) | | | OTHER FACILITIES INVOLVED (8) | |
|---|---|---|---|---|---|---|---|---|---|---|
| MONTH | DAY | YEAR | YEAR | SEQUENTIAL NUMBER | REVISION NUMBER | MONTH | DAY | YEAR | FACILITY NAME | DOCKET NUMBER 05000 |
| 02 | 21 | 00 | 00 | -- 004 -- | 00 | 03 | 22 | 00 | FACILITY NAME | DOCKET NUMBER 05000 |

| OPERATING MODE (9) | N | THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11) | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| | | | 20.2201(b) | | 20.2203(a)(2)(v) | | 50.73(a)(2)(i) | | 50.73(a)(2)(viii) |
| POWER LEVEL (10) | 0 % | | 20.2203(a)(1) | | 20.2203(a)(3)(I) | X | 50.73(a)(2)(ii) | | 50.73(a)(2)(x) |
| | | | 20.2203(a)(2)(i) | | 20.2203(a)(3)(ii) | | 50.73(a)(2)(iii) | | 73.71 |
| | | | 20.2203(a)(2)(ii) | | 20.2203(a)(4) | | 50.73(a)(2)(iv) | | OTHER |
| | | | 20.2203(a)(2)(iii) | | 50.36(c)(1) | | 50.73(a)(2)(v) | | Specify in Abstract below or in NRC Form 366A |
| | | | 20.2203(a)(2)(iv) | | 50.36(c)(2) | | 50.73(a)(2)(vii) | | |

LICENSEE CONTACT FOR THIS LER (12)

| NAME | TELEPHONE NUMBER (Include Area Code) |
|---|---|
| Mohamad Marashi | 612-295-1425 |

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

| CAUSE | SYSTEM | COMPONENT | MANUFACTURER | REPORTABLE TO EPIX | CAUSE | SYSTEM | COMPONENT | MANUFACTURER | REPORTABLE TO EPIX |
|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | |
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| SUPPLEMENTAL REPORT EXPECTED (14) | | | | EXPECTED SUBMISSION DATE (15) | MONTH | DAY | YEAR |
|---|---|---|---|---|---|---|---|
| | YES (If yes, complete EXPECTED SUBMISSION DATE). | X | NO | | | | |


**ABSTRACT** (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

An analysis of a high energy line break (HELB) on the 911ft elevation of the Turbine Building indicated flooding of the Division I 4kV switchgear room and possible loss of the Division I 4kV switchgear. The analysis indicated that the peak flood level on the 911ft elevation of the Turbine Building Division I 4kV switchgear room would cause a loss of Division I 4kV power. With an assumed loss of offsite power, Division II Emergency Diesel Generator was considered the worst case single active failure. Therefore, this event could potentially result in loss of the station AC power from both divisions of the 4kV distribution system. Modifications were installed to prevent water from entering the Division I 4kV switchgear room.

TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

Description

During September 1999, a re-analysis of a high energy line break (HELB) on the 911ft elevation of the Turbine Building[1] indicated flooding of the Division I 4kV switchgear[2] room and possible loss of the Division I 4kV switchgear. The HELB analysis for a postulated break in the feedwater[3] pump[4] discharge line was re-evaluated as part of the ongoing USAR Review Program at Monticello. The analysis assumed initiation of the local fire protection system[5], a break of two small bore service water[6] lines, a fire protection line and a service water header due to broken feedwater pipe whip.

A walkdown of the feedwater pump area was performed to verify the assumed pipe whip targets. It was verified that two of the targets, a fire protection line and a service water header were incorrectly assumed to be pipe whip targets in the original analysis. Plates were installed at the bottom of the doors leading to the 4kV switchgear room in order to restrict flow into the 4kV switchgear room until a complete verification of the analysis was performed.

During the 2000 refueling outage, analysis of the feedwater pump discharge line break was completed. The analysis indicated that the peak flood level on the 911ft elevation of the Turbine Building Division I 4kV switchgear room would cause a loss of Division I 4kV power. With an assumed loss of offsite power, Division II Emergency Diesel Generator (EDG)[7] was considered the worst case single active failure. Therefore, this event could potentially result in loss of the station AC power from both divisions of the 4kV distribution system. This event was reported to the NRC on February 21, 2000.

[1] EIIS System Code = NM
[2] EIIS System Code = EB
[3] EIIS System Code = SJ
[4] EIIS System Code = P
[5] EIIS System Code = KP
[6] EIIS System Code = KG
[7] EIIS System Code = EK

## Event Analysis

### Analysis of Reportability

This report is being submitted in accordance with 10 CFR 50.73(a)(2)(ii): "Any event or condition that resulted in the nuclear power plant being in a condition that was outside the design basis of the plant."

This event is not reportable under 10CFR50.73(a)(2)(v). The ability to safely shutdown the plant and mitigate the consequences of an accident was not compromised because the Division II 4kV distribution system was not affected.

### Safety Significance

A feedwater line break is an unlikely event. The feedwater piping has been analyzed seismically to be able to withstand the design basis earthquake. Also, this piping is included in the formal program for monitoring flow induced erosion/corrosion in piping systems and has been monitored for wall thinning. The thinning rates are within normally expected values. No piping replacement has been required in the feedwater piping. Failure due to either of these mechanisms is unlikely.

In the event of a feedwater line break, flooding of the feedwater pump area and the Division I 4kV switchgear room could result in loss of Division I 4kV switchgear. Safe plant shutdown would normally be assured in this case by Division II equipment powered by Division II EDG. This event with an assumed loss of offsite power and single active failure of the Division II EDG could result in loss of the station AC power from the 4kV distribution system. The probability of a feedwater line break at the discharge of the pump and a concurrent failure of the Division II EDG is shown to be 1.4E-7 per year. In this case, reactivity control, pressure control, level control and adequate core cooling could be accomplished using the plant DC power and AC power from inverters to supply power to HPCI[1] and RCIC[2] systems for core injection. Per the Station Blackout analysis, this would provide up to at least four hours to either recover the Division II EDG or offsite power.

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[1] EIIS System Code = BJ
[2] EIIS System Code = BN

## Cause

During the ongoing USAR Review Program at Monticello, plant personnel determined that a re-evaluation of the current HELB analysis of the feedwater pump discharge line break in the 911ft elevation of the Turbine Building was required to account for additional water sources from the fire and service water systems. Additional plant design input information was used to establish parameters for the floor drains, sumps, and sump pumps. In addition, door cracks and gaps were modeled based on plant staff walkdown measurements. The additional water sources and enhancements to the original model resulted in higher peak transient flood levels in the Turbine Building and in the Division I 4kV switchgear room.

## Actions

Corrective Actions:

A water-tight barrier was installed outside the Division I 4kV switchgear room to prevent water from entering the Division I 4kV switchgear room. Also, additional barriers were installed at the bottom of door No. 479 leading to the room outside the Division I 4kV switchgear room. A floor drain in the Division I 4kV room was positively plugged.

Preventive Actions:

The High Energy Line Break analyses is continuing to be reviewed as part of the overall USAR Review Program

## Failed Component Identification -

None

## Similar Events In the Last Ten Years

A discrepancy in the High Energy Line Break analysis was reported in May 1996. This discrepancy was of a different type. The identified discrepancy involved an error in the licensing basis HELB analysis for the Turbine Building. The error resulted in improper analyzed ambient temperature in the vicinity of Division II MCC-142, Division II MCC-143 and the 4kV switchgear rooms for the limiting feedwater HELB. The re-analysis determined that these areas could become harsh environment; however, the licensing basis analysis did not predict these areas to be harsh environments.

In September 1996, a discrepancy of a different type was identified in the mass and energy release calculated for a high energy line break in the Reactor Water Cleanup (RWCU) piping. A reanalysis of a postulated RWCU line break outside containment identified an increase in the potential radiological consequences of the accident. Results of the analysis showed that the radiological consequences were below the guidelines of 10 CFR Part 100 for offsite doses and below the guidelines of 10 CFR Part 50, Appendix A, GDC 19 for control room doses.