Northern States Power Company

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

NSP

May 3, 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 2000-009
Procedural Inadequacy Results in Mismatch of Feedwater Flow Instruments and Process Computer Calibration Causing Operation Above Licensed Power

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

Contact David Musolf, Consulting Production Engineer, at (763) 295-1201 if you require further information.

Byron Day

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
LICENSEE EVENT REPORT (LER) (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)
MONTICELLO NUCLEAR GENERATING PLANT	05000 – 263	1 OF 4

TITLE (4) Procedural Inadequacy Results in Mismatch of Feedwater Flow Instruments and Process Computer Calibration Causing Operation Above Licensed Power

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
04	04	00	00	-- 009 --	00	05	03	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)	100		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)
David Musolf	763-295-1201

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

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)

During normal full power operation on April 4, 2000, an Instrument and Control Technician performing a calibration of feedwater flow transmitters noted a small mismatch between the transmitter calibration values and the corresponding values generated by the plant process computer. Investigation revealed that the span of the instruments was changed on October 22, 1998, following an increase in maximum licensed thermal power. However, the process computer calibration constants for feedwater flow were not changed as required. These computer points are used in the reactor thermal power calorimetric calculations. It is estimated that this error resulted in a calculated reactor power of less than 3.7 MWt, or 0.2%, below actual power. The net effect of this error was that the reactor exceeded maximum licensed power by no more than 0.2% for a total of 316 days since the error was introduced. This event was the result of procedural inadequacies.

Description

On April 4, 2000, during normal full power operation, calibration of feedwater flow transmitters[1] FT-4383A and FT-4383B was being performed. The Instrument and Control (I&C) Technician doing this work noticed a slight difference between computer point[2] CFW203, "RX FW FLOW A D/P FE-4382A," and the output from transmitter FT-4383A. Even though no specific acceptance criteria were specified for the associated computer points, the I&C Technician notified computer engineering of this slight difference. Computer engineering determined that a mismatch existed between the calibration specifications for FT-4383A & B and the database calibration constants for the corresponding process computer points CFW203 and CFW204.

Investigation revealed that this discrepancy occurred when the span of FT-4383A and FT-4383B was changed on October 22, 1998, following an increase in licensed thermal power from 1670 to 1775 MWt. A small change in feedwater flow transmitter calibrated span was necessary due to an increase in feedwater temperature and flow at the new power level.

Immediate action was taken to reduce indicated reactor power to less than 1770 MWt while investigation of this event continued.

Process computer daily logs were reviewed to determine the extent to which licensed thermal power was exceeded. It was found that since October 22, 1998, reactor power exceeded maximum licensed power by up to 0.1% for 154 days and by up to 0.2% for 162 days for a total of 316 days of operation above licensed power. This condition is a violation of the Operating License.

Event Analysis

Analysis of Reportability

This event is reportable under 10 CFR 50.73(a)(2)(ii)(B) since it represents a condition that is outside of the licensing and design basis of the plant. The licensed reactor thermal power of 1775 MWt was exceeded for most of the period between October 22, 1998 and April 4, 2000, by up to 0.2%.

Although this condition was prohibited by the Monticello Operating License, the condition had little safety significance. No safety function of any plant structure or system was affected. The event is therefore not reportable in accordance with 10 CFR 50.73(a)(2)(v).

[1] EIIS Component Code: FT
[2] EIIS System Code: ID

Safety Significance

The error in reactor thermal power calculation performed by the plant process computer could potentially affect the Average Power Range Monitor (APRM) Technical Specification scram setpoints and the reactor fuel assembly Technical Specification thermal limits. Conservative operating practices prevented violation of these limits as follows:

APRM Setpoints
The Technical Specifications require that APRMs initiate a neutron flux scram in the event of a power transient to avoid exceeding the fuel cladding integrity Safety Limit. Procedures require APRM gains to be adjusted to ensure they are conservatively set with respect to Technical Specification limits by at least 1%. The 0.2% error in reactor calculated reactor thermal power introduced by the feedwater flow measurement error was conservatively bounded by the 1% margin in the APRM gain setting. Therefore, Technical Specifications related to APRM setting were not violated.

Thermal Limits
The Technical Specifications also define reactor fuel thermal limits to assure that design heat flux and power density limits are not exceeded. Operations Manual action limits for thermal margin monitoring are conservatively set at 97% of the Technical Specification limits. The conservative application of these action limits prevented a violation of the Technical Specification thermal limits.

The analysis of transients and design basis accidents is described in the Monticello Updated Safety Analysis Report (USAR). In accordance with NRC guidance, these analyses conservatively assume an initial reactor power of 102% to allow for possible instrument errors in determining power level. This 2% margin conservatively bounds the calculated overall error in the reactor thermal power calculation and the additional 0.2% error introduced by the computer calibration error.

Therefore, the 0.2% error in calculated reactor power had very limited safety significance.

Cause

The cause of this event was inadequate procedures.

In conjunction with the Monticello Power Rerate Project, calculations were performed to determine corrections to the feedwater flow transmitter span setpoints and the process computer feedwater flow algorithm. Increasing rated reactor thermal power from 1670 to 1775 MWt resulted in an increase in feedwater flow and temperature. Feedwater nozzle flow and thermal expansion coefficients were slightly affected by these changes. Calculations were completed to determine the necessary changes to the process computer feedwater flow algorithm. Calculations to determine the necessary changes to the calibrated span of the feedwater flow transmitters were also performed.

The process computer feedwater flow algorithm was properly updated based on the new calculations using the plant computer Problem/Change Report (PCR) process. The span of the flow transmitters was properly changed based on the new calculations using the plant Setpoint Change Request (SCR) process.

The system engineer who processed the SCR did not realize that an associated change to the process computer was necessary. Computer engineering was not informed that the flow transmitters were being respanned. Thus it was not recognized that an update to the process computer feedwater flow transmitter milliamp to differential pressure correlation was necessary.

Investigation determined that the current SCR process is weak. SCRs associated with setpoint changes for instruments with an associated computer point do not currently require the notification of computer engineering personnel.

Corrective Actions

Immediate corrective action was taken to reduce indicated reactor power to 1770 MWt.

Reactor power was administratively limited to 1770 MWt (indicated) until the following actions were completed:

1. The Power Rerate Program feedwater flow instrumentation calibration calculations were verified by an independent third party senior engineer.

2. The Power Rerate Program feedwater flow algorithm changes were verified by an independent third party senior engineer.

3. Necessary corrections to the process computer feedwater flow transmitter milliamp to differential pressure correlation were made using the PCR process in accordance with the original SCR and associated Work Order.

4. All instrumentation and computer points associated with the reactor thermal power process computer algorithm were verified to be consistent for operation at 1775 MWt.

Following completion of these actions, indicated reactor thermal power was returned to 1775 MWt at 1625 on April 25, 2000.

Actions to prevent similar problems in the future will be completed as follows:

1. Changes to the SCR process will be made to require the involvement of computer engineering personnel in the review of any setpoint change involving an instrument with an associated computer point. Until this process change is finalized, all SCRs will be routed through computer engineering.

2. The Component Master List (CML) database will be reviewed to ensure that all instruments with associated computer points are identified.

3. CML instrument calibration worksheets will be modified, where necessary, to ensure that specific calibration acceptance criteria are specified for computer points used in core thermal power calculations.

Failed Component Identification

Not applicable.

Similar Events

No similar events have occurred in the past at Monticello.