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

January 14, 2011

L-MT-11-001
10 CFR 50.73

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

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

LER 2011-001, Reactor Vessel Overfill in Appendix R Scenario

The Licensee Event Report (LER) for this occurrence is attached.

Summary of Commitments

This letter contains no new commitments and no revisions to existing commitments.

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

Enclosure

cc: Administrator, Region III, USNRC
Project Manager, Monticello, USNRC
Resident Inspector, Monticello, USNRC

NRC FORM 366 (10-2010)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES: 10/31/2013

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects.resource@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0066), Office of Management and Budget, Washington, DC 20503. If a means used to impose 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.

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4. TITLE
Reactor Vessel Overfill in Appendix R Scenario

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
11	12	2010	2011	001	00	01	14	2011	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)			
1	☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
	☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
	☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☒ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
	☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
10. POWER LEVEL	☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
47%	☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
	☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
	☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
	☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
Steven K. Speight	763.271.7636

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

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
O YES (If yes, complete 15. EXPECTED SUBMISSION DATE).	⊙ NO				

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

During the performance of a fire protection assessment, a determination was made that the fire protection safe shut down analysis does not address a postulated reactor vessel overfill event.

In a postulated fire event that required the evacuation of the Control Room with a loss of offsite power, High Pressure Coolant Injection and Reactor Core Isolation Cooling pumps would start if the low reactor water level setpoint is reached. For this fire, damage could result in the failure of the high reactor water level trip circuit for the High Pressure Coolant Injection and Reactor Core Isolation Cooling systems. This could result in a reactor vessel overfill.

Compensatory measures for this issue have been implemented.

EVENT DESCRIPTION

On 12 November, 2010, the site determined that a postulated reactor vessel overfill scenario exists which had not been analyzed under the Appendix R Fire Protection Program. In the scenario, High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) systems start on low-low reactor water level and fail to trip on high reactor water level due to fire damage. This could result in a reactor vessel overfill as discussed below.

For a fire requiring evacuation of the Control Room, the post fire safe shutdown is accomplished from the Alternate Shut Down System (ASDS). For the postulated scenario, a fire in the Control Room or Cable Spreading Room would cause Operations personnel to evacuate the Control Room and proceed to the ASDS panel located in the Emergency Filtration Train (EFT) Building. The scenario requires the assumption of an unlikely loss of offsite power, with consequential decrease of water inventory in the reactor. When reactor water level reaches the low-low reactor water level setpoint, the main steam isolation valves close and HPCI and RCIC pumps start.

Safety related reactor water level switches LS-2-3-672E(LIS) and LS-2-3-672F(LIS) provide a high reactor water level trip signal. When a high reactor water level condition occurs both switches actuate (2-out-of-2 logic) to trip HPCI and RCIC. Fire damage to this circuit could prevent the high reactor water level trip. If HPCI/RCIC fail to trip on high reactor water level, then the reactor vessel would continue to fill until sufficient water fills the HPCI and RCIC steam lines to stall the HPCI and RCIC pumps.

After arriving at the ASDS panel, Operations personnel could procedurally initiate a reactor vessel blow down by manual operation of the safety relief valves (SRV) to allow for low pressure reactor water inventory makeup and decay heat removal. The HPCI and RCIC steam supply lines and SRVs connect to the main steam lines at the same elevation. When the SRVs are manually opened from the ASDS panel, the valves may be subjected to high pressure steam/water flow.

The SRVs and their associated tailpipes have been analyzed preliminarily and the loads found acceptable.

EVENT ANALYSIS

The event is reportable to the NRC under 10 CFR 50.73(a)(2)(ii)(B) – Degraded or Unanalyzed Condition. The site reported the event on November 12, 2010.

This event is not a Safety System Functional Failure.

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SAFETY SIGNIFICANCE

There were no nuclear, radiological or industrial safety significant consequences related to this event.

The Monticello risk assessment group reviewed the event for risk impact. Risk of core damage and large early release are not significantly impacted by effects on the SRV function during inadvertent overfill of the reactor vessel resulting from failure of the HPCI/RCIC high reactor water level trip due to a fire that defeats the trip logic. The SRVs are not significantly impacted from a vessel overfill event as they are capable of performing their intended function in both the safety mode as well as the depressurization mode. Additionally, industry operating experience and preliminary MNGP plant specific analysis support a conclusion that the SRV tailpipes will remain intact following SRV lifts while subject to liquid and/or two phase flow.

Based on the above, the health and safety of the public have not been affected.

This LER will be updated if non-conservative changes to the safety significance discussion above are required. Compensatory measures will remain in place until corrective actions are completed.

CAUSE

The cause of this event was that previous Appendix R analyses failed to consider an insufficient consideration of HPCI and RCIC automatic initiation and failure to trip.

CORRECTIVE ACTION

Corrective actions are being tracked in the Corrective Action Program.

1. Compensatory actions, in accordance with the Fire Protection Program, have been taken in those areas where a fire could cause the postulated scenario.
2. A memorandum detailing the postulated scenario has been issued to Operations personnel as a briefing for this condition.
3. Other actions:
    i. A site specific evaluation of this condition will be performed as part of Appendix R compliance as well as Regulatory Guide 1.189 compliance.
    ii. Based on the completion of formal SRV tailpipe analysis, HPCI and RCIC will be evaluated as necessary for modifications.

PREVIOUS SIMILAR EVENTS

There have been no similar events in the last 3 years.

OTHER

A discussion was held with NRC Region III, extending the due date for this LER to 14 January 2011 (three days).