



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

March 30, 2011

L-MT-11-023  
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 Facility Operating License No. DPR-22

LER 2011-001, Revision 1, Reactor Vessel Overfill in Appendix R Scenario

A revision to the Licensee Event Report (LER) for this occurrence is attached. The Northern States Power Company has clarified, in the safety significance section, which systems make up a post-fire safe shutdown set. Additionally, a grammatical error in the Cause section has been corrected.

Summary of Commitments

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

A handwritten signature in black ink, appearing to read 'Timothy J. O'Connor', written over a large, light-colored scribble.

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

**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](mailto: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.

<b>1. FACILITY NAME</b> Monticello Nuclear Generating Plant	<b>2. DOCKET NUMBER</b> 05000 263	<b>3. PAGE</b> 1 OF 3
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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
11	12	2010	2011	001	01	03	30	2011	FACILITY NAME	DOCKET NUMBER 05000

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

**12. LICENSEE CONTACT FOR THIS LER**

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

<b>14. SUPPLEMENTAL REPORT EXPECTED</b>	<b>15. EXPECTED SUBMISSION DATE</b>	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.

**LICENSEE EVENT REPORT (LER)  
CONTINUATION SHEET**

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**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.

**LICENSEE EVENT REPORT (LER)  
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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.

Monticello's minimum post-fire safe shutdown set consists of one train of Residual Heat Removal, one train of Core Spray, Safety Relief Valves, and associated auxiliary and support systems. HPCI and RCIC systems are not credited in this Appendix R scenario.

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 a 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).