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Monticello Nuclear Generating Plant
Operated by Nuclear Management Company, LLC

August 30, 2007

L-MT-07-063
10 CFR Part 50.73

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

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

LER 2007-003 Revision 1, "Failure to Enter a Required Technical Specification Action during Control Rod Drive Exercising"

A revision to the Licensee Event Report (LER) for this occurrence is attached.

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

For T. O'Connor

Timothy J. O'Connor
Site Vice President, Monticello Nuclear Generating Plant
Nuclear Management Company, LLC

Enclosure

cc: Administrator, Region III, USNRC
Project Manager, Monticello, USNRC
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NRC FORM 366 (6-2004)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 6-30-2007
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 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs1@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose 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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TITLE (4) Failure to Enter a Required Technical Specification Action during Control Rod Drive Exercising

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
04	20	2007	2007	– 003 –	01	08	30	2007	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	4	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply) (11)							
POWER LEVEL (10)	000		20.2201(b)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(B)		50.73(a)(2)(ix)(A)
			20.2201(d)		20.2203(a)(4)		50.73(a)(2)(iii)		50.73(a)(2)(x)
			20.2203(a)(1)		50.36(c)(1)(i)(A)		50.73(a)(2)(iv)(A)		73.71(a)(4)
			20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)		73.71(a)(5)
			20.2203(a)(2)(ii)		50.36(c)(2)		50.73(a)(2)(v)(B)		OTHER Specify in Abstract below or in NRC Form 366A
			20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)		
			20.2203(a)(2)(iv)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)		
			20.2203(a)(2)(v)	X	50.73(a)(2)(i)(B)		50.73(a)(2)(vii)		
			20.2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)		
			20.2203(a)(3)(i)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Ron Baumer	763-295-1357

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

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
X	AA	ZI	6080	No					

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

ABSTRACT

Note: This LER is being revised to annotate that the interlock failure was a safety system functional failure and to clarify the safety significance of the event.

On April 20, 2007 while the plant was in Mode 4 for refueling outage 23, control rod exercising was being performed. While exercising control rod 26-31, the operators realized they did not get the same numerical light indication for the control rod they had previously tested (rod 26-35). The incorrect light indication meant that the one rod out interlock for control rod 26-35 was inoperable. The one rod out interlock for control rod 26-35 should have been declared inoperable and appropriate Technical Specification Action 3.9.2.A should have been entered prior to commencing testing on the control rod 26-31. Exercising control rod 26-31 without the appropriate actions taken for 26-35 was a violation of Monticello Technical Specifications.

The cause of the event was incorrect acceptance criteria in the procedure being used. Contributing to the event was the lack of operator proficiency and operator misdiagnosis of the issue. Corrective actions taken for the event were to replace the control rod position indication probe, revise the procedure, and additional training was requested for the operators.

Description

On April 20, 2007, the plant was in MODE 4 (cold shutdown) for refueling outage 23 with the Mode switch in refuel for Control Rod Drive (CRD) [AA] exercise testing. Control rod [ROD] 26-35 was withdrawn per procedure. Per the procedure acceptance criterion the "00" position went out when the rod was withdrawn. Following CRD exercising, the control rod was re-inserted. During these tests, it was recognized that a work order identified a problem with control rod 26-35 position indication. Having an equipment issue on the control rod position indication made the operators believe they should expect some abnormal behavior. The next control rod, 26-31, was selected and similarly exercised per procedure. When this control rod was withdrawn, the numerical indication changed from green to amber. One of the operators recalled that during withdrawal of control rod 26-35 the numerical indication remained green. The crew completed testing of Control Rod 26-31 and fully inserted the control rod. The crew stopped, reviewed the acceptance criteria of the procedure, and notified the control room supervisor that the indication of the previous control rod was not correct.

Control rod testing was suspended while the crew evaluated the indications noted on control rod 26-35. Upon review of the bases for the Technical Specification (TS) it was discovered that the full in indication for TS compliance is the double dash or green light reed switch. The "00" position indication does not provide input for the one-rod-out interlock as specified in the procedure. TS Action 3.9.2.A was entered and the drive for control rod 26-35 was de-activated.

Technical Specifications require that the full rod in position interlock be operable for each control rod for the conditions during the event. Upon discovery that control rod full in position for 26-35 was not operable, TS Action 3.9.2.A should have been entered. However, testing was performed on an additional control rod. The result was non-compliance with TS 3.9.2, "Refuel Position One-Rod-Out Interlock" since the interlock was inoperable and control rod withdrawal was not immediately suspended.

Event Analysis

Per 10 CFR 50.73 (a)(2)(i)(B), an Operation or Condition Prohibited by Technical Specifications requires a Licensee Event Report. There is no requirement for reporting in accordance with 10 CFR 50.72 for this event.

The event involves a safety system functional failure.

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

Safety Significance

The one-rod-out interlock is designed to ensure that movement of more than one control rod is restricted to prevent the reactor from becoming critical during refueling operations. During refueling operations, no more than one control rod is permitted to be withdrawn with fuel in the cell. The inoperability of the interlock for control rod 26-35 would have allowed control rod 26-35 and any other control rod to be withdrawn at the same time.

However two additional barriers exist to prevent two control rods from being withdrawn at the same time. All control rod movement is directed by procedures. Under the conditions for the event, no procedures direct multiple control rod withdrawal in Mode 4 and licensed operators are trained not to withdraw two control rods simultaneously in Mode 4.

If these barriers had failed and two control rods were withdrawn, it is possible for the reactor to become critical if the second control rod were close to control rod 26-35. These barriers did not fail and no more than one control rod was actually withdrawn at any time during this exercise evolution. At no time did the reactor become critical. This issue does impact nuclear safety due to the violation of Technical Specifications, but for the above reasons there were no industrial or radiological safety issues associated with this event.

The Probabilistic Risk Assessment (PRA) group performed an evaluation of the event and, based on discussions with Nuclear Engineering, concluded that the risk of core damage as a result of this event was minimal for the following reasons:

- The Reactor Manual Control System would have permitted the withdrawal of control rod 26-35 and any other rod. Withdrawal of three control rods could not have occurred.
- Nuclear Engineering performed a calculation with both control rods 26-35 and 26-31 fully withdrawn. The calculation indicated that the reactor would have become critical with both of these control rods fully withdrawn. No fuel damage would have occurred because the reactivity conditions of the potential criticality would have been bounded by the Control Rod Drop accident.

Cause

The cause of the event was an incorrect procedure acceptance criterion for satisfying the requirements of Tech Spec 3.9.4. A contributing cause to the event was that the Operators were not proficient in utilization of special operations procedures that are typically initiated once per cycle. A second contributing cause was that the operators did not fully understand the reason for the performance of the step and the impact of the equipment deficiency associated with control rod 26-35 position indication.

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

Corrective Action

The following corrective actions are planned or have been completed:

- The position indication probe for control rod 26-35 was replaced and the control rod full in indication was returned to operable. (Completed)
- Individuals involved have been coached and counseled on their role in this event. (Completed)
- The procedure was revised to correctly implement the technical specification. (Completed)
- The station requested additional training for operators prior to outages that will require them to exercise these special operations procedures that do not get routinely utilized. (In progress)

Failed Component Identification

Probe, Rod Position Indication – General Electric Company – Part Number: 797E111G001

Previous Similar Events

A review of station events found no events that were considered to be related to this event.