WOLF CREEK
NUCLEAR OPERATING CORPORATION

Rick L. Gardner
Plant Manager

May 3, 2010

WO 10-0033

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

Subject: Docket No. 50-482: Licensee Event Report 2010-005-00, "Reactor Trip due to Low Steam Generator Level from Trip of Main Feedwater Pump"

Gentlemen:

The enclosed Licensee Event Report (LER) 2010-005-00 is being submitted pursuant to 10 CFR 50.73(a)(2)(iv)(A) regarding an Engineered Safety Features Actuation and subsequent reactor trip at Wolf Creek Generating Station.

This letter contains no commitments. If you have any questions concerning this matter, please contact me at (620) 364-4156, or Mr. Richard D. Flannigan at (620) 364-4117.

Sincerely,

Rick L. Gardner

RLG/rlt

Enclosure

cc: E. E. Collins (NRC), w/e
G. B. Miller (NRC), w/e
B. K. Singal (NRC), w/e
Senior Resident Inspector (NRC), w/e

IE22
NRR

NRC FORM 366 (9-2007) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 08/31/2010
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 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@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 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 Trip due to Low Steam Generator Level from Trip of Main Feedwater Pump

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
03	02	2010	2010 -	005 -	00	05	03	2010	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)
100	☐ 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

FACILITY NAME	TELEPHONE NUMBER (Include Area Code)
Richard D. Flannigan, Manager Regulatory Affairs	(620) 364-4117

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	SJ	INVT	SCI	Y					

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

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

On 3/2/2010 at 1458 CST, while performing procedure SYS PN-200, "Energizing and Deenergizing Inverters PN09 or PN10," Wolf Creek Generating Station (WCGS) experienced a reactor trip due to Steam Generator (SG) Water Level – Low Low actuation signal. The unit received a feedwater isolation and auxiliary feedwater actuation (both motor-driven and turbine-driven) because of the low SG level. All control rods inserted fully and the Reactor Trip System and the Engineered Safety Feature System performed as expected.

The SG Water Level – Low Low actuation signal was initiated due to a trip of the train "A" main feedwater pump as a result of the failed transfer of inverter PN09 to an alternate power supply. Inverter PN09 did not transfer from the normal to alternate power supply due to the sticking of the reed relay on the static transfer switch circuit board.

The safety significance of this event is low. This event is bounded by analyses as reported in the WCGS Updated Safety Analysis Report (USAR) Section 15.2.7, "Loss of Normal Feedwater Flow." There were no adverse effects on the health and safety of the public.

PLANT CONDITIONS PRIOR TO EVENT

MODE – 1
Power – 100

EVENT DESCRIPTION

On 3/2/2010 at 1458 CST, Wolf Creek Generating Station (WCGS) experienced an automatic reactor trip due to Steam Generator (SG) Water Level – Low Low actuation signal [EIIS Code: JB]. The SG water level – Low Low actuation signal was caused by the loss of the speed sensor to the train "A" main feedwater (MFW) pump control circuitry [EIIS Code: SJ]. The loss of the speed sensor was caused by the loss of 120 VAC power to the circuit.

The loss of power to the speed sensor occurred when inverter PN09 [EIIS Code: SJ–INVT] was being transferred to its alternate power supply to facilitate a minor maintenance activity to replace a light bulb. The power supply was being transferred to the alternate supply using procedure SYS PN-200, "Energizing and Deenergizing Inverters PN09 or PN10." Inverter PN09 did not transfer from the normal to alternate power supply due to the sticking of the reed relay on the static transfer switch circuit board.

The unit received a feedwater isolation and auxiliary feedwater actuation (both motor-driven and turbine-driven) as a result of the SG water level – Low Low actuation signal. All control rods inserted fully and the Reactor Trip System (RTS) and the Engineered Safety Feature Systems [EIIS Code: JE] performed as expected. Decay heat removal following the reactor trip was via the atmospheric relief valves and auxiliary feedwater flow to the steam generators. At the time of the reactor trip the steam dump valves were not available as the loss of inverter PN09 caused the loss of the C-9 interlock blocking steam dump operation.

The steam dump valves became available at 1554 CST, on 3/2/2010, when inverter PN09 was reenergized from its alternate power supply.

All systems functioned as designed with the exception of the instrumentation powered by inverter PN09. PN09 does not supply any safety-related equipment. At the time of the reactor trip the train "A" diesel generator [EIIS Code: EK] and the train "A" Class 1E air conditioning unit [EIIS Code: VI] were out of service for maintenance.

BASIS FOR REPORTABILITY

The reactor trip and subsequent actuation of Engineered Safety Feature Actuation System instrumentation described in this event is reportable per 10 CFR 50.73(a)(2)(iv)(A), which requires reporting of "Any event or condition that resulted in manual or automatic actuation of any of the systems listed in paragraph (a)(2)(iv)(B) of this section." Paragraph (B)(1) of 10 CFR 50.73(a)(2)(iv) includes "Reactor protection system (RPS) including: reactor scram or reactor trip." Paragraph (B)(6) of 10 CFR 50.73(a)(2)(iv) includes "PWR auxiliary or emergency feedwater."

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ROOT CAUSE

During the performance of procedure SYS PN-200, inverter PN09 did not transfer from the normal to alternate power supply due to the sticking of the reed relay on the static transfer switch circuit board. A contributing cause was the failure to perform timely preventive maintenance (PM) on components exceeding their recommended life expectancy. Life Cycle Management and Preventive Maintenance Optimization plans determined the schedule for implementing PMs for the inverters. The inverters were on a staggered schedule for PM parts replacement during the refueling outages. PM activities on PN010 were completed in Refueling Outage 17 in Fall 2009. PM activities for PN009 were scheduled for Refueling Outage 18 in Spring 2011.

CORRECTIVE ACTIONS

The static switch control circuit board was replaced along with a number of other circuit boards and fuses. After replacement, the power supplies were transferred successfully several times.

Preventive maintenance has been established to replace the circuit boards in accordance with the manufacturer recommendations.

SAFETY SIGNIFICANCE

The safety significance of this event is low. This event is analyzed as reported in WCGS Updated Safety Analysis Report (USAR) Section 15.2.7, "Loss of Normal Feedwater Flow." Results of the analysis show that a loss of normal feedwater does not adversely affect the core, the reactor coolant system, or the steam system, since the auxiliary feedwater capacity is such that reactor coolant water is not relieved from the pressurizer relief or safety valves.

There were no adverse effects on the health and safety of the public.

OPERATING EXPERIENCE/PREVIOUS EVENTS

LER 2009-001-00 described a reactor trip due to a Main Feedwater Regulating Valve (MFRV) closure in response to failures of the primary and secondary fuses for the Westinghouse 7300 control card frame that contained the associated control cards for the MFRV.

LER 2004-002-00 described a reactor trip due to a MFRV closure caused by the valve plug in the MFRV separating from the valve stem.