WOLF CREEK
NUCLEAR OPERATING CORPORATION

Russell A. Smith
Plant Manager

December 16, 2010

WO 10-0084

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

Subject: Docket No. 50-482: Licensee Event Report 2010-012-00, "Reactor Trip due to Operators Inability to Control Steam Generator Level Oscillations at Low Power"

Gentlemen:

The enclosed Licensee Event Report 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,

Russell A. Smith

Enclosure

cc: E. E. Collins Jr (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 U.S. NUCLEAR REGULATORY COMMISSION (10-2010)	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 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 Operators Inability to Control Steam Generator Level Oscillations at Low Power

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
10	17	2010	2010	- 012 -	00	12	16	2010	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE: Mode 1

10. POWER LEVEL: 15%

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR§: *(Check all that apply)*

☐ 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)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 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

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 October 17, 2010 at 0952 CDT, Wolf Creek Generating Station was at approximately 15% power during a startup when a turbine trip and Feedwater Isolation Signal occurred due to High-High Steam Generator level. An automatic reactor trip occurred at 0953 CDT due to decreasing SG level. All control rods fully inserted and the Reactor Trip System and the Engineered Safety Feature Systems performed as expected.

Emergency boration was initiated when reactor coolant system temperature dropped below 550 degrees Fahrenheit. Emergency boration was secured after 58 minutes when RCS temperature was raised to greater than 550 degrees Fahrenheit.

The cause of the trip was the Control Room operators were unable to maintain SG levels at low power. Procedures were revised to address the issue. 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."

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PLANT CONDITIONS AT THE TIME OF THE EVENT

Mode 1
Approximately 15% power

DESCRIPTION OF THE EVENT

A reactor trip occurred during the startup on October 17, 2010. Wolf Creek Generating Station (WCGS) was coming out of a two week forced outage. All major equipment was available for the startup. Due to the length of the shutdown, the reactor core was xenon free and had low decay heat. Criticality was achieved at 1816 CDT on October 16, 2010. Main feedwater pump 'B' [EIIS Code: SJ-P] was placed in service at 2320 CDT and power was raised to 7% power. Turbine warming was completed at 0854 CDT on October 17, 2010, which allowed reactor power to be increased to 15%. At 0912 CDT, the turbine [EIIS Code: TA] was rolled to 1800 rpm in preparation for synchronization to the grid. At 0922 CDT, the reactor was at approximately 15% power.

Though feedwater pre-heating was in service, feedwater temperature decreased steadily during the power increase, initiating divergent Steam Generator (SG) level [EIIS Code: JB] oscillations. The Control Room operators took manual control of the 'B' bypass feed regulating valve [EIIS Code: JB-LCV] but were unable to dampen the level oscillations. SG level in the 'B' generator reached the High-High level setpoint of 78% at 0952 CDT, which resulted in a turbine trip and feed water isolation signal (FWIS). Motor-driven auxiliary feedwater pumps (AFW) [EIIS Code: BA-P] started, but their combined feed capacity was insufficient to maintain the current power level. The Control Room operators began to reduce reactor power to stay within the capacity of the Auxiliary Feed Water System by inserting control rods in manual. Recognizing that the power reduction could not be accomplished in time, the Control Room supervisor ordered the reactor be manually tripped. At 0953 CDT on October 17, 2010 the reactor automatically tripped [EIIS Code: JE] on SG 'A' Low-Low level of 23.5%, one second earlier than the manual trip.

Emergency response procedures were entered after the reactor trip. Due to a low initial decay heat load, full AFW flow, and pre-heating steam loads, Reactor Coolant System (RCS) average temperature fell below 550 degrees Fahrenheit. Emergency Boration [EIIS Code: CB] was initiated at 1001 CDT. The Control Room operators took additional action to terminate the cooldown by isolating major steam loads and reducing AFW flow. At 1048 CDT, operators took additional action to eliminate the cooldown and closed the Main Steam Isolation Valves [EIIS Code: SB-ISV]. With all steam loads isolated, RCS temperature recovered and the operating crew stabilized the plant at hot standby. The minimum RCS temperature reached was 538.7 degrees Fahrenheit. Emergency boration was secured at 1059 CDT. No safety limits were challenged or exceeded during the event. The unit was successfully restarted on October 19, 2010 after a 27-hour delay due to the reactor trip.

BASIS FOR REPORTABILITY

The reactor trip and actuation of ESFAS instrumentation actuation 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."

ROOT CAUSE

Control room operators were unable to maintain SG levels at low power using the main feed regulating bypass valves in automatic or manual control eventually over feeding the 'B' SG. As a result the turbine tripped on High-High SG level, which initiated a FWIS. The basis for the operators inability to control SG levels is provided below.

Guidance in plant operational procedures was not aligned with the required plant design parameters for low power operations, specifically in controlling feed water preheating, SG level control and response to a FWIS. As a result, the operation of the plant during power ascension was outside the main feedwater bypass valve optimum operating region and the feedwater preheating limitations.

Main feedwater bypass valve characteristics and SG level process control settings did not provide stable (convergent) operating characteristics during low power operations. As a result, there was an over reliance on manual feedwater control and individual operator experience to mitigate SG level oscillations.

CORRECTIVE ACTIONS

Procedures SYS AE-121, "Turbine Driven Main Feedwater Pump Startup," and GEN 00-003, "Hot Standby to Minimum Load," were revised to incorporate the optimal range for operating the main feedwater regulating valves and regulating bypass valves. This range will provide the most consistent valve flow response for a given change in valve position.

A more appropriate startup and shutdown sequence will be determined to remain within the design operational parameters of the feedwater preheating system and Feedwater bypass valves. Appropriate procedures will be revised as needed.

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 SIMILAR OCCURRENCES

LER 2009-004-00 describes a turbine trip and FWIS actuation due to high water level in the 'A' SG while in Mode 3. The cause of the event was inadequate monitoring of critical operating parameters.

LER 2010-005-00 described a reactor trip at 100% power due low SG levels due to a trip of a main feedwater (MFW) pump. The failed transfer of an inverter to its alternate power supply caused the MFW pump trip.

LER 2010-006-00 described a reactor trip at 42% power due to a trip of the 'A' MFW pump. The control room operators manually tripped the reactor due to decreasing SG levels. The cause of the MFW pump trip was a failed servo in the MFW control circuitry.