WOLF CREEK
NUCLEAR OPERATING CORPORATION

Matthew W. Sunseri
Vice President Operations and Plant Manager

January 30, 2009

WO 09-0002

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

Reference: Letter WO 08-0025, dated October 3, 2008, from M. W. Sunseri, WCNOC, to the USNRC

Subject: Docket No. 50-482: Licensee Event Report 2008-008-01, Potential for Residual Heat Removal Trains to be Inoperable during Mode Change

Gentlemen,

The reference submitted Licensee Event Report (LER) 2008-008-00 described the potential for the Residual Heat Removal Trains to be inoperable during a change from Mode 4 to Mode 3. It was submitted pursuant to 10 CFR 50.73(a)(2)(i)(B) regarding an operation or condition prohibited by Technical Specifications at Wolf Creek Generating Station.

The enclosed LER 2008-008-01 is being submitted because further evaluation was conducted which provides additional detail to the safety significance and the root cause of the event. LER 2008-008-01 supersedes LER 2008-008-00 in its entirety.

This letter contains no commitments. If you have any questions concerning this matter, please contact me at (620) 364-4008, or Mr. Richard D. Flannigan, Manager Regulatory Affairs at (620) 364-4117.

Sincerely,

M W Sunseri

Matthew W. Sunseri

MWS/rlt

Enclosure

cc: E. E. Collins (NRC), w/e
V. G. Gaddy (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: 50 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
Potential for Residual Heat Removal Trains to be Inoperable during Mode Change

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
08	04	2008	2008	- 008 -	01	01	30	2009	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

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 August 4, 2008, during a review for potential void formation to address Generic Letter 2008-01, a concern with the Residual Heat Removal (RHR) system during Mode 4 and Mode 3 was identified.

Wolf Creek Generating Station typically will line up one or both of the RHR trains to the Reactor Coolant System (RCS) for shutdown cooling and secure them prior to entering Mode 3 from Mode 4. System procedures require the RHR system to be cooled down, using the mini-flow line through the RHR heat exchanger, when it is lined up to injection mode. The physical location where the mini-flow piping connects to the suction of the RHR system does not sufficiently cool approximately 140 feet of the RHR suction line. The temperature in the RHR suction line can remain near 350 degrees F for several hours. The saturation pressure for this heated water can prevent the check valve from the Refueling Water Storage Tank (RWST) from opening, preventing flow to the suction of the RHR pumps during a Mode 3 Loss of Coolant Accident (LOCA).

A review of plant conditions during the startup of Wolf Creek from Refuel Outage 16 showed that the condition existed when changing from Mode 4 to Mode 3 on May 10, 2008.

17. NARRATIVE

PLANT CONDITIONS PRIOR TO EVENT:

MODE – 1
Power – 100

EVENT DESCRIPTION:

On August 4, 2008, during a review for potential void formation to address Generic Letter 2008-01, a concern with the Residual Heat Removal (RHR) system [EIIS Code: BP] during Mode 4 and Mode 3 was identified.

Wolf Creek typically will line up one or both of the RHR trains to the Reactor Coolant System (RCS) [EIIS Code: AB] for shutdown cooling and secure them prior to entering Mode 3 from Mode 4. System procedures require the RHR system to be cooled down using the mini-flow recirculation line, following alignment for Emergency Core Cooling System (ECCS) injection mode. The physical location at which the mini-flow piping connects to the suction of the RHR system prevents approximately 140 feet of RHR suction line between the RCS hot leg isolation valve and the mini-flow line location from being cooled below the vulnerable conditions.

If the RHR system is aligned to ECCS injection Mode with water temperature near 350 degrees F, the water will remain hot for considerable duration in the RHR suction piping. If a Mode 3 Loss of Coolant Accident (LOCA) were to occur and Safety Injection System (SIS) initiated [EIIS Code: JE], the RHR pump would start, resulting in lowering the pressure in the suction piping. This lowering of pressure will result in flashing the water into steam, depending upon the water temperature. As long as the saturation pressure in the RHR suction leg is higher than the static pressure from the Refueling Water Storage Tank (RWST), the check valve located in the supply line from the RWST, will not open and no injection will occur. As the RHR pump is started, the pressure in the RHR suction leg will decrease which will cause the hot pressurized water to flash, before the pressure reaches low enough to open the check valve. The steam void could extend to the pump suction and steam bind the pump.

A review of plant conditions during the startup of Wolf Creek from Refuel Outage 16 showed that the condition existed when changing from Mode 4 to Mode 3 on May 10, 2008. As a result, RHR was not operable as required per Technical Specification 3.5.2 and 3.5.3.

BASIS FOR REPORTABILITY:

Wolf Creek changed from Mode 4 to Mode 3 without ensuring the RHR system was operable. This event is reportable under 10 CFR 50.73(a)(2)(i)(B) as operation or condition prohibited by Technical Specifications and under 10 CFR 50.73(a)(2)(ii)(B) for being in an unanalyzed condition.

ROOT CAUSE:

The root cause of the failure to ensure RHR operability is the RHR system design was not adequate to support all modes of RHR operation without impacting each other.

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CORRECTIVE ACTIONS:

Operations procedures were revised to ensure, when RHR is aligned to ECCS injection mode, that the temperature of the RHR suction lines from the RCS isolation valve are within prescribed limits.

SAFETY SIGNIFICANCE:

At the time the condition was discovered, Wolf Creek was in Mode 1, at 100% power, and both RHR systems were operable. The potential existed that during a forced outage, or refueling outage, the RHR system would not have functioned if a LOCA were to occur in Mode 3 or 4, with the water in the RHR suction pipe above the prescribed temperature limits.

An evaluation was conducted to assess the safety significance of this event for various conditions. The pertinent conditions are addressed below.

Injection Phase Small Break LOCA

For the Injection Phase Small Break LOCA the results of the plant specific analysis demonstrates that the limits of 10CFR50.46 will not be exceeded if a small break LOCA occurs in Mode 3 or 4, provided that operator establishes flow from a high head centrifugal charging pump (CCP) within 10 minutes after event initiation. Break sizes smaller than 6 inches may require additional operator actions to depressurize the RCS or start additional SI pumps. However, more time is available for the operator to respond to accomplish the required actions and current procedures direct such activities. Therefore, a small break LOCA occurring in Mode 3 or Mode 4 can be successfully mitigated with flow available from one high head charging/safety injection (SI) pump in a timely manner and the RHR pumps need not be relied on for short-term LOCA consequences mitigation. As a result, a loss of the RHR injection capability during shutdown conditions would pose insignificant consequences.

Injection Phase Large Break LOCA Evaluation

As documented in WCAP-12476, a probabilistic risk assessment approach using the Structural Reliability and Risk Assessment (SRRA) has shown that, even with the increased reliance on operator actions and the reduction of available safety systems, the risk of core damage resulting from a large break LOCA is less in Mode 3 and Mode 4 than for Mode 1.

The SRRA analysis indicates that the probability of a large pipe break occurring at Mode 3 or Mode 4 conditions is significantly lower (by more than an order of magnitude) than at Mode 1 conditions. Given that the probability of a large break occurring at any conditions is quite small, and that the piping has had the proof test of full pressure integrity for the limiting scenarios, the possibility of a large break occurring during shutdown is extremely remote.

The relative risk was calculated as the ratio of the frequency of core damage in either Mode 3 or 4 to Mode 1. This resulted in mean relative risk probabilities of 1/34 for Mode 3 and 1/6.9 for Mode 4, when compared to the risk for large break LOCAs during Mode 1. The lower risk posed by a large break LOCA in Modes 3 and 4 provides justification that a thermal-hydraulic analysis of a large break LOCA is not warranted.

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Recirculation Phase LOCA Evaluation

A qualitative assessment was used for the potential loss of RHR capability during recirculation after successfully mitigating a 6" small break LOCA or less.

With only one high head CCP and one safety injection pump available, it would take about 3.3 hours to deplete the RWST water inventory to the low level set-point for ECCS sump recirculation. After this time, should the RHR recirculation capability be lost as a result of steam binding during a postulated small break LOCA, the operating CCP and/or SI pump would have no suction source from the RHR pump discharge. However, the minimum ECCS flow requirements for removing the decay heat in the core could be met by use of the reactor makeup water transfer pumps to draw water from the reactor makeup water storage tank (RMWST) and blend it with the highly borated water from the boric acid tank (BAT). With this alignment, up to 150 gpm makeup capability could be achieved and the operation could last 14 hours. According to plant specific emergency procedures, a flow rate of 150 gpm is more than sufficient to meet the minimum ECCS flow requirement for decay heat removal at approximately 30 hours after shutdown.

Both the emergency and abnormal response procedures provide guidance to deal with situations in which the ECCS sump recirculation is potentially lost. Some of the options include alternative make-up sources to provide the required core cooling flow as described above. In addition, the procedures provide guidance to dump steam to the condenser from the intact steam generators (S/G) or to dump steam using intact S/G ARVs, if RHR is not available. With the venting of the S/Gs, reflux condensation cooling would be established and reduce the amount of required ECCS make-up flow due to core boil-off. It also reduces RCS pressure thus minimizing break flow. This process could be continued indefinitely, provided a backup feedwater source is available.

Based on a review of the plant information during past outages (including forced outages), the earliest time that the RHR suction piping was vulnerable to steam flashing is about 6 hours after shut-down. Considering the additional 3.3 hours duration for safety injection following a small break LOCA event initiation and gradual decrease in decay heat generation in the core, a worst case make-up flow to the reactor vessel would be on the order of 210 gpm. However, because the S/Gs would be in service as discussed above, it is estimated that the actual RWST make-up requirement would be less than the 150 gpm required for long term core cooling because of reflux condensation. In start-up scenarios, decay heat would be much lower and thus require even less make-up.

Based on the preceding discussion, the loss of RHR capability during recirculation would not have a significant impact on the longterm core cooling because the operator is expected to take appropriate actions according to the OFN procedures to assure that a minimum ECCS flow is provided to make up the decay heat boil-off in the core. Thus, a coolable geometry as required by 10CFR50.46 can be maintained.

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Risk Significance Evaluation

The Wolf Creek Probabilistic Safety Assessment (PSA) shutdown model was used to obtain risk insights for the postulated conditions present during past refuel outages when two trains of RHR were non-functional in modes 3 and 4. Refuel 14 data provided the longest duration for the condition in question and was used to determine the most limiting delta Core Damage Frequency (CDF) and Incremental Conditional Core Damage Probability (ICCDP). PSA determined these values to be 1.91 E-04 (delta CDF) and 2.31 E-06 (ICCDP). The ICCDP calculation assumed a 106-hour duration for unavailability of both the injection and recirculation function of RHR.

OPERATING EXPERIENCE/PREVIOUS EVENTS:

None