WOLF CREEK
NUCLEAR OPERATING CORPORATION

Stephen E. Hedges
Site Vice President

January 3, 2012

WO 12-0002

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

Subject: Docket No. 50-482: Licensee Event Report 2011-011-00, "Inadequate Analysis Assumptions Resulting in Deficient Control Room Evacuation Procedure"

Gentlemen:

The enclosed Licensee Event Report (LER) is being submitted pursuant to 10 CFR 50.73(a)(2)(ii)(B) regarding an unanalyzed condition that could potentially affect post fire safe shutdown equipment at the Wolf Creek Generating Station.

Commitments contained in this LER have been stated on the attachment. If you have any questions concerning this matter, please contact me at (620) 364-4156, or Mr. Gautam Sen at (620) 364-4175.

Sincerely,

Stephen E. Hedges

SEH/rlt

Attachment
Enclosure

cc: E. E. Collins (NRC), w/a, w/e
J. R. Hall (NRC), w/a, w/e
N. F. O'Keefe (NRC), w/a, w/e
Senior Resident Inspector (NRC), w/a, w/e

IE22
NRR

LIST OF REGULATORY COMMITMENTS

The following table identifies those actions committed to by WCNOC in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments. Please direct questions regarding these commitments to Mr. Gautam Sen at (620) 364-4175.

REGULATORY COMMITMENTS

Regulatory commitment	Due
A complete review of the assumptions that are used in thermal hydraulic analysis SA-08-006, Rev. 2, will be performed to ensure that the assumptions are complete and accurate.	June 15, 2012

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 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, 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
Inadequate Analysis Assumptions Resulting in Deficient Control Room Evacuation Procedure

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
11	03	2011	2011	011	00	01	03	2012	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)
Gautam Sen, Manager Regulatory Affairs	(620) 364-4175

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 November 3, 2011, during the 2011 Triennial Fire Protection Inspection, it was determined that procedure OFN RP-017, "Control Room Evacuation," had two deficiencies. For a postulated fire in the control room, the procedure does not adequately protect the steam generators from overfill and does not adequately protect the pressurizer from filling to above 100% indicated water level, possibly causing the primary system to go solid.

The direct cause of the event is an inadequate analysis assumption translated into procedure OFN RP-017. When the event scenarios supporting OFN RP-017 were developed, the engineers who worked on the Post Fire Safe Shutdown analyses did not fully consider the potential adverse effect of automatic functions and improperly credited closure of the main steam isolation valves from the control room.

Procedure OFN RP-017 was revised to provide compensatory measures that would prevent overfill of either the steam generators or the pressurizer.

PLANT CONDITIONS AT THE TIME OF THE EVENT

Mode 1
100 percent power
No inoperable structures, components or systems contributed to this event.

DESCRIPTION OF THE EVENT

On November 3, 2011, during the 2011 Triennial Fire Protection Inspection, it was determined that procedure OFN RP-017, "Control Room Evacuation," had two deficiencies. For a postulated fire in the control room, the procedure does not adequately protect the steam generators [EIIS Code: SB] from overfill and does not adequately protect the pressurizer [EIIS Code: AB-PZR] from filling to above 100% indicated water level.

Procedure OFN RP-017 did not specify the need to isolate normal feedwater [EIIS Code: SJ] to prevent overfill of the steam generators and provided no guidance to mitigate normal feedwater from feeding the steam generators if the main steam isolation valves (MSIV) [EIIS Code: SB-V] fail to close. This could cause overfill of the steam generators and water admission to the turbine driven auxiliary feedwater (AFW) pump turbine [EIIS Code: BA-TRB]. Water admission to the turbine driven AFW pump turbine could cause loss of the pump, a component required for achieving safe hot shutdown. It was assumed, that for certain scenarios, the MSIVs would close using the "all-close" switches in the control room. The MSIVs are verified closed later in the procedure, at approximately 20 minutes. Preliminary thermal hydraulic analysis determined that at approximately 3 minutes after the reactor is tripped, if the main feed pumps [EIIS Code: SJ-P] are not tripped, and flow of feedwater into the steam generators is not thereby stopped, the steam generators could overfill, causing water to be admitted into the AFW pump turbine steam line resulting in potential damage to the pump.

Procedure OFN RP-017 controls water level in the pressurizer by throttling one of the four boron injection tank (BIT) outlet valves [EIIS Code: CB-V], EM HV8801B. A Safety Injection (SI) was not assumed to occur. While reviewing the thermal hydraulic analysis (SA-08-006, Rev. 2) scenarios supporting OFN RP-017, it was observed that the low pressurizer pressure SI set point was reached in some of the scenarios. Therefore, an SI signal was possible. If an SI occurs, all four of the BIT valves may open and throttling only one of the outlet valves may not prevent overfill of the pressurizer. The procedure does not provide guidance to mitigate train-A components from injecting water to the primary system due to a spurious or valid SI until late in the procedure. The train-A components could cause a pressurizer overfill condition prematurely and challenge the primary system pressure boundary.

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BASIS FOR REPORTABILITY

Since a Post Fire Safe Shutdown (PFSSD) issue is identified in which no or insufficient guidance is available to Operations personnel to readily mitigate the postulated fire induced equipment maloperation, the issue is considered reportable under 10 CFR 50.72(b)(3)(ii)(B) and 10 CFR 50.73(a)(2)(ii)(B) as an unanalyzed condition that significantly degrades plant safety.

Since procedure OFN RP-017 would not have provided Operations personnel with the most conservative actions, Wolf Creek Nuclear Operating Corporation is reporting this condition pursuant to 10 CFR 50.73(a)(2)(ii)(B) for any event or condition that resulted in the nuclear power plant being in an unanalyzed condition that significantly degraded plant safety.

ROOT CAUSE

The direct cause of the event is an inadequate analysis assumption translated into procedure OFN RP-017. When the event scenarios were developed in support of procedure OFN RP-017, the engineers who worked on the PFSSD analyses did not fully consider the potential adverse effect of automatic functions and improperly credited closure of the main steam isolation valves from the control room. Therefore, the most conservative assumptions were not translated into the procedure.

CORRECTIVE ACTIONS

The apparent cause evaluation (CR 00045442) for this issue has been reviewed with the PFSSD engineers in effort to ensure an understanding of the improper assumptions that were applied in the development of thermal hydraulic analysis SA-08-006, Rev. 2.

Procedure OFN RP-017 was revised to provide compensatory measures that would prevent overfill of either the steam generators or the pressurizer.

A complete review of the assumptions that are used in thermal hydraulic analysis SA-08-006, Rev. 2, will be performed to ensure that the assumptions are complete and accurate. This action will be complete by June 15, 2012.

SAFETY SIGNIFICANCE

This issue has low safety significance. There were no actual consequences since no fire has occurred in the control room that required evacuation. A fire in the control room of such magnitude and severity as to cause an evacuation and plant shutdown is extremely unlikely. Based on the Fire Hazards Analysis (E-1F9905), the combustible loading in the control room is low and interior finish materials meet or exceed the surface flammability requirements of applicable standards. Cables entering the control room are IEEE 383 rated. Large concentrations of cables in the control room trenches are protected with an automatic Halon extinguishing system, and automatic smoke detectors are located in the control cabinets and trenches.

OPERATING EXPERIENCE/PREVIOUS SIMILAR OCCURRENCES

LER 2010-003-00 reported a condition where a postulated fire induced hot short could have prevented operation of the train 'B' diesel generator if a fire occurred in the control room. This condition was due to an inadequate review of control room circuitry for impact on the PFSSD analyses following a control room fire.

LER 2010-008-00 reported a condition where a postulated fire in the control room could cause a flow imbalance in the Essential Service Water system and cooling flow to other essential components could be reduced to below the minimum required flow. This was caused by a latent design deficiency.