WOLF CREEK
NUCLEAR OPERATING CORPORATION

R. A. Muench
Vice President Technical Services

JAN 1 4 2002
ET 02-0005

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

Subject: Docket No. 50-482: Licensee Event Report 2001-001-00

Gentlemen:

The enclosed Licensee Event Report (LER) 2001-001-00 is being submitted, pursuant to 10 CFR 50.73(a)(2)(v), to identify a potential condition that could have resulted in the failure of equipment to adequately perform its safety function.

Commitments made by Wolf Creek Nuclear Operating Corporation in the enclosed LER are identified in the Attachment.

If you should have any questions regarding this submittal, please contact me at (620) 364-4034, or Mr. Tony Harris at (620) 364-4038.

Very truly yours,

RAMuench

Richard A. Muench

RAM/ CLS

Enclosure
Attachment

cc: J. N. Donohew (NRC), w/e; w/a
W. D. Johnson (NRC), w/e; w/a
E. W. Merschoff (NRC), w/e; w/a
Senior Resident Inspector (NRC), w/e; w/a

IE22

LIST OF COMMITMENTS

The following table identifies those actions committed to by Wolf Creek Nuclear Operating Corporation (WCNOC) in this document. Any other statements in this submittal are provided for information purposes and are not considered to be commitments. Please direct questions regarding these commitments to Mr. Tony Harris, Manager Regulatory Affairs at Wolf Creek Generating Station, (620) 364-4038.

COMMITMENT	Due Date/Event
A long-term resolution of this issue (which will include revision to the flooding calculation, and may include throttling the CST isolation valve) will be completed on or before September 27, 2002. The interim action of locking the valve in the throttled position will remain in place until long-term resolution of this condition is implemented.	9/27/2002

NRC FORM 366 (7-2001)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 7-31-2004
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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4. TITLE

Potential Submergence Of Safety-Related Equipment Due To an Inadequate Flooding Calculation.

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
11	16	2001	2001	-- 001 --	00				FACILITY NAME	DOCKET NUMBER

9. OPERATING MODE	1
10. POWER LEVEL	100

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

	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)	X	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)		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)		

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
Karl A. (Tony) Harris, Manager Regulatory Affairs	(620) 364-4038

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 EXPECTED SUBMISSION DATE)	X	NO				

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

On November 16, 2001, Wolf Creek Nuclear Operating Corporation (WCNOC) personnel identified an error in an internal flood calculation. The calculation under estimated the input flow rate from a pipe break in two Auxiliary Building rooms and over estimated the drain flow rate from these rooms. As a result of correcting these values in the calculation, the calculation results now predict a flood water level that would submerge safety-related equipment required to switch the suction of the Auxiliary Feedwater (AFW) pumps from the Condensate Storage Tank (CST) to the Essential Service Water (ESW) system. Evaluations of event scenarios indicate that flooding of safety-related AFW equipment could occur and/or the decay heat removal function of AFW would not be accomplished.

The isolation valve for this line has been throttled, thus reducing the input flow rate and the resultant flood water level to ensure components required to fulfill the AFW system safety function are not submerged.

The cause for including these incorrect assumptions in this calculation is indeterminate due to the historical nature of this condition (the calculation was performed by an Architect/Engineer's staff in 1986). Based on a sampling review of other calculations, this issue is believed to be an isolated case.

This condition is considered to be of minimal safety significance.

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

Background:

The Auxiliary Feedwater (AFW) system [EIIS Code: BA] automatically supplies feedwater to the steam generators to remove decay heat from the Reactor Coolant System [EIIS Code: AB] upon the loss of normal feedwater supply. The AFW pumps normally take suction through a common suction line from the non-safety related Condensate Storage Tank (CST) [EIIS Code: KA]. Should the CST level decrease below a preset value, the AFW suction is automatically swapped to the safety-related Essential Service Water (ESW) system [EIIS Code: BI]. In addition to providing the normal source of water for the AFW system, the CST provides the source of make-up water to the main condenser hotwell for normal operations through a 10" pipe (AP-007-HBD-10"). Manual butterfly valve APV012 (Condensate Storage Tank Outlet Isolation Valve) provides capability to isolate the CST from the Main Condenser. Sections of this pipe are located in Auxiliary Building room 1207, which is connected with room 1206. These rooms are pipe chases and contain the safety-related pressure transmitters and motor operated valves that function to provide automatic switch-over of the AFW pump suction from the CST to the ESW system, as well as various Turbine-Driven AFW pump steam drain trap components.

Per 10 CFR 50 Appendix A, General Design Criteria (GDC) 4, flood calculations to determine the effects of internal pipe failures at Wolf Creek have been performed. These calculations consider parameters including input flow rate, drain flow rate, and resultant flood water level. The results of these calculations are used in the design process to ensure adequate protection is provided so that those portions of essential structures, systems, or components whose failure could compromise the integrity of the reactor coolant system or reduce the functioning of any plant feature required for a post-accident safe shutdown are designed, constructed, and protected so that they do not fail or cause such a failure.

Plant Conditions Prior to the Event:

MODE – 1
Power -- 100 percent
Normal Operating Temperature and Pressure

Event Description

On November 16, 2001, in the process of providing resolution to an issue raised during a routine NRC engineering inspection, Wolf Creek Nuclear Operating Corporation (WCNOC) Engineering personnel determined that the basis in calculation FL-04 for the flood rate into Auxiliary Building rooms 1206 and 1207 was incorrect. The assumed input into the rooms was based on the CST to main condenser hotwell make-up flow rate. However, the calculation should have been based on the potential flow rate given a Safe Shutdown Earthquake and subsequent double-ended guillotine break of the CST to hotwell make-up line. Subsequently, WCNOC personnel determined that the drain rate assumed in the calculation was also incorrect in that elevation differences were used, instead of drainpipe flow capacity. Rooms 1206 and 1207 are connected and contain multiple trains of safety-related equipment associated with providing automatic swap-over of the suction for the AFW pumps from the CST to the ESW system supply. The affected equipment is not qualified for submergence.

Subsequent evaluation of the condition indicated that the input flow rate should have been approximately 5600 gpm, which is well above the 3500 gpm assumed in calculation FL-04. The drain flow rate should have been approximately 1700 gpm, which is well below the 3940 gpm assumed in calculation FL-04. Based on the new values, the resultant flood level in rooms 1206 and 1207 would be approximately seven (7) feet. This

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level would submerge the AFW low suction pressure transmitters (ALPT037, ALPT038, ALPT039), located approximately two (2) feet above floor level; the AFW to ESW swap-over valves (ALHV030, ALHV031, ALHV032, ALHV033, ALHV034, ALHV035, ALHV036), located approximately five (5) feet above floor level; and various Turbine-Driven AFW pump steam drain trap components installed approximately two and one-half (2.5) feet above floor level.

Upon discovery of this condition, an isolation valve in the CST to Condenser Hotwell make-up line, APV012, was throttled open and locked in the throttled position. The throttled valve will reduce the input flow rate into the rooms from a break of the CST to hotwell make-up line and will limit the resultant flood water level in the subject rooms, thus ensuring components required to fulfill the AFW system safety function are not affected.

Basis for Reportability:

This condition is reportable under 10 CFR 50.73(a)(2)(v), which states that licensees shall report "any event or condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to: (A) Shut down the reactor and maintain it in a safe shutdown condition; (B) Remove residual heat; (C) Control the release of radioactive material; or (D) Mitigate the consequences of an accident." 10 CFR 50.73(a)(2)(vi) states that "events covered in paragraph (a)(2)(v) of this section may include one or more procedural errors, equipment failures, and/or discovery of design, analysis, fabrication, construction, and/or procedural inadequacies. However, individual component failures need not be reported pursuant to paragraph (a)(2)(v) of this section if redundant equipment in the same system was operable and available to perform the required safety function."

The condition reported is associated with flooding of Auxiliary Building rooms 1206 and 1207. The flooding calculations associated with these rooms contained errors. Re-evaluation of the resultant flood levels indicates that the AFW system would be affected such that its safety function to remove residual heat would not be fulfilled. Thus, the condition meets the reporting requirements of 10 CFR 50.73(a)(2)(v).

In addition, the condition reported meets the corresponding 10 CFR 50.72 reporting requirement, 10 CFR 50.72(b)(3)(v). On November 16, 2001, an 8-hour 10 CFR 50.72 report was made (NRC event number 38509).

Root Cause:

The root cause for use of incorrect values in the calculation is indeterminate. These errors date back to February 1986 and the responsible Architect/Engineering personnel are no longer associated with Wolf Creek Generating Station. Without additional information from the Architect/Engineer, a specific cause for the calculation errors cannot be determined. This information is unavailable to WCNOC. Based on review of a sampling of other flooding calculations, WCNOC believes that this is an isolated case.

Corrective Actions:

The CST outlet isolation valve, APV012, was locked in a throttled open position based on evaluations made with the vendor's valve sizing software and consideration of the drain system capacity. Revision 23 to WCNOC procedure AP 21G-001, "Control of Locked Component Status," was issued on January 4, 2002, to reflect the valve's locked throttled position. This locked throttled position will have no adverse impact on the non safety-related condenser hotwell make-up function or any other aspect of plant operation.

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A sample of other flooding calculations was reviewed. No errors similar to those found in this calculation were found.

A long-term resolution of this issue (which will include revision to the flooding calculation, and may include throttling the CST isolation valve) will be completed on or before September 27, 2002. The interim action of locking the valve in the throttled position will remain in place until the long-term resolution of this condition is implemented.

Safety Significance:

The safety significance of this condition is considered minimal. This is because the consequences of a flooding event due to either random or consequential failure of line AP-007-HBD-10" in Auxiliary Building rooms 1206 and 1207 is not significant from a core damage frequency (CDF) perspective. In addition, the consequences of the postulated flooding event are not significant from a Large Early Release Frequency (LERF) perspective. Specific LERF estimates were not determined since the likelihood of proceeding to a Core Damage end state is low, and since flooding of components in this area will not affect the containment heat removal or containment isolation safety functions.

An evaluation was performed to determine the annual CDF due to failure of the subject line (AP-007-HBD-10") with subsequent flooding of Rooms 1206 and 1207. Failure of the line was considered from two perspectives: first, random failure of the line causing an initiating event; and second, consequential failure of the line due to an initiating event. A random failure of the subject line is postulated to result in a "Transient Without Power Conversion System Available" type of initiating event due to loss of condenser vacuum, or other secondary side failure associated with an opening in line AP-007-HDB-10".

The consequential failure of the line was postulated to result from a seismic initiator. The evaluation results were:

1) Core damage frequency considering the random failure of the CST to hotwell make-up line: 8.4E-08 per year.

2) Core damage frequency considering the consequential failure of the CST to hotwell make-up line from a seismic initiator: 5.8E-09 per year.

The scenarios associated with this condition are based upon a seismic event that results in the failure of the non-seismic 10-inch make-up line, but does not impact the structural integrity of the non-seismically qualified CST. In the unlikely scenario that a seismic event of this magnitude should occur, the failure of the 10-inch line would most likely be accompanied by a structural failure of the CST. Due to the structural failure of the CST, rooms 1206 and 1207 would not flood and automatic swap-over of the AFW pump suction from the CST to the ESW system would occur on low AFW pump suction pressure as designed. In this case, where the CST fails, the AFW safety function would remain functional.

The calculation errors have not resulted in any actual event or condition. No system, structure or component has been adversely affected. Furthermore, plant operations have not been adversely affected.

Should the initiating event occur with the CST level at the minimum allowed by Technical Specifications (281,000 gallons), there would not be sufficient water in the CST to fulfill the AFW safety function during cool down to conditions allowing entry into Residual Heat Removal (RHR) cooling. Initially, CST volume would be

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reduced through the suction to the AFW pumps and through the failed CST to condenser hotwell make-up line. During this time, the flood water level in rooms 1206 and 1207 would submerge the suction pressure transmitters needed to provide automatic switching from the CST to the ESW system. After CST level decreases below the level of the hotwell make-up line, the floor drains in rooms 1206 and 1207 would remove the water to a level that will allow plant operators to perform manual switching the suction of the AFW pumps from the CST to the ESW system. However, these operator actions are not credited in the design bases. In the absence of these actions, there would be insufficient water in the CST to fulfill the AFW safety function during cool down to entry into Residual Heat Removal (RHR) cooling.

Should the initiating event occur with the CST at the high end of the normal operating band, sufficient water would be available to provide cool down to RHR entry conditions without switching suction of the AFW system to the ESW system. In this case, the safety function of the AFW system would not be lost.

A review was performed to ensure that the resultant flood water level in Auxiliary Building rooms 1206 and 1207 of seven (7) feet would not adversely impact the function of the structure. Based on the weight of the maximum volume of water, WCNOC has concluded, based on engineering judgment, that the resultant flooding would not cause the structural failure of floors or walls.

Previous Occurrences:

A review of Licensee Event Reports (LERs) associated with calculation errors from 1996 though this event date was performed. No relevant LERs were identified.