WOLF CREEK
NUCLEAR OPERATING CORPORATION

Terry J Garrett
Vice President, Engineering

June 14, 2005
ET 05-0006

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

Subject: Docket No. 50-482: Licensee Event Report 2005-002-00, Reactor Coolant System Pressure Boundary Leakage Due to Small Cracks in Steam Generator Lower Head Bowl Drain Lines

Gentlemen:

The enclosed Licensee Event Report (LER) 2005-002-00 is being submitted pursuant to 10 CFR 50.73(a)(2)(ii)(A) regarding Reactor Coolant System leakage at Wolf Creek Generating Station.

Wolf Creek Nuclear Operating Corporation has made no commitments in the enclosed LER. If you have any questions concerning this matter, please contact me at (620) 364-4084, or Mr. Kevin Moles at (620) 364-4126.

Very truly yours,

Terry J. Garrett

TJG/rlg

Enclosure

cc: J. N. Donohew (NRC), w/e
W. B. Jones (NRC), w/e
B. S. Mallett (NRC), w/e
Senior Resident Inspector (NRC), w/e

IE22

NRC FORM 366 (6-2004) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 06/30/2007

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

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4. TITLE

Reactor Coolant System Pressure Boundary Leakage Due to Small Cracks in Steam Generator Lower Head Bowl Drain Lines

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
									N/A	05000
04	15	2005	2005	- 002 -	00	06	14	2005	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE: 6

10. POWER LEVEL: 0%

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)*
Kevin J. Moles, Manager Regulatory Affairs	(620) 364-4126

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
B	AB	SG	W120	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 April 15, 2005, during Wolf Creek Generating Station (WCGS) refueling outage number 14, during performance of Alloy 600 bare metal visual examinations, the presence of boric acid deposits indicated a leak in a weld in the steam generator (SG) D lower head bowl drain line. Subsequently on April 20, 2005, boric acid deposits indicating a leak in a weld in the SG C lower head bowl drain line were discovered. The Reactor Coolant System (RCS) leakage constitutes degradation of a principal safety barrier and is considered reportable to the requirements of I0 CFR 50.73(a)(2)(ii)(A), "The condition of the nuclear power plant, including its principal safety barriers, being seriously degraded." SG C and D bowl drain leaks were subsequently repaired, examined and tested. SG A and B bowl drains were also repaired similar to the SG C and D bowl drains as a preventive measure. The completed weld repairs were all found acceptable.

The pressure boundary leakage path is suspected to be the nozzle coupling to vessel weld. The most likely technical cause of the SG bowl drain leak is primary water stress corrosion cracking (PWSCC).

The safety significance of this event is low. Although welds of the steam generator bowl drains may be susceptible to PWSCC that can result in small leaks, industry experience with PWSCC shows that complete failure of the weld joints is considered to be very unlikely.

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

Background:

Wolf Creek Generating Station (WCGS) is a Westinghouse Pressurized Water Reactor [EIIS: RCT] with four steam generators (SG) [EIIS: SG] connected to the reactor coolant system (RCS) [EIIS: AB]. The SGs are Westinghouse Model F SGs. The four SGs are vertical shell and U-tube evaporators with integral moisture separating equipment. The reactor coolant flows through the inverted U-tubes, entering and leaving through the nozzles [EIIS: NZL] located in the hemispherical bottom head of the SG. The bottom head is divided into inlet (hot leg) and outlet (cold Leg) chambers by a vertical partition plate extending from the head to the tube sheet. However, there is a small semi-circular hole at the center bottom of this plate to allow draining the bowl through one common drain line. The SG bottom head is primarily carbon steel. The primary side of the tube sheet is clad with Inconel, and the interior surfaces of the reactor coolant channel head and nozzles are clad with austenitic stainless steel. The SG bowl drains are located in the center of the lower channel head. The opening to the SG bowl is beneath the partition plate that separates the hot leg side of the channel head from the cold leg side. A small passage in the partition plate above the drain hole connects the bowl drain to the hot and cold leg channel heads simultaneously. The bowl drain was constructed by hard roll expanding an Alloy 600 sleeve into a clearance hole through the generator shell. The sleeve was seal welded to the stainless steel bowl cladding at the inner surface of the bowl. The lower end of the sleeve was seal welded to a butter layer of Alloy 82/182. A stainless steel coupling was welded below the sleeve termination to form the bowl drain nozzle on the SG outer shell. The coupling was welded to the butter layer using Alloy 82/182 filler material. A gap was left between the lower end of the sleeve and top end of the coupling to compensate for thermal expansion during welding.

Plant Conditions Prior to the Event:

Mode – 6
Power – 0%
RCS temperature was approximately 90 degrees F at atmospheric pressure.

Event Description:

On April 15, 2005, during Wolf Creek Generating Station (WCGS) refueling outage number 14, Alloy 600 inspection program examinations were being performed. Bare metal visual examinations discovered the presence of boric acid deposits indicating a leak in a weld in the steam generator (SG) D lower head bowl drain line. A subsequent visual examination of SG C on April 20, 2005, revealed boric acid deposits indicating a leak in a weld in the SG C lower head bowl drain line. Liquid penetrant testing was performed on SG A and SG B lower head bowl drains and no relevant indications were identified.

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

Basis for Reportability:

This event is being reported to the NRC pursuant to 10 CFR 50.73(a)(ii)(A), "any event or condition that resulted in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded." Conditions that represent welding or material defects in the primary coolant system which cannot be found acceptable under ASME Section XI standards are reportable to this criterion. This event is also being reported pursuant to 10 CFR 50.73 (a)(2)(i)(B), "any operation or condition prohibited by the plant's Technical Specifications" and 10 CFR 50.36 (c)(2)(i), limiting condition for operation of a nuclear reactor not met. Technical Specification (TS) Limiting Condition for Operation (LCO) 3.4.13(a) limits RCS Operational Leakage to "No pressure boundary LEAKAGE" while in MODES 1 through 4. Condition B of TS 3.4.13 requires that if pressure boundary leakage exists, the unit is to be in MODE 3 within 6 hours and to be in MODE 5 within 36 hours. Because leakage of reactor coolant through the SG channel head bowl drains was very minimal, it was not detected during power operation. Leakage was only detected after shutdown by the visual observation of a small quantity of boric acid crystals. Therefore, WCGS operated in a condition prohibited by Technical Specifications.

Root Cause:

The most likely technical cause of the through wall leak is primary water stress corrosion cracking (PWSCC). This is due to the original weld filler material being Alloy 82/182, which has been shown to be susceptible to PWSCC. This phenomenon may occur when Alloy 82/182 is exposed to a combination of contact (wetted surface) with the primary system water, sufficient stress and elevated temperatures. The crack likely formed at the gap between the stainless steel coupling and the Alloy 82/182 overlay weld filler material and eventually surfaced in the middle of the large fillet weld.

Corrective Actions:

The following corrective actions were taken on SGs C and D, and the same preventive measures were also taken on SGs A and B. All of the Alloy 82/182 weld filler material in the S/G bowl drain area that could be exposed to primary water was either machined out or isolated using materials not susceptible to PWSCC. Following machining out the cavity containing the Alloy 82/182 material, a seal weld was applied to the end of each bowl drain liner (tube) using Alloy 52 weld material. The coupling nozzle was then tack welded in place and the remainder of the cavity was filled with Alloy 52 weld material to the designed final depth.

Safety Significance:

Numerous industry analyses have documented the low safety significance of PWSCC. The driving stresses for PWSCC crack initiation and growth in primary system nozzle welds are usually residual stresses from welding. Industry experience with several types of nozzle welds also shows axial cracks to occur more frequently than circumferential cracks.

When axial PWSCC cracks occur in pipe or pipe to nozzle welds, they are contained within the susceptible weld material. They only grow across the width of the weld and do not extend into the base metal. Such short axial cracks may cause leaks but cannot cause structural failures.

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

Safety Significance (Con't):

Though less likely, PWSCC can cause circumferential cracks. The residual stresses that drive PWSCC are very unlikely to be uniform around the circumference of the weld. These variations in stress and microstructure through the weld volume usually result in cracks growing at different rates in different locations so that a small circumferential length of the crack grows through wall causing a leak before the cracking affects the structural integrity of the weld joint. Therefore, sudden rupture of a steam generator bowl drain attachment weld is considered to be very unlikely.

Industry experience shows growth of a PWSCC crack is slow, and there would be considerable time available for detection of a leak within the range of installed leakage detection allowing for orderly plant shutdown to repair the weld. The likelihood of failure to detect and appropriately respond to a small leak, before it might propagate to allow significant RCS leakage, is very low.

Previous Occurrences:

A search of internal operating experience did not identify any previous occurrences of cracks in Alloy 600/82/182 material due to PWSCC at WCGS.