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

**LER 311 / 05 – 002 – 00**
**Salem Generating Station Unit 2**
**Facility Operating License DPR- 75**
**Docket No. 50-311**

This Licensee Event Report (LER) entitled "Reactor Coolant Instrument Line Through-Wall Leak" is being submitted pursuant to the requirements of 10CFR 50.73(a)(2)(ii)(A).

The attached LER contains no commitments.

Should there be any questions regarding this matter please contact Howard Berrick at 856-339-1862.

Sincerely,

C. Fricker
Salem Plant Manager

Attachment

HGB

C Distribution
LER File 3.7

IE22

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

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

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.

# LICENSEE EVENT REPORT (LER)

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

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**4. TITLE**

Reactor Coolant Instrument Line Through-Wall Leak

| 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 |
| 4 | 4 | 2005 | 2005 | 002 | 00 | 6 | 3 | 2005 | 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)
- ☐ 20.2201(d)
- ☐ 20.2203(a)(1)
- ☐ 20.2203(a)(2)(i)
- ☐ 20.2203(a)(2)(ii)
- ☐ 20.2203(a)(2)(iii)
- ☐ 20.2203(a)(2)(iv)
- ☐ 20.2203(a)(2)(v)
- ☐ 20.2203(a)(2)(vi)
- ☐ 20.2203(a)(3)(i)
- ☐ 20.2203(a)(3)(ii)
- ☐ 20.2203(a)(4)
- ☐ 50.36(c)(1)(i)(A)
- ☐ 50.36(c)(1)(ii)(A)
- ☐ 50.36(c)(2)
- ☐ 50.46(a)(3)(ii)
- ☐ 50.73(a)(2)(i)(A)
- ☐ 50.73(a)(2)(i)(B)
- ☐ 50.73(a)(2)(i)(C)
- ☒ 50.73(a)(2)(ii)(A)
- ☐ 50.73(a)(2)(ii)(B)
- ☐ 50.73(a)(2)(iii)
- ☐ 50.73(a)(2)(iv)(A)
- ☐ 50.73(a)(2)(v)(A)
- ☐ 50.73(a)(2)(v)(B)
- ☐ 50.73(a)(2)(v)(C)
- ☐ 50.73(a)(2)(v)(D)
- ☐ 50.73(a)(2)(vii)
- ☐ 50.73(a)(2)(viii)(A)
- ☐ 50.73(a)(2)(viii)(B)
- ☐ 50.73(a)(2)(ix)(A)
- ☐ 50.73(a)(2)(x)
- ☐ 73.71(a)(4)
- ☐ 73.71(a)(5)
- ☐ OTHER

Specify in Abstract below or in NRC Form 366A

**12. LICENSEE CONTACT FOR THIS LER**

| FACILITY NAME | TELEPHONE NUMBER (Include Area Code) |
|---|---|
| Howard G. Berrick, Senior Licensing Engineer | 856-339-1862 |

**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 |
|---|---|---|---|---|---|---|---|---|---|
| A | AB | TBG | - | No | | | | | |

**14. SUPPLEMENTAL REPORT EXPECTED**

☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)* ☒ NO

**15. EXPECTED SUBMISSION DATE**

| MONTH | DAY | YEAR |
|---|---|---|
| | | |


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

On April 4, 2005, during a Boric Acid program field walkdown, an accumulation of boric acid was identified on ⅜-inch instrument tubing, low-pressure side, associated with 21 Reactor Coolant Pump (RCP) flow transmitter 2FT416. The 2FT416 instrument tubing line was field examined in an effort to assess structural integrity of the tube. The low-pressure tubing was found to possess an accumulation of crystallized dried boric acid, which indicates that the leak was relatively new. The boric acid was removed. A tubing tray pressure plate restraint was then removed, revealing a corrosion point spot. The removed pressure plate had evidence of corrosion at the same location as the corroded point spot on the tubing. Extensive walkdowns of instrumentation tubing in containment revealed evidence of additional instrumentation tube leakage. Corrective actions included replacement of instrumentation tubing, tightening loose fittings, swiping, sampling and cleaning, as necessary.

This event is reportable in accordance with 10CFR50.73(a)(2)(ii), "Any event or condition that resulted in: (A) The condition of the nuclear power plant, including its principal safety barriers, being seriously degraded".

# LICENSEE EVENT REPORT (LER)

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

## PLANT AND SYSTEM IDENTIFICATION

Westinghouse - Pressurized Water Reactor
Reactor Coolant System (PWR) - {AB/TBG}*
* Energy Industry Identification System (EIIS) codes and component function identifier codes appear in the text as {SS/CCC}.

## IDENTIFICATION OF OCCURRENCE

Event Date: April 4, 2005
Discovery Date: April 4, 2005

## CONDITIONS PRIOR TO OCCURRENCE

Salem Unit 2 was in Mode 1 (POWER OPERATION) at approximately 100% power at the time of the event. No structures, systems or components were inoperable at the time of the occurrence that contributed to the event.

## DESCRIPTION OF OCCURRENCE

On April 4, 2005, during a Boric Acid program field walkdown, an accumulation of boric acid deposits was identified on instrument tubing associated with 21 RCP flow transmitter 2FT416 low-pressure side {AB/TBG}. The Technical Specification (TS) 3.4.11.1.b for ASME Code Class 2 components was not entered, as it was not evident that structural integrity had been impacted. The 2FT416 instrument tubing line was field examined in an effort to assess structural integrity of the tubing. The low-pressure tubing was found to possess an accumulation of dry boric acid. The dry boric acid was white with some evidence of brownish color observed. The boric acid's appearance was crystallized indicating it was relatively new. There was no evidence of any other sources of leakage nearby the affected tubing.

A tubing tray pressure plate restraint was unbolted and removed, revealing a corroded point spot on the top tubing approximately 0.500 inch from the edge of the right side of the plate. The boric acid was removed. The removed pressure plate revealed corrosion of the plate at the same location of the corroded point spot on the tubing. The corrosion on the plate demonstrated evidence of corrosion with no appreciable loss of plate thickness. The subsequent structural integrity evaluation determined that the 2FT416 low-pressure side instrumentation tubing, in its current condition was acceptable, and would be monitored while the unit was taken offline on April 5, 2005 for the 2R14 refueling outage. NRC notification was not made.

Subsequent 2R14 walk-downs of all Reactor Coolant System (RCS) flow transmitter tubing, accumulator sensor tubing and pressurizer instrument tubing were performed. Two additional leaks were discovered. One was at the 22 RCP flow transmitter instrumentation line to 2FT424. The other was a loose fitting at a block valve to the 23 RCP flow transmitter 2FT435. The affected lines were un-insulated.

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

## DESCRIPTION OF OCCURRENCE (contd)

Accumulation of dried boron on the outside of the tubing was the only indication of leakage. The affected tubing is stainless steel tubing that contains reactor coolant. The tubing outside diameter for those RCS instrument lines in question is 0.375 inches. The affected lines are ASME Code Class 2 components and are designed to maintain the RCS pressure boundary.

This event is being reported pursuant to 10CFR50.73(a)(2)(ii) as a condition that resulted in the nuclear power plant, including its principal safety barriers, being seriously degraded. Specifically, the through-wall leak is considered a degradation of the RCS pressure boundary.

## CAUSE OF OCCURRENCE

The failure mechanism of the tubing is believed to be chloride-induced transgranular stress corrosion cracking initiated on the outside diameter surface. The presence of local stresses due to point contact with the pressure plate and the presence of contaminants on the outside surface of the tubing contributed to the cracking. The source of the chloride contamination was determined to be from historic Containment Fan Coil Unit Service Water leakage events. The structural integrity evaluation that concluded the condition of the 2FT416 low-pressure side instrumentation tubing was acceptable was inadequate.

## PREVIOUS OCCURRENCES

Salem and Hope Creek Generating Station LERs for years 2002 through 2005 were reviewed for similar occurrences of stress corrosion cracking of RCS tubing and none were noted. There was, however, a similar occurrence in 1998.

LER 311/98-007, originally issued August 27, 1998 and supplemented on January 28, 1999, was titled Reactor Coolant Instrument Through-wall Leak. The apparent cause for this event was transgranular stress corrosion cracking initiated on the outside diameter surface. The corrective actions included inspection and swiping of tubing, cleaning based on swipe results, and the replacement of approximately seven tubes. The corrective actions taken, though appropriate, may have been limited.

## SAFETY CONSEQUENCES AND IMPLICATIONS

There were no actual safety consequences associated with this event.

UFSAR Section 15.3.1.1 describes a rupture of small diameter piping. A single centrifugal charging pump could accommodate an RCS rupture approximately equal to a 0.375-inch diameter hole. With a rupture of this size, the operational level in the Pressurizer could be maintained, permitting the operator to execute an orderly shutdown. The 0.375-inch instrument tubing has a wall thickness of 0.065 inches, for an internal diameter of 0.245 inches. The area from a break in the 0.375-inch tubing is below the 0.375-inch diameter analyzed in Chapter 15 of the UFSAR and would, therefore, be well within the capability of a single centrifugal charging pump. The release of radioactivity would not be expected to exceed previously analyzed dose limits. The affected tubing lines are capable of being isolated from the RCS by a manual root valve located within the bioshield area.

## SAFETY CONSEQUENCES AND IMPLICATIONS (contd)

The RCS flow instruments are designed with a common high-pressure tap and individual low pressure taps. A failure of a low-pressure tap would cause the affected channel to fail high, leaving two channels available for the reactor protection function. No indications of RCS flow anomalies were noted. The safety function of providing a reactor trip on a low RCS flow condition was not impacted by this event.

Based on the above, there was no impact to the health and safety of the public.

A review of this event determined that a Safety System Functional Failure (SSFF) as defined in the Nuclear Energy Institute (NEI) 99-02 did not occur.

## CORRECTIVE ACTIONS

1. All tubing sections identified to have through wall leakage were replaced during the Salem 2R14 refueling outage.
2. The RC flow transmitter tubing from the instrument root valves to the transmitters, and all drain piping associated with the accumulator drain valves were cleaned to acceptable chloride levels.
3. A chloride contamination control program will be established to preclude events of this nature.
4. An evaluation of the structural integrity assessment and its impact on reporting requirements is in progress. Additional corrective actions will be taken at the completion of the evaluation.

## COMMITMENTS

This LER contains no commitments.