**PSEG Nuclear LLC**
P.O. Box 236, Hancocks Bridge, NJ 08038-0236

PSEG
Nuclear LLC

NOV 0 9 2011

10CFR50.73

LR-N11-0339

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington DC 20555-001

Supplemental LER 311/2011-005-01
Salem Nuclear Generating Station Unit 2
Facility Operating License No. DPR-75
NRC Docket No. 50-311

Subject: Completion of a Plant Shutdown in Accordance With Technical Specification 3.0.3

This Licensee Event Report, "Completion of a Plant Shutdown in Accordance With Technical Specification 3.0.3" is being submitted pursuant to the requirements of the Code of Federal Regulations 10CFR50.73 (a)(2)(i)(A).

The attached supplemental LER contains no commitments. Should you have any questions or comments regarding this submittal, please contact Mr. B. Thomas at 856-339-2022.

Sincerely,

Carl J. Fricker
Site Vice President - Salem

Attachments (1)

IE22
NRR

NOV 0 9 2011

cc

Mr. W. Dean, USNRC - Administrator - Region I
Mr. R. Ennis, USNRC - Licensing Project Manager - Salem
USNRC Senior Resident Inspector - Salem (X24)
Mr. P. Mulligan, NJBNE Manager IV
Mr. H. Berrick, Salem Commitment Tracking Coordinator
Mr. L. Marabella, Corporate Commitment Tracking Coordinator

NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION
(9-2007)

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 10/31/2013



# LICENSEE EVENT REPORT (LER)

| 1. FACILITY NAME | 2. DOCKET NUMBER | 3. PAGE |
|---|---|---|
| Salem Generating Station - Unit 2 | 05000311 | 1 of 5 |

4. TITLE Completion of a Plant Shutdown in Accordance With Technical Specification 3.0.3

| 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 |
| 07 | 14 | 2011 | 2011 | 0 0 5 | 1 | 11 | 09 | 2011 | | 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)(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) |
|---|---|
| Brian J. Thomas, Principal Engineer | (856) 339 -2022 |

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 |
|---|---|---|---|---|---|---|---|---|---|
| E | BQ | RV | C710 | 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)*

This report is being made in accordance with 10CFR50.73 (a)(2)(i)(A) "The completion of any nuclear plant shutdown required by the plant's Technical Specification."

At 2038 on July 14, 2011, during the performance of the Salem Unit 2 Emergency Core Cooling System (ECCS) fill and vent monthly surveillance test, a leakage path was identified from the Boron Injection Tank (BIT) relief valve 2SJ10 piping. Investigation determined that the leakage was approximately 15 gpm due to lifting of the 2SJ10 relief valve. Upon closure of the BIT inlet valve 2SJ4 to isolate the flow through the relief valve, a through wall socket weld crack developed on the 2SJ10 relief valve inlet piping. The BIT is part of the flow path of the high head safety injection system. Without the high head safety injection flow path operable, Technical Specification (TS) 3.0.3 was entered.

The 2SJ10 relief valve lifting was the result of missed opportunities to correct a component design application deficiency. The socket weld cracking was determined by laboratory analysis to be caused by a combination of fatigue and transgranular stress corrosion cracking. The affected 2SJ10 piping has been replaced with a new straight run of pipe and the relief valve internals were replaced. Design changes will be performed to resolve 1/2SJ10 operating margin, eliminate unnecessary non-isolatable socket welds in the BIT system and replace the remaining non-isolatable socket welds in the BIT system with low carbon weld filler and install a stiffener on the 2SJ10 relief valve piping.

NARRATIVE

## PLANT AND SYSTEM IDENTIFICATION

Westinghouse – Pressurized Water Reactor (PWR/4)

Emergency Core Cooling System / Relief Valve {BQ/RV}

* Energy Industry Identification System {EIIS} codes and component function identifier codes appear as {SS/CCC}

## IDENTIFICATION OF OCCURRENCE

Event Date: July 14, 2011

Discovery Date: July 14, 2011

## CONDITIONS PRIOR TO OCCURRENCE

Salem Unit 2 was at 100% power prior to the leakage from the Emergency Core Cooling System relief valve. No structures, systems or components were inoperable at the time of the discovery that contributed to the event.

## DESCRIPTION OF OCCURRENCE

At 2038 on July 14, 2011, during the performance of the Salem Unit 2 Emergency Core Cooling System (ECCS) fill and vent monthly surveillance test, a leakage path from the Boron Injection Tank (BIT) relief valve {BQ/RV} 2SJ10 piping was identified by Operations personnel. The magnitude of the leak was estimated to be approximately 15 gpm.

The leak was isolated by closing the BIT isolation valves. The BIT is part of the flow path of the high head safety injection system; therefore, with the identified leakage in the high head safety injection flow path, Operations declared both trains of the ECCS inoperable and Technical Specification (TS) 3.0.3 was entered.

TS 3.0.3 states " When a Limiting Condition for Operation is not met except as provided in the associated ACTION requirements, within one hour action shall be initiated to place the unit in a MODE in which the specification does not apply by placing it, as applicable, in:

1. At least HOT STANDBY within the next 6 hours,

2. At least HOT SHUTDOWN within the following 6 hours, and

3. At least COLD SHUTDOWN within the subsequent 24 hours.

NARRATIVE

## DESCRIPTION OF OCCURRENCE (cont'd)

The magnitude of the leak was estimated to be approximately 15 gpm.

On July 14, at 2053 Salem Operations declared an Unusual Event (UE) due to unidentified leakage greater than 10 gpm in accordance with the Salem Emergency Plan. Nuclear Regulatory Commission (NRC) notifications were performed in accordance with the Emergency Plan requirements.

At 0233 on July 15, 2011, Salem Unit 2 entered Mode 3 with all control rods inserted in the core and at 1259 Salem Unit 2 achieved Mode 5, Cold Shutdown.

The UE was terminated at 0339 on July 15, 2011, upon removal of power to the BIT isolation valves.

This report is being made in accordance with 10CFR50.73 (a)(2)(i)(A) "The completion of any nuclear plant shutdown required by the plant's Technical Specification."

## CAUSE OF OCCURRENCE

Investigation determined that the leakage was approximately 15 gpm due to lifting of the 2SJ10 relief valve. Upon closure of the BIT inlet valve 2SJ4 to isolate the flow through the relief valve, a socket weld through wall crack developed on the 2SJ10 relief valve inlet piping.

The socket weld cracking was determined by laboratory analysis to be caused by a combination of fatigue and transgranular stress corrosion cracking (TGSCC). The fatigue cracking most likely initiated at the root of the weld during an event in 1990 where the 2SJ10 lifted and chattered. During this 1990 event the adjacent socket weld failed. After 1990, the weld crack continued to propagate due to TGSCC caused by stagnant flow conditions and high levels of oxygen. On July 14, 2011, during the fill and vent surveillance, the 2SJ10 relief valve lifted. The relief valve lifting was identified as the final fatigue high frequency vibration loading that opened the crack resulting in through wall leakage.

The 2SJ10 relief valve lifting was the result of missed opportunities to correct a component design application deficiency. Insufficient margin exists between the relief valve setpoint and the system operating pressure. During the 2R18 refueling outage, the 2SJ10 was removed for IST testing and failed to lift within 3% of its set-pressure. No failure mechanism was determined prior to reinstalling the valve into the system. A separate apparent cause analysis has been initiated. The relief valve was reinstalled into the system following adjustment and satisfactory as-left testing.

| 1. FACILITY NAME | 2. DOCKET | 6. LER NUMBER | | | 3. PAGE |
|---|---|---|---|---|---|
| | | YEAR | SEQUENTIAL NUMBER | REVISION NUMBER | |
| Salem Generating Station Unit 2 | 05000311 | 2011 | - 0 0 5 - | 01 | 4 of 5 |

NARRATIVE

## PREVIOUS OCCURRENCES

A review of LERs at Salem Station identified two similar events, LER 311/1990-005 "Technical Specification 3.0.3 Entry; Two ECCS Subsystems Inoperable," and LER 272/2005-002 "Technical Specification 3.0.3 Required Plant Shutdown – Sample Line Leak," dated June 20, 2005. The apparent causes were (1) A defect in the root of the weld of a pipe to cap socket weld joint for LER 311/1990-005, and (2) A crack in the pipe weld of the drain valve due to stress corrosion aggravated by dissolved oxygen for LER 272/2005-002.

## SAFETY CONSEQUENCES AND IMPLICATIONS

There were no actual consequences to this event. The leakage through the 2SJ10 BIT relief valve was isolated with the closure of BIT inlet valve 2SJ4 after approximately 4 minutes. Upon closure of the BIT inlet valve to isolate the flow through the relief valve, a through wall socket weld crack developed on the 2SJ10 relief valve inlet piping, which remained isolated. The magnitude of the leak was estimated to be approximately 15 gpm with the majority of the leakage passing through the seat of the 2SJ10 relief valve. The BIT is part of the flow path of the high head safety injection system; therefore, with the identified leakage in the high head safety injection flow path both trains of the ECCS were declared inoperable. Technical Specification (TS) 3.0.3 was entered and an orderly shutdown of the unit was completed.

An assessment of the high head injection system pumps was performed to determine the potential impact of an additional 15 gpm of leakage from the system. This assessment determined that the high head charging pumps would be able to deliver the required high head injection flow to the reactor core in the event of a design basis accident with an additional 15 gpm of flow.

A structural analysis of the condition of the weld that existed prior to the July 14, 2011, fill and vent surveillance was performed. This analysis determined that the defect in the weld would not have failed at the maximum pressure of the high head safety injection system. Prior to the July 14, 2011 surveillance, if a safety injection actuation signal had occurred, the high head injection pumps would start and the BIT inlet and outlet valves would open. Opening of the BIT inlet and outlet valves would pressurize the cracked weld to a pressure equal to or less than the maximum high head injection system pressure.

A review of this event determined that a Safety System Functional Failure (SSFF) as defined in NEI 99-02, Regulatory Assessment Performance Indicator Guidelines, did not occur. This event did not result in a condition that would have prevented the fulfillment of a safety function of a system needed to shutdown the reactor and maintain it in a safe shutdown condition, remove residual heat, control the release of radioactive material, or mitigate the consequences of an accident.

# LICENSEE EVENT REPORT (LER)

| 1. FACILITY NAME | 2. DOCKET | 6. LER NUMBER | | | 3. PAGE |
|---|---|---|---|---|---|
| | | YEAR | SEQUENTIAL NUMBER | REVISION NUMBER | |
| Salem Generating Station Unit 2 | 05000311 | 2011 | - 0 0 5 - | 01 | 5 of 5 |

NARRATIVE

## CORRECTIVE ACTIONS

1. The affected weld was removed and replaced with a new straight run of pipe to minimize welded fittings exposed to the stagnant aerated borated water conditions.
2. The 2SJ10 relief valve was removed from the system and the valve internals were replaced. The relief valve was tested satisfactory and placed back in service.
3. An extent of condition visual examination of small bore and large bore welds in the safety injection piping (from the high head safety injection/charging pumps to the BIT outlet valves in the main header) was completed in Unit 2 with no indication of cracks found.
4. A visual examination of identical welds in Unit 1 was performed, and the inspected welds were found to have no cracks.
5. A design change will be performed to resolve the 1/2SJ10 operating margin.
6. Design changes will be performed to eliminate unnecessary non-isolatable socket welds in the BIT system and replace the remaining non-isolatable socket welds in the BIT system with low carbon weld filler.
7. A design change will be performed to install a stiffener on the Unit 2 2SJ10 piping. A stiffener currently exists on the Unit 1 1SJ10 piping.
8. An evaluation to review relief valve programmatic deficiencies is in progress.

## COMMITMENTS

No commitments are made in this LER.