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JUL 11 2008
10CFR50.73

LR-N08- 0143

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

LER 311/08-002
Salem Nuclear Generating Station Unit 2
Facility Operating License No. DPR-75
NRC Docket No. 50-311

SUBJECT: Technical Specification 3.0.3 Shutdown Due to All Steam Flow Channels Being Inoperable

This Licensee Event Report, "Technical Specification 3.0.3 Shutdown Due to All Steam Flow Channels Being Inoperable" is being submitted pursuant to the requirements of the Code of Federal Regulations 10CFR50.73 (a)(2)(i)(A), "the completion of any nuclear plant shutdown required by the plant's Technical Specification," and 10CFR50.73 (a)(2)(i)(B), "any operation prohibited by the plant's Technical Specification."

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

Sincerely,

Robert Braun
Site Vice President - Salem

Attachments (1)

IE22
NRR

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

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 08/31/2010

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 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)

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4. TITLE Technical Specification 3.0.3 Shutdown Due to All Steam Flow Channels Being Inoperable

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
05	12	2008	2008	0 0 3	0	07	11	2008		DOCKET NUMBER

9. OPERATING MODE

1

10. POWER LEVEL

84%

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 Thomas	(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
N/A									

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 May 12, 2008, Salem Unit 2 was at approximately 84% power when power ascension was placed on hold. During refueling outage 2R16 the steam generators were replaced. Based on review of plant indications, a determination was made that all eight main steam flow reactor protection channels were reading lower than expected. The steam flow channels were reading up to 14% lower than the feedwater flow channels. As a result of these lower than expected indications, the steam flow channel input into the safety injection actuation and main steam line isolation logic was determined to be non-conservative and Technical Specification (TS) 3.0.3 was entered at 1500 hours. A load reduction was commenced and Salem Unit 2 entered Mode 3 (Hot Standby) at 2140 hours. At 2248 the high steam line flow safety injection signal was blocked in accordance with procedures when reactor coolant system temperature was lowered below 543°F and TS 3.0.3 was exited.

The causes of this event were incomplete administrative controls for the use of the Computerized Scaling Manual (CSM) for steam flow instrumentation calibration and a post modification test plan that did not identify and correct the incorrect steam flow calibration. The steam flow channels were recalibrated and additional corrective actions include procedure changes for the CSM and post modification testing.

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," and 10CFR50.73 (a)(2)(i)(B), "any operation ...prohibited by the plant's Technical Specification."

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NARRATIVE

PLANT AND SYSTEM IDENTIFICATION

Westinghouse – Pressurized Water Reactor (PWR/4)

Engineered Safety Function Actuation System {JE/PDT}

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

IDENTIFICATION OF OCCURRENCE

Event Date: May 12, 2008

Discovery Date: May 12, 2008

CONDITIONS PRIOR TO OCCURRENCE

Salem Unit 2 was in Operational Mode 1 at 84% reactor power.

No structures, systems or components were inoperable at the time that contributed to the event.

DESCRIPTION OF OCCURRENCE

On May 12, 2008, Salem Unit 2 was at approximately 84% power when power ascension was placed on hold. During refueling outage 2R16 the steam generators were replaced. Based on review of plant indications, a determination was made that all eight main steam flow reactor protection channels were reading lower than expected. The steam flow channels were reading up to 14% lower than the feedwater flow channels. As a result of these lower than expected indications, the steam flow channel input into the safety injection actuation and main steam line isolation logic was determined to be non-conservative and Technical Specification (TS) 3.0.3 was entered at 1500 hours. A load reduction was commenced and Salem Unit 2 entered Mode 3 (Hot Standby) at 2140 hours. At 2248 the high steam line flow safety injection signal was blocked in accordance with procedures when reactor coolant system temperature was lowered below 543°F and TS 3.0.3 was exited.

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," and 10CFR50.73 (a)(2)(i)(B), "any operation ...prohibited by the plant's Technical Specification."

CAUSE OF OCCURRENCE

The causes of this event were incomplete administrative controls for the use of the Computerized Scaling Manual (CSM) for steam flow instrumentation calibration and a post modification test plan that did not identify and correct the incorrect steam flow calibration.

The CSM was originally designed to calculate steam flow loop gains to normalize steam flow indications using empirical data with the plant operating at power. During the steam generator replacement design change, a combination of previous operating data and predicted operating data were input into the CSM. Due to unconsidered effects such as improved steam quality, the previous operating data did not accurately represent post-outage conditions. The CSM uses the empirical operating data to calculate the steam flow venturi constant. As these additional effects were not recognized, Engineering assumed that the venturi flow constant would not change substantially and utilized the gains from the CSM for initial steam flow calibrations. Following the event it was determined that the venturi flow constant had changed by approximately 10%, which resulted in the steam flow indication error.

Additionally, the modification acceptance test procedure was not effectively implemented so as to detect the steam flow deviation prior to exceeding the Technical Specification allowable value.

PREVIOUS OCCURRENCES

A review of LERs for Salem Units 1 and 2 for the previous three years did not identify any previous similar events.

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NARRATIVE

SAFETY CONSEQUENCES AND IMPLICATIONS

During the start up of the plant following the 2R16 refueling outage, steam flow indications were reading up to 14% lower than expected based on comparison to the actual feedwater flow and plant calorimetric data. Based upon these indications, the eight steam flow protection channels (two per loop) were declared inoperable. The steam flow protection channels in conjunction with a low steam line pressure are used in the safety analysis in response to a main steam line break. An evaluation of the hot zero power (HZP) steam line break (SLB) analysis was performed with an increased steam flow setpoint of 20%. As documented in the analysis of record, the high steam flow setpoint is reached very quickly with the protection delayed until the low steam pressure portion of the coincidence logic is satisfied. The sensitivity runs performed with a 20% increased steam flow setpoint had no impact on the SLB transient. In addition to the transient analysis, a review of the SLB mass energy release inside containment and outside containment was performed. With the steam line flow deviation experienced during startup following 2R16 there was no effect to the containment integrity analysis for a steam line break inside containment and there was no effect to the compartment temperature response for steam line breaks outside containment. Based on the above, although the steam line flow measurements were beyond the required TS setpoint values, 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 NEI 99-02, Regulatory Assessment Performance Indicator Guidelines, did not occur. This event did not prevent the ability of a system to fulfill its safety function to either shutdown the reactor, remove residual heat, control the release of radioactive material, or mitigate the consequences of an accident.

CORRECTIVE ACTIONS

1. The CSM user guide will be revised to clearly state the purpose and limitation for use of the CSM.

2. The post modification acceptance test procedure will be revised to provide additional guidance for integrated startup test plans including key parameter monitoring, clarification of roles and responsibilities, and the conduct of a test plan challenge board.

COMMITMENTS

No commitments are made in this LER.