

Cheryl A. Gayheart
Vice President - Farley

**Southern Nuclear
Operating Company, Inc.**
Farley Nuclear Plant
Post Office Drawer 470
Ashford, Alabama 36312

Tel 334.814.4511
Fax 334.814.4575



June 2, 2014

Docket No.: 50-348
50-364

NL-14-0838

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001


Joseph M. Farley Nuclear Plant – Units 1 and 2
Licensee Event Report 2014-003-00
Scaling Errors Result in Inoperable Steam Flow Channels for
Durations Longer Than Allowed by Technical Specifications

Ladies and Gentlemen:

This Licensee Event Report, "Scaling Errors Result in Inoperable Steam Flow Channels for Durations Longer Than Allowed by Technical Specifications," is being submitted pursuant to the requirements of the Code of Federal Regulations, 10 CFR 50.73(a)(2)(i)(B) as an operation or condition prohibited by Technical Specifications.

This letter contains no NRC commitments. If you have any questions regarding the submittal, please contact Mr. Bill Arens at (334) 814-4765.

Sincerely,


Ms. C. A. Gayheart
Vice President – Farley
CAG/WNA

Enclosure: Units 1 and 2 Licensee Event Report 2014-003-00

U. S. Nuclear Regulatory Commission

NL-14-0838

Page 2

cc: Southern Nuclear Operating Company
Mr. S. E. Kuczynski, Chairman, President & CEO
Mr. D. G. Bost, Executive Vice President & Chief Nuclear Officer
Mr. B. L. Ivey, Vice President – Regulatory Affairs
Mr. C. R. Pierce, Regulatory Affairs Director
Mr. D. R. Madison, Vice President – Fleet Operations
Mr. R. R. Martin, Regulatory Affairs Manager – Farley
Mr. J. E. Purcell, Nuclear Technical Specialist – Farley
Ms. K. A. Walker, Senior Engineer - Farley
RTYPE: CFA04.054

U. S. Nuclear Regulatory Commission
Mr. V. M. McCree, Regional Administrator
Mr. S. A. Williams, NRR Project Manager – Farley
Mr. P. K. Niebaum, Senior Resident Inspector - Farley
Mr. J. R. Sowa, Resident Inspector - Farley
Mr. R. E. Martin, Senior Project Manager- Farley

Joseph M. Farley Nuclear Plant – Units 1 and 2

NL-14-0838

**Scaling Errors Result in Inoperable Steam Flow Channels for
Durations Longer Than Allowed by Technical Specifications**

Enclosure

Units 1 and 2 Licensee Event Report 2014-003-00



LICENSEE EVENT REPORT (LER)
(See Page 2 for required number of digits/characters for each block)

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 FOIA, Privacy and Information Collections Branch (T-5 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollections.Resource@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.

1. FACILITY NAME Joseph M. Farley Nuclear Plant, Unit 1	2. DOCKET NUMBER 05000 348	3. PAGE Page 1 of 4
---	--------------------------------------	-------------------------------

4. TITLE
Scaling Errors Result in Inoperable Steam Flow Channels for Durations Longer Than Allowed by Technical Specifications

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
04	02	2014	2014	003	00	06	2	2014	Farley Nuclear Plant – Unit 2	05000 364
									FACILITY NAME	DOCKET NUMBER

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

1	<input type="checkbox"/> 20.2201(b)	<input type="checkbox"/> 20.2203(a)(3)(i)	<input type="checkbox"/> 50.73(a)(2)(i)(C)	<input type="checkbox"/> 50.73(a)(2)(vii)
	<input type="checkbox"/> 20.2201(d)	<input type="checkbox"/> 20.2203(a)(3)(ii)	<input type="checkbox"/> 50.73(a)(2)(ii)(A)	<input type="checkbox"/> 50.73(a)(2)(viii)(A)
	<input type="checkbox"/> 20.2203(a)(1)	<input type="checkbox"/> 20.2203(a)(4)	<input type="checkbox"/> 50.73(a)(2)(ii)(B)	<input type="checkbox"/> 50.73(a)(2)(viii)(B)
	<input type="checkbox"/> 20.2203(a)(2)(i)	<input type="checkbox"/> 50.36(c)(1)(i)(A)	<input type="checkbox"/> 50.73(a)(2)(iii)	<input type="checkbox"/> 50.73(a)(2)(ix)(A)
100	<input type="checkbox"/> 20.2203(a)(2)(ii)	<input type="checkbox"/> 50.36(c)(1)(ii)(A)	<input type="checkbox"/> 50.73(a)(2)(iv)(A)	<input type="checkbox"/> 50.73(a)(2)(x)
	<input type="checkbox"/> 20.2203(a)(2)(iii)	<input type="checkbox"/> 50.36(c)(2)	<input type="checkbox"/> 50.73(a)(2)(v)(A)	<input type="checkbox"/> 73.71(a)(4)
	<input type="checkbox"/> 20.2203(a)(2)(iv)	<input type="checkbox"/> 50.46(a)(3)(ii)	<input type="checkbox"/> 50.73(a)(2)(v)(B)	<input type="checkbox"/> 73.71(a)(5)
	<input type="checkbox"/> 20.2203(a)(2)(v)	<input type="checkbox"/> 50.73(a)(2)(i)(A)	<input type="checkbox"/> 50.73(a)(2)(v)(C)	<input type="checkbox"/> OTHER
	<input type="checkbox"/> 20.2203(a)(2)(vi)	<input checked="" type="checkbox"/> 50.73(a)(2)(i)(B)	<input type="checkbox"/> 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

LICENSEE CONTACT William N. Arens – Licensing Supervisor	TELEPHONE NUMBER (Include Area Code) (334) 814-4765
---	--

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
X	JE	FT	F180	Y	X	JE	FT	B080	Y

14. SUPPLEMENTAL REPORT EXPECTED <input type="checkbox"/> YES (If yes, complete 15. EXPECTED SUBMISSION DATE) <input checked="" type="checkbox"/> NO	15. EXPECTED SUBMISSION DATE MONTH: DAY: YEAR:
--	--

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

At 1819 CDT on April 2, 2014, with Unit 1 operating in Mode 1 at 100 percent power, a review of scaling data identified that Unit 1 steam generator steam flow transmitter calibration procedures contained incorrect scaling data. Further analysis determined that the 1B Steam Flow Channel Q1C22FT0484 was outside of its allowable Technical Specification tolerance since being calibrated with the incorrect scaling data during the Fall 2013 refueling outage. This steam flow channel was subsequently re-scaled and returned to operable status on April 5, 2014 at 0821.

At 1300 CDT on April 8, 2014, with Unit 2 operating in Mode 1 at 100 percent power, the continuing review of scaling data also identified that Unit 2 steam generator steam flow transmitter calibration procedures contained incorrect scaling data. As a result, the 2C Steam Flow Channel Q2C22FT0494 was outside of its allowable Technical Specification tolerance since being calibrated during the Spring 2013 refueling outage. This steam flow channel was subsequently re-scaled and returned to operable status on April 9, 2014 at 0040.

Since both of these channels had been incorrectly scaled since the beginning of each unit's operating cycle (October, 2013 for Unit 1 and May 2013 for Unit 2) this represents a condition prohibited by Technical Specifications and is reportable pursuant to 10 CFR 50.73(a)(2)(i)(B).



**LICENSEE EVENT REPORT (LER)
CONTINUATION SHEET**

Estimated burden per response to comply with this mandatory collection request: 60 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the FOIA, Privacy and Information Collections Branch (T-5 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to Infocollect.Resource@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.

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REV NO.	
Joseph M. Farley Nuclear Plant, Unit 1	05000 348	2014	- 003	- 00	2 of 4

NARRATIVE

Westinghouse - Pressurized Water Reactor
Energy Industry Identification Codes are identified in the text as [XX].

Requirement for Report

This report is required per 10 CFR 50.73(a)(2)(i)(B) as a condition prohibited by Technical Specifications due to the Unit 1 Steam Flow Channel Q1C22FT0484 [FT] and the Unit 2 Steam Flow Channel Q2C22FT0494 [FT] being inoperable without the applicable Required Actions of Technical Specification 3.3.2 (Engineered Safety Feature Actuation System (ESFAS) Instrumentation) being performed within the required Completion Time.

Unit Status at Time of Event

From the time of the first discovery on April 2, 2014 of an incorrectly scaled steam flow channel until both channels were properly scaled, both units operated in Mode 1 at 100 percent thermal power.

Description of Event

At 1819 on April 2, 2014, during an extent of condition review of an unrelated scaling issue, Engineering personnel identified that Unit 1 procedures for calibration of steam flow channels contained incorrect scaling data due to the transmitter span information not being updated from the previous operating cycle. Further review identified that as a result of the incorrect scaling data, steam flow channel Q1C22FT0484, the 1B Steam Generator Channel III Steam Flow instrument, was outside of its allowed Technical Specification tolerance. The steam flow channel was declared inoperable and the applicable Technical Specification Required Action was entered at 2104 on April 2, 2014. The steam flow instrument was subsequently re-calibrated using a corrected steam flow calibration procedure and restored to operable status at 0821 on April 5, 2014. Since this steam flow instrument had been incorrectly calibrated during the most recent refueling outage in the Fall of 2013, the 72-hour Required Action of Technical Specification 3.3.2 (ESFAS Instrumentation) to place the channel in the tripped condition that became applicable on October 25, 2013 was not met due to station personnel being unaware of the scaling error. The remainder of the Unit 1 steam flow channels, although impacted by the scaling error, remained within the allowable Technical Specification tolerance.

On April 8, 2014, during continued extent of condition reviews, Engineering personnel identified that Unit 2 procedures for calibration of steam flow channels also contained transmitter span data that was not updated from the previous cycle. Further review of Unit 2 steam generator scaling data identified that steam flow channel Q2C22FT0494, the 2C Steam Generator Channel III Steam Flow instrument, was outside of its allowed Technical Specification tolerance. The steam flow channel was declared inoperable and the applicable Technical Specification Required Action was entered at 1145 on April 8, 2014. The steam flow instrument was subsequently re-calibrated using a corrected steam flow calibration procedure and restored to operable status at 0040 on April 9, 2014. Since this steam flow instrument had been incorrectly calibrated during the most recent refueling outage in the Spring of 2013, the 72-hour Required Action of Technical Specification 3.3.2 (ESFAS Instrumentation) to place the channel in the tripped condition that became applicable on May 8, 2013 was not met due to station personnel being unaware of the scaling error. The remainder of the Unit 2 steam flow channels, although impacted by the scaling error, remained within the allowable Technical Specification tolerance.

**LICENSEE EVENT REPORT (LER)
CONTINUATION SHEET**

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
Joseph M. Farley Nuclear Plant, Unit 1	05000 348	YEAR	SEQUENTIAL NUMBER	REV NO.	3 of 4
		2014	- 003	- 00	

NARRATIVE

Cause of Event

The direct cause of this event is due to steam flow instrument transmitter span data tables not being updated with current cycle information prior to instrument calibration during the most recent refueling outage for each unit. These instrument tables had not been updated due to a combination of an omission in procedure FNP-0-ETP-4509 (Scaling Engineer Guidelines) and knowledge weaknesses among the Engineering personnel performing the scaling activities.

Safety Assessment

Steam flow channels perform a safety function by providing a high steam flow input to main steam line isolation logic circuitry. Each of the three steam generators is equipped with two redundant steam flow transmitters. A high steam flow signal from one of the two steam flow transmitters on two of the three steam generators coincident with a low-low reactor coolant system (RCS) average temperature signal from two of three RCS temperature channels generates a main steam line isolation signal that causes closure of all main steam line isolation valves.

During the time periods of the Unit 1 FT-484 and Unit 2 FT-494 out-of-tolerance condition, the redundant steam flow transmitters remained capable of performing their safety function. Therefore, sufficient inputs to the main steam line isolation circuitry were available to actuate a main steam line isolation at the proper setpoint. At no time was there a loss of safety function. Additionally, Unit 1 FT-484 and Unit 2 FT-494 remained capable of actuating during a high steam flow condition, although slightly above the allowed setpoint.

A diverse means of providing a main steam line isolation in the event of a steam line break is the low-steam-pressure main steam line isolation signal. This function remained fully capable of performing the main steam line isolation function during the periods that these two steam flow channels were known to be out of tolerance.

A diverse means of providing a main steam line isolation in the event of a steam line break inside Containment is the high-high Containment pressure main steam line isolation signal. This function remained fully capable of performing the main steam line isolation function during the periods that these two steam flow channels were known to be out of tolerance.

Steam flow channels also provide a control function associated with the steam generator water level control system. The steam flow channel scaling errors caused minimal impact to the steam generator water level control systems with no observable effects on the function of these systems.

Based on the above considerations this condition is considered to have low safety significance.

Corrective Action

Unit 1 steam flow transmitter FT-484 was rescaled per work order SNC564824 and returned to operable status at 0821 on April 5, 2014. Unit 2 steam flow transmitter FT-494 was rescaled per work order SNC566091 and was returned to operable status at 0040 on April 9, 2014. The remainder of the instruments impacted by this scaling error issue have been rescaled per separate work orders.

Calibration procedures for all protection instrument channels on both units controlled by the procedure FNP-0-ETP-4509 (Scaling Engineer Guidelines) have been reviewed with no other instances of scaling errors being identified that impacted channel operability.

**LICENSEE EVENT REPORT (LER)
CONTINUATION SHEET**

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
Joseph M. Farley Nuclear Plant, Unit 1	05000 348	YEAR	SEQUENTIAL NUMBER	REV NO.	4 of 4
		2014	- 003	- 00	

NARRATIVE

The error in procedure FNP-0-ETP-4509 (Scaling Engineer Guidelines) has been corrected. Additionally, an improved and less cumbersome process for incorporating scaling data into calibration procedures is being pursued.

Training of Engineering personnel regarding this event has been conducted. An engineer training plan for scaling and normalization processes will be developed.

Additional Information

Joseph M. Farley Nuclear Plant Unit 2 LER 2013-001-00 was submitted on July 26, 2013 to report two instances of a condition prohibited by Technical Specifications due to the untimely renormalization of Steam Flow Transmitter FT-494 at the beginning of two operating cycles.

Joseph M. Farley Nuclear Plant Unit 1 LER 2013-003-00 was submitted on January 3, 2014 to report a condition prohibited by Technical Specifications due to the untimely renormalization of Steam Flow Transmitter FT-495 at the beginning of an operating cycle.