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May 23, 1990

Alabama Power
the southern electric system

10CFR50.73

Docket No. 50-364

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

Gentlemen:

Joseph M. Farley Nuclear Plant - Unit 2
Licensee Event Report No. LER 89-009-01

Joseph M. Farley Nuclear Plant, Unit 2 Licensee Event Report No. LER 89-009-01 is being submitted in accordance with 10CFR50.73. This is a revision to LER 89-009-00 submitted on October 10, 1989.

If you have any questions, please advise.

Respectfully submitted,

W. G. Hairston, III

WGH,III/JAR:mgd 16.16

Enclosure

cc: Mr. S. D. Ebneter
Mr. G. F. Maxwell

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NRC FORM 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Joseph M. Farley - Unit 2	0 5 0 0 0 3 6 4	1 OF 0 7

TITLE (4) Inaccurate Feedwater Flow Indication Could Have Prevented Proper Operation of the Power Range Nuclear Instrumentation and the Over-Temperature-Delta-Temperature Loops

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
0 5	2 5	8 9	8 9	0 0 9	0 1	0 5	2 3	9 0		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 5 5

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
20.405(a)(1)(ii)	X 50.36(c)(2)	X 50.73(a)(2)(vii)	OTHER (Specify in Abstract below and in Text, NRC Form 366A)
20.405(a)(1)(iii)	X 50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
20.405(a)(1)(iv)	50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	
20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER	
	AREA CODE	
D. N. Morey, General Manager-Nuclear Plant	2 0 5	8 9 9 - 5 1 5 6

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
B	S J	V	K 0 8 5	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

YES (If yes, complete EXPECTED SUBMISSION DATE) | X NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

On 5-25-89 with thermal power at approximately 55%, it was determined that the power range nuclear instrumentation channels (NIs) were reading approximately seven percent low. This occurred because equalizing valves were leaking on two of the three feedwater flow instruments. This feedwater flow is used in a heat balance calculation for determining reactor power.

The equalizing valves were repaired and the NI percent power indications were corrected. Since the incore-excore axial flux difference (AFD) calibration had been completed prior to discovery of the feedwater flow error, an AFD calibration correction was performed on 5-26-89 using flux map data and the corrected power of 55%. The over-temperature delta temperature (OTDT) reactor trip setpoint is reduced for high positive or high negative AFDs.

Alabama Power and the NSSS vendor incorrectly determined that the power range NI error was offset by a constant 7%. However, on 3-2-90, an NRC inspector concluded that the NI error was proportional to power and, in addition, he questioned the AFD calibration correction performed on 5-26-89. Alabama Power performed an investigation which determined that the NI error was proportional to power and that the 5-26-89 AFD calibration adjustment was inadequate. On 3-13-90, the AFD calibration was corrected.

On 4-25-90, Westinghouse issued an evaluation that (1) confirmed that the NI error was proportional to power, (2) verified that the calibration methodology utilized on 3-13-90 was correct, and (3) determined that the AFD calibration error was of no safety significance.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Farley Nuclear Plant - Unit 2	0 5 0 0 0 3 6 4	8 9	— 0 0 9	— 0 1	0 2 OF 0 7

TEXT (If more space is required, use additional NRC Form 366A's) (17)

Plant and System Identification

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

Summary of Event

On 5-25-89 with thermal power at approximately 55%, it was determined that the power range nuclear instrumentation channels (NIs) were reading approximately seven percent low. This occurred because equalizing valves were leaking on two of the three feedwater flow instruments. This feedwater flow is used in a heat balance calculation for determining reactor power.

The equalizing valves were repaired and the NI percent power indications were corrected. Since the incore-excore axial flux difference (AFD) calibration had been completed prior to discovery of the feedwater flow error, an AFD calibration correction was performed on 5-26-89 using flux map data and the corrected power of 55%. The over-temperature delta temperature (OTDT) reactor trip setpoint is reduced for high positive or high negative AFDs.

Alabama Power and the NSSS vendor incorrectly determined that the power range NI error was a constant 7% offset. However, on 3-2-90, an NRC inspector concluded that the NI error was proportional to power and, in addition, he questioned the AFD calibration correction performed on 5-26-89. Alabama Power performed an investigation which determined that the NI error was proportional to power and the the 5-26-89 AFD calibration adjustment was inadequate. On 3-13-90, the AFD calibration was corrected.

On 4-25-90, Westinghouse issued an evaluation that (1) confirmed that the NI error was proportional to power, (2) verified that the calibration methodology utilized on 3-13-90 was correct, and (3) determined that the AFD calibration error was of no safety significance.

Description of Event

At 0443 on 5-21-89, power generation began following the sixth refueling outage. At 1040 on 5-21-89, FNP-2-STP-109 (Power Range Neutron Flux Channel Calibration) was performed. This procedure is a calorimetric heat balance which uses feedwater flow to determine thermal reactor power for calibrating the power range NI percent power indication. The thermal power was determined to be approximately 25% of rated thermal power. The NIs, which indicated 31%, were adjusted to agree with this calculation as required by FNP-2-STP-109 and Technical Specifications.

Later, on 5-21-89, reactor power was reduced to approximately 15% in order to perform the scheduled main turbine overspeed test. Following completion of the overspeed test, power was increased. At 0500 on 5-22-89, another FNP-2-STP-109

was performed. It was determined that reactor power was approximately 35% of rated thermal power. The NIs were adjusted to agree with this calculation as required. The Shift Supervisor noted that other indications of thermal power did not appear to match the NIs. The Shift Supervisor discussed the discrepancy with his relief and with the on-call Emergency Director. One consideration was that a problem might exist with the feedwater flow indicators. The feedwater flow indicators were filled and vented. No problem was identified. The investigation continued; however, an automatic reactor shutdown occurred later that day before an additional heat balance could be performed.

On 5-23-89, during the power ascension following the reactor shutdown, it was noted by the on-call Emergency Director and the on-call Operations Manager that turbine load was substantially less than it had been for corresponding power levels prior to the reactor shutdown. Due to this, management believed the nuclear power indication problem had been resolved.

No NI adjustments were required on 5-23-89 after the plant startup.

On 5-23-89, with indicated reactor power at approximately 31%, the incore-excore cross calibration was performed per FNP-2-STP-121. In this test, calorimetric data, including feedwater flow, is taken to determine power levels. These power levels are used in the calculation of calibration constants for the excore AFD channels. At the time, it was not realized that the calorimetric calculation was low, which resulted in the use of incorrect powers for the calculation of calibration constants for the AFD channels. The incorrect calibration constants were input on 5-24-89.

On 5-24-89 at 2205, with indicated reactor power at 48%, NI high flux trip setpoints were adjusted from 80% to the normal value of 109%. The trip setpoint is conservatively set at 80% power following a refueling outage until the initial FNP-2-STP-109 calorimetric measurement is performed.

On 5-24-89 at the end of the evening shift, the Shift Supervisor notified management that additional investigation had indicated that a discrepancy still existed between the NIs and other indications of reactor power. Previous low power FNP-2-STP-109 (calorimetric) results were then reviewed in order to positively identify what values were different for this power level. This showed a difference in feedwater flow indications. An additional FNP-2-STP-109 was performed using an alternate feedwater flow indication which revealed that two of the three feedwater indicators used for calorimetric readings were indicating low. The feedwater flow indicators were then re-calibrated. This occurred during the day shift on 5-25-89. During the calibration, the equalizing valves on two of the three indicators were found to be leaking, causing these indicators to give an erroneous flow indication. The equalizing valves were repaired and the NIs were recalibrated per FNP-2-STP-109, changing the indicated power from 48% to 55%. Subsequent performances of FNP-2-STP-109 indicated agreement between the calorimetric results and other indications of thermal power. The agreement between NI power and other indications was monitored closely during the subsequent power increase to 100%.

When the power level error was discovered at 55% power, Reactor Engineering incorrectly determined that an existing procedure would correct the AFD calibration. Using this procedure the revised AFD calibration constants were entered on 5-26-89.

The plant staff and the NSSS vendor originally determined (incorrectly) that the difference between the actual reactor power and the NI-indicated power was a constant 7% offset.

On 3-2-90, an NRC inspector questioned Alabama Power's dispositioning of the calorimetric error. The NRC inspector concluded that the NI error was proportional to power level and, therefore, was larger at higher power levels than FNP had originally believed. The inspector also questioned the methodology used to correct the AFD calibration constants.

Following the NRC inspection, Alabama Power determined that (1) the NI error was proportional to power, not a constant offset, and (2) the 5-26-89 AFD calibration adjustment was inadequate. Based on these conclusions, the AFD calibration was corrected and the channels were recalibrated on 3-13-90.

An investigation was conducted by Alabama Power to review several issues associated with the NI calibration error. On 4-25-90, the evaluation was completed. The results confirmed that the power level offset had been proportional to power rather than constant, and that the OTDT trip setpoint AFD penalty had been miscalibrated from the time the excore channels were originally recalibrated on 5-24-89 until the calibration was corrected on 3-13-90. The AFD miscalibration did not affect OTDT because AFDs as large as +9% or -35% never occurred during this period.

Cause of Event

(1) Leaking equalizing valves on two of the three feedwater flow indicators introduced error into the calorimetric calculation, causing the calculated reactor power to be less than actual power.

(2) Agreement between indicators of thermal power was not verified prior to changing the power range NI high flux setpoint from 80% to 109%.

(3) When the power level error was discovered, Reactor Engineering incorrectly determined that an existing procedure would correct the AFD calibration.

(4) Improper engineering evaluation by the plant staff and the NSSS vendor that the calorimetric error was a constant offset rather than proportional to power.

Reportability Analysis and Safety Assessment

This event is reportable since a single cause or condition caused two or more independent channels to become inoperable in a system designed to shut down the reactor. Since personnel did not realize at the time that the channels were inoperable, Technical Specification action statement requirements were not implemented for the inoperable NI and OTDT channels.

This event was not safety significant for the following reasons:

1. The reactor did not exceed 55% of rated thermal power during the period when the power range nuclear instruments were miscalibrated.

2. The effect of the NI miscalibration on the power range neutron flux low setpoint; the power range neutron flux high setpoint; and the P-7, P-8, P-9, and P-10 permissives has been evaluated for plant operation up to 55% thermal power. The evaluation determined that the conclusions of the FSAR remained valid for the miscalibration.

3. An evaluation has determined that adequate OTDT DNB protection was still available during full power operation with the miscalibration of the axial flux difference penalty in the OTDT equation.

4. Since the unit operated at relatively small axial offsets, the axial flux difference calibration error did not affect the OTDT setpoint. The other inputs to the OTDT protection system remained correctly calibrated.

5. The error in calculation of axial flux difference was insignificant at the small AFDs at which the unit operated. This was demonstrated by periodic surveillance testing during the interval in which the miscalibration existed.

6. Other systems were available to initiate a reactor trip, such as overpower delta-temperature, high power range positive rate, pressurizer high pressure and manual trip.

Had this event occurred at a higher power level with all worse case instrumentation channel errors and setpoint errors present, then the reactor could have operated at a thermal power level greater than that assumed for accident analysis.

If escalation to 100% power was attempted with an erroneous power range indication, numerous primary and secondary plant anomalies would have alerted the operators to stop the power increase. The anomalies include:

1. Delta temperature disagreement with power range channels.

2. Low main feed pump suction pressure.

3. High steam and feedwater flow.

4. High generator megawatts.

5. Inability of the turbine to go much higher than 100% design load.

6. OTDT and overpower delta-temperature (OPDT) rod stops.

Further, the OPDT trip would have prevented normal operation above 108% thermal power.

There was no effect on the health and safety of the public.

Corrective Action

1. The equalizing valves on the subject feedwater flow indicators have been repaired.

2. The power range nuclear instruments were adjusted to within two percent of actual thermal power as required by Technical Specifications surveillance.

3. Errors in the calculation of the AFD and OTDT trip setpoints have been corrected.

4. A summary of this event was initially sent to all Shift Supervisors. A revised summary of this event will be sent to all licensed personnel and the Reactor Engineering staff for their review of the lessons learned.

5. The return-to-service checklist which is performed following outages has been revised to ensure that the following are considered:

   (a) The equalizing valves on the feedwater flow indicators used in the FNP-1/2-STP-109 calorimetrics are checked for leakage;

   (b) The power range nuclear instrumentation trip setpoints are not reset from 80% to 109% until disagreements among power level indicators are resolved; and

   (c) Power will not be increased above 50% until discrepancies among power level indicators are resolved.

6. A newer method developed by Westinghouse for performing an incore-excore calibration adjustment has been incorporated into Appendix B of FNP-1/2-STP-121. This method is independent of power and, therefore, would have corrected the calorimetric error if used for the 5-23-89 data.

Corrective Action (continued)

7. A summary of this event and the lessons learned have been discussed with appropriate NSSS vendor personnel.

Additional Information

No similar LERs have been submitted by Farley Nuclear Plant.
The equalizing valve in the three valve manifold was manufactured by Kerotest.
The part number is ALD-10066.