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Southern Nuclear Operating Company
the southern electric system

J. D. Woodard
Vice President
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December 22, 1992

Docket No. 50-348

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

Joseph M. Farley Nuclear Plant - Unit 1
Licensee Event Report No. LER 92-008-00

Gentlemen:

Joseph M. Farley Nuclear Plant, Unit 1, Licensee Event Report No. 92-008-00 is being submitted in accordance with 10 CFR 50.73.

If you have any questions, please advise.

Respectfully submitted,

J. D. Woodard

MGE:cht

Enclosure

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

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NRC Form 366 (6-89) U.S. NUCLEAR REGULATORY COMMISSION

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

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Joseph M. Farley Nuclear Plant - Unit 1	0 5 0 0 0 3 4 8	1 OF 4

TITLE (4)
Reactor Trip on Low Steam Generator Level Coincident With Feedwater Flow Less Than Steam Flow

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQ NUM	REV	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
1 2	1 3	9 2	9 2	0 0 8	0 0	12	22	92		0 5 0 0 0
										0 5 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR (11)

OPERATING MODE (9)	1											
POWER LEVEL	1 0 0		20.402(b)		20.405(c)	X	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)		50.36(c)(2)		50.73(a)(2)(vii)		OTHER (Specify in Abstract below)
			20.405(a)(1)(iii)		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	
R. D. Hill - General Manager Nuclear Plant	205	899-5156

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORT TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORT TO NPRDS	
X	J B	F U	W 1 2 1	Y							

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES(If yes, complete EXPECTED SUBMISSION DATE)	X NO				

ABSTRACT (16)

On 12-13-92 at 0257, the Unit 1 reactor tripped on feedwater flow less than steam flow coincident with low steam generator water level on the C steam generator (SG). At the time of the trip, a monthly surveillance test procedure (STP) was in progress on C SG feedwater flow protection channel III. The prerequisites for this STP require verification of a two position main control board (MCB) handswitch to ensure the redundant channel (protection channel IV) provides an uninterrupted steam flow control signal input to the SG water level control circuitry during testing. In this case, the selector switch was already in the channel IV position and the operators verified the switch position as part of the STP prerequisites. The SG feedwater flow channel was successfully placed in test. The STP proceeded successfully until a channel III steam flow multiplier-divider (NMD) card was removed per procedure. When this card was removed, the steam flow signal to the C SG water level control circuitry was lost. The loss of the steam flow signal was the result of a control circuit relay in the Westinghouse 7300 control system failing to energize when the MCB selector switch for steam flow input was selected to the redundant channel (IV). The C SG steam flow signal provides input to the water level control circuitry for the C SG main feedwater flow control valve and also to the common feed pump speed control circuitry. With the loss of the steam flow input, feed pump speed control for both main feed pumps and the main feedwater flow control valve associated with the C SG became erratic. This resulted in oscillations in feedwater flow and steam generator level which eventually led to the reactor trip. The failure of the relay to energize was attributed to a cracked fuse on the control system relay card. The card was replaced, and the unit returned to power operations at 1152 on 12-14-92.

Plant and System Identification

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

Summary of Event

On 12-13-92 at 0257, the Unit 1 reactor tripped on feedwater flow less than steam flow in coincidence with low SG water level on the C SG. At the time of the trip, FNP-1-STP-215.6B was in progress on C SG feedwater flow protection channel III. The prerequisite of the STP requiring the selection of the redundant C SG steam flow channel (channel IV) using the MCB selector switch had been verified. Due to the failure of a control circuit relay to energize when the redundant steam flow channel IV was selected, the C SG steam flow input to the C SG water level control circuitry was actually provided by channel III until the NMD card was removed during the performance of the STP. This resulted in oscillations in feedwater flow and SG level which led to the reactor trip.

Description of Event

On 12-13-92, FNP-1-STP-215.6B was in progress on Unit 1. This STP provides functional testing of channel III Westinghouse 7300 protection system feedwater flow circuits for the reactor protection system [JB]. In addition, this channel provides isolated steam flow and feedwater flow control signals to the C SG water level control circuitry. Part of the prerequisites for the STP is the verification of an alternate controlling channel for feedwater flow and for steam flow before the STP commences. In this case, the redundant channel, channel IV, had been selected previously and the operators properly verified the handswitches as a part of the STP prerequisites.

The steam flow signal was lost to the SG water level control circuitry when the steam flow NMD card was removed. This action affected main feed pump speed control and the C SG main feedwater flow control valve. The resulting oscillations in SG water level control eventually led to the reactor trip when level in the C SG decreased to the 25 percent setpoint. The 25 percent low level setpoint on one SG in coincidence with feedwater flow less than steam flow on the respective SG initiated the reactor trip. The feedwater flow less than steam flow signal on C SG was present due to the STP that was in progress.

The channel III steam flow input was lost due to a control circuit relay in the 7300 control system that failed to energize. This relay would have had to energize in order to complete the electrical portion of the transfer from channel III to channel IV. The relay failed due to a cracked fuse.

Cause of Event

This event was caused by a cracked fuse. This fuse failure prevented the electrical transfer of the redundant steam flow signal input to the C SG water level control circuitry from being completed during conduct of a protection system feedwater flow STP.

FNP has concluded that the fuse was most likely broken during the relay card's removal and subsequent reinsertion during 7300 system testing during the Fall outage on Unit 1.

No similar cracked fuse failures have occurred at FNP.

Reportability Analysis and Safety Assessment

This event is reportable because of the actuation of the reactor protection system. After the trip, the following safety systems operated as designed:

- main feedwater was isolated by automatic closure of the flow control valves and bypass valves;
- auxiliary feedwater pumps started automatically and provided flow to the steam generators;
- source range nuclear detectors energized automatically; and
- pressurizer heater and spray valves operated automatically as required to maintain reactor coolant system pressure.

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

Corrective Action

The relay card with the cracked fuse was replaced.

All applicable 7300 protection system functional test STPs for Unit 1 and Unit 2 will be be changed prior to use to require I & C personnel to be in phone communication with the control room operator prior to removing cards. This would alert the operator to monitor potentially affected control parameters and take action as necessary such as replacing the card immediately if parameters indicated the loss of a controlling signal.

This event will be discussed with all Instrumentation and Control personnel with emphasis on exercising caution when handling electronic cards. This action will be completed by January 31, 1993.

A sampling fuse inspection will be performed on 7300 relay cards to verify fuse integrity at the next available opportunity. This action will also serve to ensure this failure is singular in nature.

TEXT

Additional Information

The failed relay card is manufactured by Westinghouse, part number 2837A87G04.

This event would not have been more severe if it had occurred under different operating conditions.

No similar LERs have been submitted by Farley Nuclear Plant.