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FACIL:50-335 St. Lucie Plant, Unit 1, Florida Power & Light Co. 05000335
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RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 91-005-00:on 910701,reactor trip from 100% power on low steam generator water level occurred.Caused by de-energized feedwater regulating valve due to deficient procedure. Plant maint procedures revised.W/910729 ltr.

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FPL

July 29, 1991

L-91-208
10 CFR 50.73

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

Gentlemen:

Re: St. Lucie Unit 1
Docket No. 50-335
Reportable Event: 91-05
Date of Event: July 1, 1991
Reactor Trip on Low Steam Generator Water Level

The attached Licensee Event Report is being submitted pursuant to the requirements of 10 CFR 50.73 to provide notification of the subject event.

Very truly yours,

D A Sager

D. A. Sager
Vice President
St. Lucie Plant

DAS/JJB/kw

Attachment

cc: Stewart D. Ebneter, Regional Administrator, USNRC Region II
Senior Resident Inspector, USNRC, St. Lucie Plant

DAS/PSL #481-91

IE22 1/1

FPL Facsimile of NRC Form 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92
ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
St. Lucie Unit 1	0 5 0 0 0 3 3 5	1 OF 0 4

TITLE (4) Reactor Trip from 100% Power on Low Steam Generator Water Level caused by a De-energized Feedwater Regulating Valve due to a Deficient Procedure.

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 7	0 1	9 1	9 1	- 0 0 5 -	0 0	0 7	2 9	9 1	N/A	0 5 0 0 0
									N/A	0 5 0 0 0

OPERATING MODE (9)	1
POWER LEVEL (10)	1 0 0

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)	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 and in Text NRC Form 366A)*
	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)
Tom Kilroy, Shift Technical Advisor	4 0 7 4 6 5 - 3 5 5 0

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
D	J B	L I	S 1 8 5	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

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

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

On July 1, 1991, St. Lucie Unit 1 was operating in Mode 1 at 100% power. Utility Maintenance personnel were attempting to remove LIC-9013-B, the Channel B narrow range level indicator for the 1A Steam Generator. At 1137 FIC-9011, the 1A Feedwater Regulating Valve Digital Controller, lost all power. The 1A Steam Generator water level decreased, and the Reactor Control Operator opened the 100% Main Feedwater Bypass Valve in an attempt to regain level. At 1138 the reactor tripped automatically on low level in the 1A Steam Generator. Standard Post-Trip Actions were performed, and the plant was stabilized in Mode 3, Hot Standby.

The root cause of this event was due to a deficient procedure which did not completely isolate electrical power to LIC-9013-B during maintenance. With this indicator controller unisolated, two leads were inadvertently shorted, resulting in an overload and blown power supply fuse. As transfer to the back up power supply occurred, that power supply's breaker tripped resulting in a complete loss of power to FIC-9011. This loss of power resulted in a close demand signal from the controller to the 1A Feedwater Regulating Valve. The Feedwater Regulating Valve went closed, and water level decreased in the 1A Steam Generator to the reactor trip setpoint.

Corrective actions as a result of this event: a revision of plant maintenance procedures to direct the isolation of power to the steam generator level indicators, and a procedure revision to require supervisory and department head review of proposed work on sensitive systems. Also, a plant modification for replacing the steam generator level indicators with a model which has pin type connectors is under consideration.

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

DESCRIPTION OF THE EVENT

On July 1, 1991, St. Lucie Unit 1 was operating in Mode 1 at 100% power. Utility Maintenance (I&C) personnel were attempting to remove LIC 9013-B, 1A Steam Generator Level Indicator/Controller, Channel B (EIIS:LIK), in order to perform maintenance work on it. Auxiliary Feedwater Actuation System (AFAS) (EIIS:BA) and Reactor Protective System (RPS) (EIIS:JC) Channel B Low Steam Generator Level bistables were bypassed, and the plant sensitive systems procedure was being followed.

At 1137, as utility Maintenance personnel proceeded with their work, the Reactor Control Operator (RCO) noticed amperage drops on the 1A Steam Generator Feedwater Pump (SGFP) (EIIS:SJ), and that the Main Feedwater Regulating Valve Digital Controller, FIC-9011 (EIIS:FIK) was de-energized. The Assistant Nuclear Plant Supervisor (ANPS) immediately instructed the Maintenance personnel to terminate their work. 1A Steam Generator (EIIS:SG) water level dropped rapidly, and the RCO attempted to regain steam generator level by opening the 100% Main Feedwater Bypass Valve. As level in the 1A Steam Generator continued to decrease, the ANPS instructed the RCO to manually trip the reactor, in anticipation of the automatic reactor trip on low steam generator level. However, at 1138, before the reactor could be manually tripped, it tripped automatically on low steam generator level (setpoint: 20.5% narrow range). All control rods inserted fully, and EOP-1, Standard Post-Trip Actions, was performed. All safety functions were met.

During the post trip recovery, water level in the 1A Steam Generator was regained by using the 1A 100% Main Feedwater Bypass Valve. AFAS-2 actuated due to the 1B Steam Generator level reaching the initiation setpoint (19.5% narrow range) post trip, and remaining below the AFAS reset setpoint after the time delay. Several minutes into the event, power was restored to FIC-9011 and LIC-9005, the 1A Main Feedwater 15% Bypass Valve Controller (EIIS:LIK), which had also lost power during the event. Normal steam generator levels were regained, AFAS was reset, two sets of Safety Function Status Checks were performed per EOP-2, Reactor Trip Recovery, and Unit 1 was stabilized in Mode 3, Hot Standby.

CAUSE OF THE EVENT

The root cause of the low water level in the 1A Steam Generator and subsequent reactor trip was due to a procedural deficiency in that all energized terminal leads to the level controller were not identified. As a result of the procedural deficiency, only two of the three control functions from the level indicator were disabled. Bypass keys were properly used for isolating one channel of Steam Generator level input to the RPS and the AFWAS, but not for the non-safety related Steam Generator High Level circuit. The plant work order relied on I&C procedure 1-1400153A, Calibration of Steam Generator Level, to provide instructions for removal of LIC-9013-B from service, as well as Administrative procedure 0010142, Unit Reliability - Manipulation of Sensitive Systems. (This procedure administratively controls maintenance work on equipment which may be trip sensitive.) The steps in the I&C procedure did not provide guidance to fully isolate the controller from the Steam Generator High Level control circuit.

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

Subsequently, an inadvertent shorting of the controller's neutral lead against an energized terminal caused an overload and resulted in a blown fuse on the main power supply. When automatic transfer to the alternate power supply occurred with the short still present, that power supply's output breaker tripped, resulting in a loss of power to the 1A Feedwater Regulating Valve Controller, FIC-9011, and the Main Feedwater 15% Bypass Valve Controller, LIC-9005.

Loss of power to FIC-9011 caused a close demand signal to be sent to the 1A Feedwater Regulating Valve, FCV-9011. The Feedwater Regulating Valve went shut, resulting in the rapid loss of water level in the 1A Steam Generator. The Main Feedwater 100% Bypass Valve was manually opened in an attempt to regain level, but its response was not rapid enough to overcome the steam flow/feed flow mismatch which existed. The reactor tripped automatically on low steam generator level as sensed by 2 out of the 3 available RPS trip bistables.

Other causal factors for this event include the environmental conditions of a confined work area with limited access. The Maintenance worker inadvertently shorted the energized lead because of this confined work area. Also, the job planning was less than adequate in that all energized leads to the indicator were not identified.

ANALYSIS OF THIS EVENT

This event is reportable to the Nuclear Regulatory Commission under 10 CFR 50.73.a.2.iv as "...any event or condition that results in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protective System (RPS)."

The plant response to this event was bounded by the accident analysis of the St. Lucie Unit 1 FUSAR, section 15.2.8, "Loss of Normal Feedwater Flow". The actual plant response was much more conservative than the accident analysis because of the following:

1) Only one Feedwater Regulating Valve closed in this event, instead of the total loss of normal feedwater assumed in the analysis.

2) The reactor tripped automatically on low steam generator level at 20.5% narrow range in the 1A Steam Generator. In the accident analysis, this low steam generator trip function is assumed not to take place, and the reactor is assumed to trip on high pressurizer pressure.

The Auxillary Feedwater Actuation System functioned as required during this event. 1A Steam Generator level was restored following the trip due to the 100% Main Feedwater Bypass Valve being open, thus AFAS-1 never actuated. AFAS-2 initiated because 1B Steam Generator level decreased to 19.5% narrow range post-trip, and stayed below this value until the AFAS time delay had timed out. The 1B and 1C Auxillary Feedwater Pumps started as expected, but were not needed to restore water level to the 1B Steam Generator, as the 1B Steam Generator Feedwater Pump was available throughout the entire event.

The plant response during the reactor trip was observed to be normal for the given conditions. All safety systems functioned as designed and all safety functions were met. At no time during this event was the health and safety of the general public endangered.

CORRECTIVE ACTIONS

Corrective actions brought about as a result of this event include:

1) I&C Maintenance procedures1-1400153A and 2-1400153A, Calibration of Steam Generator Level, used by the Instrument and Controls personnel during removal of LIC-9013-B, were changed to include cautions to utilize the High Level Bypass Keyswitch.

2) I&C Maintenance will have warning labels posted for all Sigma indicators (inside the Reactor-Turbine-Generator Control Boards) that affect control functions to ensure heightened awareness by those personnel who perform maintenance on these instruments.

3) To address job planning generically, Plant Administrative procedure 0010142, Unit Reliability - Manipulation of Sensitive Systems, will be revised to explicitly require the cognizant supervisor and the department supervisor to review and approve of all work on sensitive systems prior to beginning the job.

4). To address the work environment, a plant modification for Unit 1 and Unit 2 to replace the existing Steam Generator Level Indicator Sigmas with plug in type instruments is under evaluation. These instruments can be removed from the front of the control board without the need to lift any leads.

5) The faulted 1A Steam Generator narrow range level indicator, FIC-9011, was repaired and placed back in service.

ADDITIONAL INFORMATION

Affected Component Identification

Sigma Level Indicator / Controller
Model Number : 9262

Previous Similar Events

For the most recent previous similar event, see LER #335-90-007, "Manual Reactor Trip due to Unisolable Digital Electrohydraulic Fluid Leak on the # 3 Governor Valve." The root cause of this event was less than adequate procedural guidance in parts specification for Maintenance personnel working in the field, which resulted an O-ring failure and a leak in the DEH system.