

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR: 9512190145    DOC.DATE: 95/12/14    NOTARIZED: NO    DOCKET #  
 FACIL: 50-335 St. Lucie Plant, Unit 1, Florida Power & Light Co.    05000335  
 AUTH.NAME    AUTHOR AFFILIATION  
 BENKEN, E.J.    Florida Power & Light Co.  
 SAGER, D.A.    Florida Power & Light Co.  
 RECIP.NAME    RECIPIENT AFFILIATION

SUBJECT: LER 95-010-00: on 951116, manual reactor trip occurred due to degraded 24 volt DC power supply to 1B SG main feedwater controller. Tested & replaced power supply for 1B feedwater controller. W/undated ltr.

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L-95-334  
10 CFR 50.73

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

Re: St. Lucie Unit 1  
Docket No. 50-335  
Reportable Event: 95-010  
Date of Event: November 16, 1995  
Manual Reactor Trip Due to Low Steam Generator Water Level  
Caused by Main Feedwater Regulating Valve Closure

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  
Vice President  
St. Lucie Plant

DAS/EJB

Attachment

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

133106

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PDR ADDCK 05000335  
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**LICENSEE EVENT REPORT (LER)**

(See reverse for required number of digits/characters for each block)

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-8 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

FACILITY NAME (1)

ST LUCIE UNIT 1

DOCKET NUMBER (2)

05000335

PAGE (3)

1 OF 6

TITLE (4)

Manual Reactor Trip Due to Low Steam Generator Water Level Caused by Main Feedwater Regulating Valve Closure

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 NAME	DOCKET NUMBER
11	16	95	95	-- 010	-- 00	12	14	95	N/A	N/A
									FACILITY NAME	DOCKET NUMBER
									N/A	N/A
									FACILITY NAME	DOCKET NUMBER
									N/A	N/A

OPERATING MODE (9)	POWER LEVEL (10)	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR 5: (Check one or more) (11)			
1	100	<input type="checkbox"/> 20.2201(b)	<input type="checkbox"/> 20.2203(a)(1)	<input type="checkbox"/> 20.2203(a)(2)(i)	<input type="checkbox"/> 20.2203(a)(2)(ii)
		<input type="checkbox"/> 20.2203(a)(2)(ii)	<input type="checkbox"/> 20.2203(a)(2)(iii)	<input type="checkbox"/> 20.2203(a)(2)(iv)	<input checked="" type="checkbox"/> 50.73(a)(2)(vii)
		<input type="checkbox"/> 20.2203(a)(3)(i)	<input type="checkbox"/> 20.2203(a)(3)(ii)	<input type="checkbox"/> 50.36(c)(1)	<input type="checkbox"/> 50.73(a)(2)(viii)
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		<input type="checkbox"/> 20.2203(a)(2)(iv)	<input type="checkbox"/> 50.36(c)(2)	<input type="checkbox"/> 50.73(a)(2)(iv)	Specify in Abstract below or in NRC Form 366A

LICENSEE CONTACT FOR THIS LER (12)

NAME

Edwin J. Benken, Licensing Engineer

TELEPHONE NUMBER (Include Area Code)

(407) 467 - 7156

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
X	SJ	RJX	AC38	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 15 single-spaced typewritten lines) (16)

On November 16, 1995, at 1741 hours, St. Lucie Unit 1 was manually tripped from 100 percent power due to decreasing water level in the 1B Steam Generator. The level in the 1B Steam Generator was restored immediately following the reactor trip, and Auxiliary Feedwater was used to supply feedwater to both Steam Generators. All safety function status checks were satisfactorily completed and the plant was stabilized in Mode 3 following the trip.

The event was caused by a degraded 24 volt DC power supply to the 1B Steam Generator Main Feedwater Controller. The degraded power supply, coupled with a momentary voltage drop associated with the start of a plant fire pump, resulted in a voltage loss to the feedwater controller. Loss of voltage to the controller caused the partial closure of the Main Feedwater Regulating Valve supplying feed to the 1B Steam Generator. Subsequent testing showed that the power supply had degraded, and was supplying voltages incompatible with proper controller operation.

Corrective actions: 1) The power supply for the 1B feedwater controller was tested and replaced. 2) The 1A and Unit 2 feedwater power supplies were tested and found to be satisfactory. 3) The failure mode of the power supply is being evaluated. 4) A plant modification was developed to provide a vital source of power to the controller which is less susceptible to voltage fluctuations. 5) Plant staff is evaluating the Feedwater and Condensate systems for additional power supplies which could represent single point trip vulnerabilities. 6) Plant staff is evaluating the cause of additional alarms which were received during the momentary voltage drop.

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

FACILITY NAME (1)  ST. LUCIE UNIT 1	DOCKET  05000335	LER NUMBER (6)			PAGE (3)  2 OF 6
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
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TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

**DESCRIPTION OF THE EVENT**

On November 16, 1995, St. Lucie Unit 1 was operating at 100 percent reactor power. At approximately 1735, Operations personnel were performing a periodic surveillance on the 1A and 1B Fire Pumps (EIS:KP), in accordance with an approved plant operating procedure. This surveillance is performed to demonstrate that the two plant Fire Pumps will automatically start and operate on a loss of Fire System pressure.

At 1740, the 1A and 1B Fire Pumps were started in accordance with the surveillance procedure. When the pumps started, a voltage transient on the 480 volt load center caused several control room annunciators (EIS:IB) to alarm. These annunciators included D-33, "Vacuum Breaker Vlv. Overload", B-11, "SU Transformer Lockout Relay Trip," and Q-36, "Radiation Monitoring Failure". The alarms were immediately reset and cleared. As the utility licensed operator was acknowledging the alarms, the 1B Steam Generator low level alarm, G-9 (EIS:IB) annunciated. Control board indications confirmed that the level in the 1B Steam Generator was decreasing. The utility licensed operator attempted to take manual control of the Main Feedwater Regulating Valve (EIS:SJ) to increase Steam Generator level. This action was not successful, and the 1B Steam Generator level continued to decrease. At 1741, with the 1B Steam Generator level at approximately 40 percent, a manual reactor trip was initiated. All Control Element Assemblies (CEAs) (EIS:AA) fully inserted, and the main turbine was tripped as a result of the reactor trip.

Emergency Operating Procedure (EOP)-1, "Standard Post Trip Actions," was immediately implemented by the control room staff. While performing EOP-1, a utility licensed operator observed that the 1B Steam Generator level was increasing at an abnormally high rate. The operating crew reset and closed the 15 percent feedwater bypass valves (EIS:SJ) and initiated a close signal to the isolation valve for the Main Feedwater Regulating Valve (EIS:SJ). Level in the 1B Steam Generator continued to increase as the Main Feedwater Regulating Isolation valve was closing, and the 1A and 1B Main Feedwater Pumps (EIS:SJ) were stopped to prevent overfill of the SG. The 1B Steam generator level increase was terminated at approximately 85 percent narrow range level.

The Automatic Feedwater Actuation System (AFAS) (EIS:BA) for the 1A Steam Generator actuated as designed, to supply feedwater to the 1A Steam Generator. Because of the high level in the 1B Steam Generator following the plant trip, AFAS-2 (EIS:BA) automatically reset as designed prior to actuation. The 1B Auxiliary Feedwater (AFW) Pump was started manually at 1757 and both Steam Generator levels were controlled to within the normal band.

Following the completion of Standard Post Trip Actions, the control room staff entered the Reactor Trip Recovery procedure, 1-EOP-02, with all safety functions being met. The plant was maintained stable in Mode 3 while awaiting the completion of a post trip review.

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

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
ST. LUCIE UNIT 1	05000335	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	3 OF 6
		95	-- 010	-- 00	

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**CAUSE OF THE EVENT**

The reactor was manually tripped by the utility licensed operators to mitigate a loss of feedwater transient by early operator action. The cause of this event was the failure of a 24 volt DC power supply to maintain proper voltage to the 1B Main Feedwater Regulating Valve Level Indicating Controller (LIC 9021) (EIS:JB).

When large motors such as the plant Fire Pumps are started, there is a momentary, transient voltage drop on the 480 volt electrical bus (EIS:EC) as the motor comes to full speed. This voltage drop cascades into the dependent loads such as the panel which powers the feedwater regulating system controller power supply. In the case of the Feedwater Regulating System, the 480 volt bus which supplies the Fire pump (1B2), also supplies power panel-102A (EIS:ED), which provides 120 volt AC power to the "B" Feedwater Control System 24 volt DC power supply (refer to Figure 1). The feedwater controller power supply is designed to handle input voltage fluctuations of 10 percent, and maintain an output of 24 volts. During subsequent testing, however, it was found that the power supply had become degraded, and was producing output voltages as low as 20.6 volts, or approximately 17 percent degraded. The drop in voltage to the controller resulted in a loss of control signal to the Main Feedwater Regulating Valve supplying feedwater to the 1B Steam Generator. This resulted in the partial closure of the Main Feedwater Regulating Valve.

Following the trip, the 1B Feedwater Regulating Valve was found to be approximately 50 percent open. Failure of the feedwater regulating valve in the 50 percent open position was the cause of both the decrease in S/G level prior to the trip, and the rapid increase in Steam Generator level following the plant trip. Instrument and Control technicians were able to reset the feedwater controller following the event and feedwater regulating valve control was returned to normal.

**ANALYSIS OF THE EVENT**

This event is reportable under 10 CFR 50.73 a.2.iv, as an event or condition that resulted in a manual or automatic actuation of an engineered safety feature (ESF), including the Reactor Protection System.

The plant response to this event is bounded by the accident analysis of the St. Lucie Unit 1 Updated Final Safety Analysis Report (UFSAR), section 15.2.8, "Loss of Normal Feedwater Flow." The actual plant response was more conservative than this analysis due to the following:

- 1) Only one Main Feedwater Regulating Valve was affected by this event. As a result, only one Steam Generator experienced a partial loss of normal feedwater flow, rather than a complete and instantaneous loss of feedwater flow to both Steam Generators, as assumed in the UFSAR analysis. The partial loss of inventory caused by the feedwater regulating valve closure did not significantly degrade primary to secondary heat removal during this event.
- 2) The reactor was manually tripped by utility licensed operators prior to SG level reaching the Reactor Protection System (RPS) setpoint for a low water level trip. The UFSAR analysis assumes that Steam Generator inventory is further reduced to a level corresponding to the automatic reactor trip from the Reactor Protection System (RPS).



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

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
ST. LUCIE UNIT 1	05000335	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	4 OF 6
		95 --	010 --	00	

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**ANALYSIS OF THE EVENT Continued**

Operator actions to secure main feedwater flow to the 1B Steam Generator (SG) mitigated the effects of excessive post trip feed which was caused by the Main Feedwater Regulating Valve remaining in the 50 percent open position. Automatic functions were available to trip the Main Feedwater pumps, had a manual trip not been initiated. Both the Auxiliary Feedwater Actuation System (AFAS) and Steam Bypass Control System (SBCS) functioned as designed following the plant trip.

All plant safety functions were met during this event, and therefore the health and safety of the public were not affected.

**CORRECTIVE ACTIONS**

- 1) Testing was performed on the 1A and 1B Main Feedwater Controller power supplies. As a result of this testing, the 1B Main Feedwater regulating controller power supply was replaced. The 1A Main Feedwater controller power supply tested satisfactorily.
- 2) The Unit 2 Main Feedwater regulating controller power supplies were tested during the Unit 2 refueling outage. These power supplies tested satisfactorily.
- 3) St. Lucie Action Report (STAR) 951925 was generated to determine the cause of the feedwater regulating controller power supply failure. The Preventative Maintenance (PM) program will be evaluated for enhancements as required.
- 4) A plant modification was approved to provide a normal source of power to the Feedwater Regulating System from a vital 120 volt electrical bus. When implemented, this change will provide a primary power supply to the feedwater controller which is less likely to be affected by voltage fluctuations.
- 5) St. Lucie Action Report (STAR) 951901 was generated to evaluate other power supplies in the Feedwater and Condensate system. Power supplies identified by this evaluation will be reviewed for possible inclusion into a PM program.
- 6) The plant will evaluate for corrective action, the cause of the spurious control room alarms received during momentary voltage fluctuations due to large motor starts.



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LICENSEE EVENT REPORT (LER)  
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
ST. LUCIE UNIT 1	05000335	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	5 OF 6
		95	010	00	

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**ADDITIONAL INFORMATION**

Failed Components Identified

Description: 24 VDC Power Supply  
Model: 24D75  
Manufacturer: Acopian

Previous Similar Events

- LER 389-95-002, "Automatic Reactor Trip on Low Steam Generator Water Level due to a Failed Level Transmitter." This event on February 21, 1995, was initiated by a level transmitter failure, which caused the closure of a Main Feedwater Regulating Valve. The transmitter failure was due to a manufacturing design deficiency.
- LER 389-92-004, "Manual Reactor Trip due to Low "A" Steam Generator Level." This event was initiated by the failure of lead/lag circuitry in the feedwater control system which resulted in closure of the Main Feedwater Regulating Valve. The cause was attributed to age related failure of a lead/lag circuitry component.
- LER 335-88-003, "Reactor Trip on Low Steam Generator Water Level due to Main Feed Regulating Valve Equipment Failure." This event, on March 27, 1988, was initiated when a Feedwater Regulating valve positioner malfunctioned causing closure of the valve. The cause of the event was attributed to the failure of the valve positioner.

LICENSEE EVENT REPORT (LER)  
TEXT CONTINUATION

FACILITY NAME (1)  ST. LUCIE UNIT 1	DOCKET  05000335	LER NUMBER (8)		PAGE (3)  6 OF 6
		YEAR 95 --	SEQUENTIAL NUMBER 010 --	

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

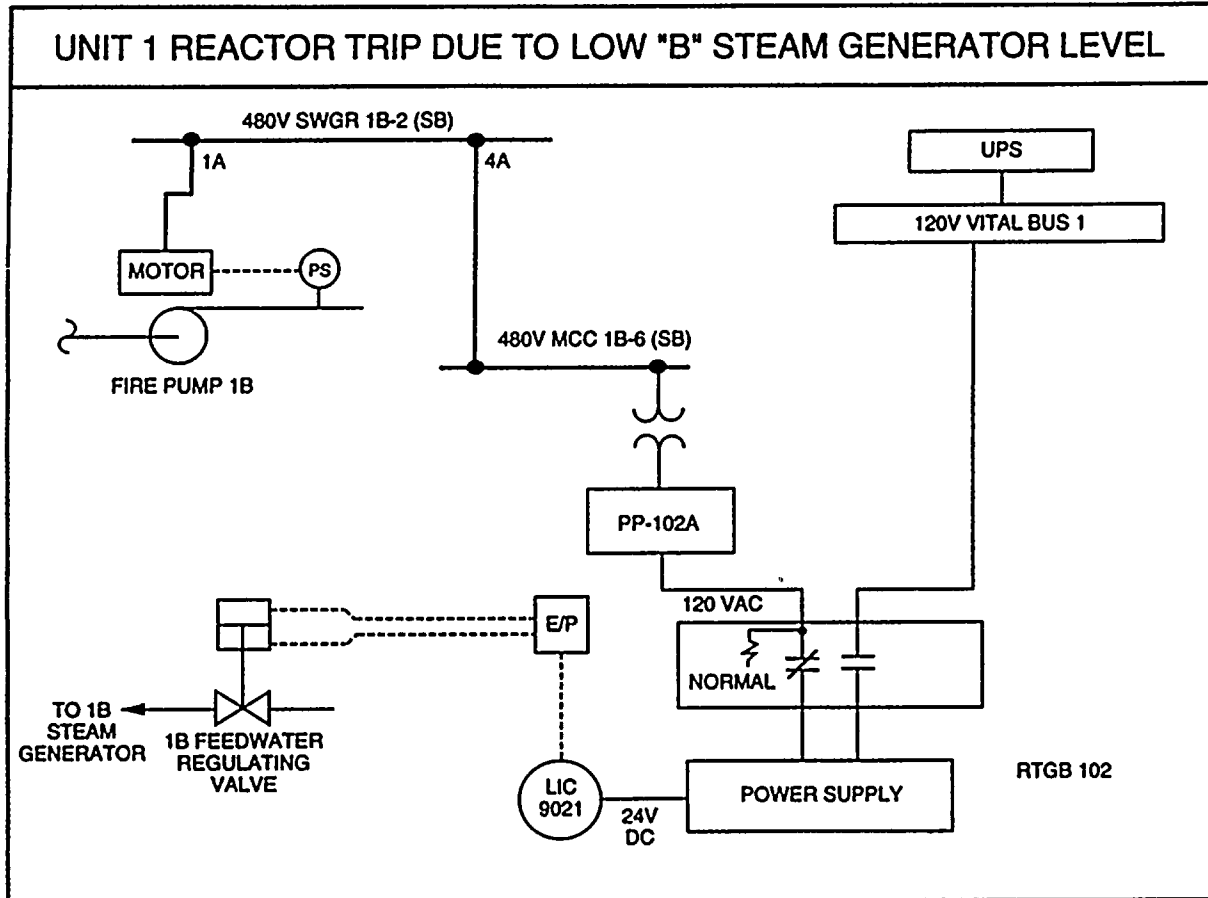


FIGURE 1 - POWER SUPPLY TO 1B FEEDWATER REGULATING SYSTEM