CATEGORY 1

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

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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 96-002-00:on 960222,control element assembly dropped on Unit 1 during periodic testing.Caused by failed silicon control rectifier in upper gripper power switch module.Power switch modules replaced.W/960321 ltr.

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MAR 21 1996

L-96-75
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: 96-002
Date of Event: February 22, 1996
Manual Reactor Trip During Unit Shutdown
Following Dropped Control Element Assembly

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,

W. H. Bohlke
Vice President
St. Lucie Plant

WHB/EJB

Attachment

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

IE22
/11

NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

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

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TITLE (4)

Manual Reactor Trip During Unit Shutdown Following Dropped Control Element Assembly

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
									N/A	N/A
02	22	96	96	-- 002 --	00	03	21	96	FACILITY NAME N/A	DOCKET NUMBER N/A

OPERATING MODE (9)	1
POWER LEVEL (10)	100

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

	20.2201(b)		20.2203(a)(2)(v)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)
	20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
	20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
	20.2203(a)(2)(ii)		20.2203(a)(4)	X	50.73(a)(2)(iv)		OTHER
	20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Edwin J. Benken, Licensing Engineer	(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	AA	SCR	C490	Y					
X	JB	FCV	F130	Y	X	JI	PCV	C635	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 15 single-spaced typewritten lines) (16)

On February 22, 1996, while at 100 percent power, a Control Element Assembly (CEA) dropped on Unit 1 during periodic testing due to a failed silicon control rectifier (SCR). The CEA could not be recovered and a unit shutdown was initiated in accordance with Technical Specification (TS) 3.0.3. An Unusual Event notification was made due to the TS required shutdown and inability to realign the CEA within the action time. At approximately 25 percent power the unit was manually tripped due to increasing Steam Generator (SG) levels caused by a failed main feedwater regulating valve (MFRV). Additionally, a steam bypass control valve operated improperly during the event. The plant was stabilized in Mode 3 and the Unusual Event was terminated.

The cause of the dropped CEA was a failed SCR in the upper gripper power switch module. Entry into TS 3.0.3 was caused by the inability to realign the dropped CEA within the allowable TS action time. The cause of the feedwater regulating valve failure was a leaking instrument air line. The steam bypass control valve improper operation was due to an air booster relay misadjustment.

Significant Corrective Actions: 1) Power switch modules for the CEA were replaced and additional CEA testing was performed. Additional root cause testing of the SCR failure is being performed by the vendor. 2) Air lines were replaced on MFRVs and preventive maintenance (PM) enhancements are being made. 3) Steam Bypass Control System (SBCS) valves were inspected and repaired and actuators and positioners are being upgraded. SBCS procedure enhancements were made.

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DESCRIPTION OF THE EVENT

On February 22, 1996, at 0855, St. Lucie Unit 1 was in Mode 1 at 100% power. The 1B Charging Pump (EIIS:CB), the 1C Component Cooling Water Pump (EIIS:CC) and the 1B Boric Acid Makeup Pump (EIIS:CB) were out of service for repair and preventive maintenance. Operations was performing the Control Element Assembly (CEA) (EIIS:AA) periodic exercise surveillance in accordance with Technical Specification (TS) Surveillance Requirement 4.1.3.1.2 . The St. Lucie Unit 1 CEA configuration consists of 7 regulating and 2 shutdown groups. Regulating CEA groups 1, 2 and 3 were successfully tested during the surveillance. At 1003, while exercising the last CEA in group 4, a previously tested CEA in Group 2, CEA 20, dropped its full length. Turbine power was reduced to match reactor power and stabilized at approximately 93 percent.

Control room utility licensed operators entered TS action 3.1.3.1.e which states: "With one full length CEA misaligned from any other CEA in its group by 15 or more inches, operation in Modes 1 and 2 may continue provided that the misaligned CEA is positioned within 7.5 inches of other CEAs in its group in accordance with the time constraints shown in Figure 3.1-1a." Figure 3.1-1a allows the operators 60 minutes to recover and align the CEA. The control room utility licensed operators also referred to Off-Normal Operating Procedure 1-0110030, " CEA Off-Normal Operation and Realignment," during the event.

At 1030, utility non-licensed Instrument & Control (I&C) personnel reported that there were blown fuses on the power supply to CEA 20 (EIIS:AA). Upon further assessment by I&C personnel, it was determined that repairs could not be completed within the 60 minutes allowed for CEA recovery .. The utility licensed operators entered TS Action 3.1.3.1. f, which states: "With one full length CEA misaligned from any other CEA in its group by 15 or more inches beyond the time constraints shown in Figure 3.1-1a, reduce power to less than or equal to 70 percent of rated thermal power prior to completing action f.1 or f.2."

Action -f.1 Restore the CEA to operable status within the specified alignment requirements, or

Action -f.2 Declare the CEA inoperable and satisfy the shutdown margin requirements of Specification 3.1.1.1. After declaring the CEA inoperable, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6 provided:

a) Within one hour the remainder of the CEAs in the group with the inoperable CEA shall be aligned to within 7.5 inches of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits shown in Figure 3.1-2; Thermal Power level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

b) The Shutdown margin requirement of Specification 3.1.1.1 is determined at least once per 12 hours.

DESCRIPTION OF THE EVENT Continued

At 1100, utility licensed operators began a downpower to less than 70 percent power per TS requirements. CEA 20 could not be recovered and was declared inoperable. The remainder of the CEAs in group 2 could not be aligned to CEA 20 within the one hour allowed by action 3.1.3.1.f.2, due to power dependent insertion limits (PDIL) . The utility licensed operators determined that entry into Technical Specification 3.0.3 was required. This Specification states: "When a Limiting Condition for Operation (LCO) is not met, except as provided in the associated action requirements, within 1 hour actions shall be initiated to place the unit in a Mode in which the specification does not apply by placing it, as applicable in "At least Hot Standby within the next 6 hours..."

At 1200, the control room operators entered TS 3.0.3. The operators continued to reduce reactor power to Hot Standby in order to shut down and repair CEA 20. At 1205, the operations Nuclear Plant Supervisor (Shift Supervisor) declared an Unusual Event based on Increased Awareness due to the inability to realign the CEAs within the required action time of TS 3.1.3.1.f.2. Also, at 1205, Reactor Engineering verified that Shutdown Margin requirements continued to be satisfied.

The 1A Main Feedwater Pump (EIIS:SJ) had been removed from service as part of the normal shutdown procedure and steam generator (SG) levels were being maintained with the 1B Main Feedwater Pump (EIIS:SJ). At approximately 1300 with the plant at 30 percent reactor power, SG levels began to deviate from setpoint. The utility licensed operator attempted to stabilize SG level by adjusting the level setpoint. At approximately 26 percent power, the control room licensed operator placed FCV-9011, the 1A Main Feedwater Regulating Valve (MFRV) controller (EIIS:JB) in manual, and attempted to close the valve. Feedwater flow did not change following the "close" signal and a control room annunciator (E-33) (EIIS:IB) for MFRV low air pressure was received. A utility licensed operator was dispatched to the valve to investigate and reported that the 1A MFRV (FCV-9011) (EIIS:JB) had an air leak at the upper side actuator.

At 1313, the 1A steam generator water level continued to increase to approximately 82 percent and the control room utility licensed operators manually tripped the reactor. The operators entered Emergency Operating Procedure (EOP) - 1, "Standard Post Trip Actions." At 1316, the 1B Main Feedwater Pump tripped due to high steam generator water level (approximately 91 percent) in SG 1A. Steam generator level subsequently returned to normal and auxiliary feedwater (AFW) (EIIS:BA) was initiated to provide feedwater flow to the steam generators. At 1320, in accordance with EOP - 1, it was determined that all safety functions were being met . The operators then transitioned from EOP -1 to EOP-2, "Reactor Trip Recovery" procedure.

Due to the excess feedwater flow to 1A steam generator, coupled with the reactor trip, the Reactor Coolant System (RCS) temperature was reduced causing pressurizer level to decrease to approximately 24 percent. The Control Room operators isolated RCS letdown (EIIS:CB) at 1323 and took manual control of Charging Pumps to restore pressurizer level per procedure. At 1330, with the reactor in Mode 3, the Unusual Event was terminated in accordance with procedure. At 1415, charging and letdown was restored to automatic operation. The unit was stabilized in Mode 3.

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DESCRIPTION OF THE EVENT Continued

The following additional equipment deficiencies were observed during the Unit shutdown:

Steam Bypass Control System (SBCS) (EIIS:JI) pressure control valve PCV-8801 (EIIS:JI) is one of four steam bypass valves which modulate in sequence to control main steam pressure and RCS average temperature. The valves quick-open on a reactor trip, and remain closed during low power operation. Control room operators noted that PCV-8801 was not modulating properly and it was placed in manual control following the trip. Additionally, SBCS valve PCV-8804 (EIIS:JI) was removed from service during the shutdown due to a control signal discrepancy. A subsequent inspection of PCV-8804 revealed a worn spool valve within the positioner.

CAUSE OF THE EVENT

The unit shutdown was required in accordance with Technical Specifications due to an unrecoverable, dropped CEA. A manual reactor trip was initiated by utility licensed operators to mitigate increasing steam generator level during the unit shutdown. Event causes and additional component failures are discussed in detail below.

Control Element Assembly (CEA) Drop

The failure of a silicon control rectifier (SCR) in the upper gripper power switch module was determined to be the cause of the CEA drop. The A phase SCR for the CEA 20 Upper Gripper Coil was found in a shorted condition. The shorted SCR created a phase fault which caused the 240V supply fuses to open. The Upper Gripper Coil de-energized resulting in the CEA drop.

There are five gripper coils used to position each CEA. Each of the five coils is energized through SCRs which rectify the 240V AC power source and control the electrical current sent to each coil. The upper gripper coil is continuously energized and used to hold the CEA in place when a CEA movement demand is not present.

No history of failure of SCRs in the power switch modules has occurred at St. Lucie Plant or was identified by Combustion Engineering (CE) as occurring at other CE plants. Additionally, the Nuclear Plant Reliability Data System (NPRDS) was reviewed to identify any similar reported failures. No history of failure trends was established from this review.

CAUSE OF THE EVENT Continued

Main Feedwater Regulating Valve Failure

The steam generator level transient resulted from erratic operation of the "A" Main Feedwater Regulating Valve (FCV)-9011. FCV-9011 operated erratically due to the failure of a copper instrument air line between the lockup regulator and actuator upper air chamber. Excessive air leakage from the actuator resulted in unstable control of the valve and the corresponding steam generator levels. A post-event inspection of the air line confirmed high cycle fatigue as the failure mechanism. The preventive maintenance program for these valves did not require that tubing be replaced on a periodic basis.

Steam Bypass Control System (SBCS) Valves

Steam Bypass Control System valve PCV-8801 did not modulate properly during the unit shutdown due to an air booster relay which was out of adjustment. This relay is a pneumatic gain device which amplifies the output of the signal air to the actuator diaphragm. The relay misadjustment was due to insufficient procedural guidance addressing the performance of steam bypass control valve booster relay tuning for dynamic conditions.

Steam Bypass Control System Valve PCV-8804 was found to have an output signal from its positioner with no input demand because of a worn spool valve within the positioner which allowed supply air to bleed off to the non-vented signal line. Additionally, during the inspection performed on the PCV-8804 positioner, a powdered black substance was found which was subsequently determined to have originated from a rubber diaphragm in the actuator. Engineering and maintenance personnel determined that the as found condition of PCV-8804 would not have prevented the valve from operating as designed.

ANALYSIS OF THE EVENT

This event is reportable under 10 CFR 50.73 (a) (2) (i) (A), "the completion of any nuclear plant shutdown required by the plant's Technical Specifications and under 10 CFR 50.73 (a)(2)(i)(B), "any operation or condition prohibited by the plant's Technical Specifications..." Additionally, the event is reportable under 10 CFR 50.73 (a) (2) (iv), "any event or condition that resulted in the manual or automatic actuation of any engineered safety feature (ESF) including the reactor protective system..."

Limiting Condition for Operation (LCO) 3.0.3 delineates the time limits for placing the unit in a non-applicable Mode when plant operation cannot be maintained within the Technical Specification LCO and its requirements. Once it was determined that CEA 20 could not be aligned to the remaining CEAs in group 2 within the allowed action time of TS 3.1.3.1 f.2, the operators exited LCO 3.1.3.1.f.2 and entered TS 3.0.3. Control room operators complied with TS requirements at all times during the event, and at no time were Technical Specifications violated.

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ANALYSIS OF THE EVENT Continued

The plant response to the dropped CEA is bounded by the accident analysis of the St. Lucie Unit 1 Updated Final Safety Analysis Report , Section 15.2.3, "CEA Drop Accident." Bounding End-of-Cycle kinetics parameters are used in the analysis and the reactivity insertion is selected to conservatively bound the most reactive CEA being inserted. The analysis described in Section 15 assumes a full length CEA drop into the core with constant turbine demand and without any operator action. Reactor power is assumed to return to 100 percent by negative moderator feedback. The actual plant response was more conservative than that described in the analysis, as the reactor was not permitted to return to full power, but rather, turbine load was reduced to match reactor power following the CEA insertion.

The 1A MFRV erratic operation resulted from an air leak at the upper side actuator. This caused the SG water level to increase. Based on increasing steam generator water level associated with the valve malfunction, the utility licensed operators, using conservative operational decision making, manually tripped the reactor. In the absence of operator action, a SG high water level turbine trip would have occurred which in turn would have resulted in an automatic reactor trip. This trip is designed to protect the turbine from moisture carry over. The manual reactor trip was an uncomplicated reactor trip since all reactor safety functions were verified to be met. SG levels were controlled and maintained using auxiliary feedwater following the Unit trip, and did not exceed 93 percent narrow range during the event.

The SBCS valves are air operated globe valves which function to limit the pressure rise in the steam generators following a turbine trip or load rejection to avoid lifting the main steam safety valves (MSSV). Should the SBCS valves fail to operate properly, the atmospheric dump valves provide a means for controlled cooldown of the Reactor Coolant System. Because the ASME Code main steam safety valves (MSSV) provide the ultimate overpressure protection for the steam generators, the Steam Bypass Control System has no safety function and therefore is not designed to the requirements applicable to protection systems. The main steam safety valves did not lift following the manual trip. Manual control of PCV-8801 and the isolation of PCV-8804 did not adversely impact the maintenance of safety functions.

All safety functions were maintained during the event and the health and safety of the public were not affected.

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CORRECTIVE ACTIONS

Control Element Assemblies

1. Instrument & Control Maintenance replaced power switch modules to control element assembly (CEA) 20. The load transfer coil frequency suppression resistor and diode were also replaced. The upper gripper low frequency resistor and diode were verified as good. The cables to the upper gripper and load transfer coils were meggered to verify insulation integrity and coil resistances were verified as acceptable. Post maintenance testing of CEA 20 was performed resulting in an acceptable condition.

2. The failed upper gripper power switch module was sent to the vendor for inspection and root cause determination.

Main Feedwater Regulating Valve

1. Maintenance replaced the instrument air lines on the actuators for both Main Feedwater Regulating valves FCV-9011 and FCV-9021. The failed copper tubing was sent to the FPL Materials Lab for a failure analysis. The analysis confirmed high cycle fatigue as the failure mechanism.

2. Critical valves on both St. Lucie Unit 1 and Unit 2 were inspected for similar copper tubing leaks. No similar tubing failures were found.

3. FPL identified and reviewed additional, critical valve actuators. Preventive maintenance enhancements are being made, to include the periodic replacement of air line tubing in conjunction with actuator overhaul, to preclude further failures of this type.

Steam Bypass Control System Valves

1. Maintenance repaired PCV-8801 and PCV-8804. Each SBCS valve was checked and verified to operate properly.

2. Chemistry performed sample analysis and verified that Instrument Air System quality was satisfactory. This check was performed due to a black residue that was found during the trouble shooting of PCV-8804.

3. A procedure change was made to provide specific guidance for the setting of SBCS air booster relays to address dynamic conditions.

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

CORRECTIVE ACTIONS Continued

Steam Bypass Control System Valves - continued

4. The actuators and positioners for all Unit 1 SBCS valves are being upgraded during the upcoming Unit 1 refueling outage. This activity was planned prior to this event to improve the control characteristics of the SBCS. The Unit 2 valves were overhauled during the most recent Unit 2 refueling outage.

ADDITIONAL INFORMATION

Failed Components Identified

Description: CEA Power Switch Module - Silicon Control Rectifier
Model: N-0045
Manuf: Combustion Engineering

Description: Actuator for FCV-9011
Model 474-5-16
Manuf: Fisher Controls

Description: Actuator for PCV-8801
Model: D-100-160(8")
Manuf: CVC (Copes Vulcan)

Description: Positioner for PCV-8804
Model: 5321030-2
Manuf: Bailey Meter

Previous Similar Events

LER 335-90-008 Reactor Shutdown due to Unrecoverable Dropped CEA caused by an Improperly Installed Fuse." In this event, the failure of a fuse which was improperly installed resulted in a dropped CEA and a unit shutdown. An Unusual Event was declared and the Unit was shutdown in accordance with Technical Specifications.