CATEGORY 1

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FACIL:50-400 Shearon Harris Nuclear Power Plant, Unit 1, Carolina 05000400

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CLARK,B.H.	Carolina Power & Light Co.
RECIP.NAME	RECIPIENT AFFILIATION

SUBJECT: LER 99-004-00:on 990312,unit trip was noted.Caused by degraded condition of SG water level flow control valve. Replaced positioners on all three FW regulating valves.With 990408 ltr.

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Serial: HNP-99-063
10CFR50.73

SHEARON HARRIS NUCLEAR POWER PLANT UNIT 1
DOCKET NO. 50-400
LICENSE NO. NPF-63
LICENSEE EVENT REPORT 1999-004-00

Sir or Madam:

In accordance with 10CFR50.73, the enclosed Licensee Event Report is submitted. This report describes a condition which resulted in an automatic trip of the reactor and the automatic actuation of various Engineering Safety Features Actuation Systems.

Sincerely,

B.H. Clark
General Manager
Harris Plant

CWF/cwf

Enclosure

c: Mr. J. B. Brady (HNP Senior NRC Resident)
Mr. R. J. Laufer (NRC - NRR Project Manager)
Mr. L. A. Reyes (NRC Regional Administrator, Region II)

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NRC FORM 366 (6-1998) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 06/30/2001
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 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-6 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. If an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4)
Unit trip due to the degraded condition of a steam generator water level flow control valve.

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 05000
3	12	1999	1999	- 004 -	00	4	8	1999	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)							
			20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
POWER LEVEL (10)	100		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
			20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		or in NRC Form 366A

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Carey W. Fleming, Principal Analyst - Licensing	(919) 362-2313

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
E	JB	FCV	Bailey	Y					

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

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

At 06:39 on March 12, 1999, the unit experienced an automatic reactor trip from 100% power. The unit tripped on a high water level trip signal on one of the three steam generators. The 'hi-hi' steam generator water level signal trip setpoint is 82.4 percent on the narrow range level scale. Just prior to the automatic trip, the 'C' Steam Generator (SG) water level was observed to be increasing in an uncontrolled manner with its Feedwater Regulating Valve (FRV) in automatic control. The control room operators took manual control of the 'C' FRV, in an attempt to restore water level to its normal range. The FRV response, as observed from the controller output signal and the resulting feed flow indications, was slower and not as uniform as expected. After approximately eight minutes of attempting to control the 'C' steam generator water level in manual, the performance of the valve positioner degraded to the point where operator control was highly questionable. The operators had set a limit, at which they would initiate a manual reactor trip; however, an automatic trip occurred prior to their taking the planned action.

The most probable failure mechanism for the 'C' FRV is a combination of a loose stroke adjustment screw on the air-operated valve positioner and a sticking pilot valve within that same positioner. This degradation in the material condition of the valve's positioner is attributed to an inadequate preventative maintenance program/schedule for the Feedwater Regulating Valves. Corrective actions that have been taken include: 1) replacing the positioners on all three Feedwater Regulating Valves, including a verification that the stroke adjustment screw on each is not loose, and 2) Determining that similar air operated valve positioners do not have a potential for similar failures which may jeopardize plant safety or reliability. Planned corrective actions include reviewing and revising, as applicable, the preventative maintenance schedules for Bailey air operated valve positioners.

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

I. DESCRIPTION OF EVENT

At 06:39 on March 12, 1999, the unit experienced an automatic reactor trip from 100% power. The unit tripped on a high water level trip signal on one of the three steam generators. The 'hi-hi' steam generator water level signal trip setpoint is 82.4 percent on the narrow range steam generator water level scale. This trip signal causes the main turbine to trip and a main feedwater isolation signal to be sent to various components for the purpose of limiting any further increase in water level, thereby protecting the turbine and other main steam system components from water damage. When the plant is above approximately 10 percent power, the turbine trip signal will cause an automatic reactor trip.

Just prior to the automatic trip, the 'C' Steam Generator (SG) water level was observed to be increasing in an uncontrolled manner with its Feedwater Regulating Valve (FRV) (EIIS: JB, FCV) in automatic control. The control room operators took manual control of the 'C' FRV, in an attempt to restore water level to its normal range. The FRV response, as observed from the controller output signal and the resulting feed flow indications, was slower and not as uniform as expected. After approximately eight minutes of attempting to control the 'C' steam generator water level in manual, the performance of the valve positioner degraded to the point where operator control was ineffective (i.e., some control was available, but the control characteristics were not as expected). The operators had set a limit, at which they would initiate a manual reactor trip; however, an automatic trip occurred prior to their taking the planned action. The reactor trip recovery proceeded normally with minor equipment deficiencies noted on some non-safety secondary systems. Following the trip, level in the 'C' steam generator was reduced to its post-trip level band by a combination of 'shrink' and steaming for decay heat removal.

The reactor was not manually tripped prior to automatic action due to the Unit Senior Control Operator (USCO) establishing a verbal limit which was only 0.4 percent of narrow range level indication less than the trip setpoint. Both the investigation team and Operations management have determined that this guidance provided by the USCO was not optimal. Procedures existing at the time of the event did not specify a manual trip limit for steam generator levels for the given situation.

An investigation team devised a plan and performed testing and inspections on the 'C' FRV controller, as well as inspections and comparisons of the 'A' and 'B' valve controllers. When the 'C' FRV was disconnected from its operator, an abnormal (e.g., jerky) stroke of the valve persisted on the operator (i.e., a positioner problem and not a mechanical binding of the valve itself). Disassembly of the valve revealed a loose stroke adjustment screw, which was not the case in the 'A' and 'B' positioners. Inspection of a pilot valve assembly in the 'C' positioner indicated that dirt build-up and wear on the pilot valve stem. Although the pilot valve stems for 'A' and 'B' FRV positioners also experienced some wear, the investigation team still considered the wear issue to be a likely cause for the symptoms observed. The team believes that constant modulation and vibration of the 'C' FRV without more frequent preventative maintenance was the cause of the positioner failure.

II. CAUSE OF EVENT

The most probable failure mechanism for the 'C' FRV failure is a combination of a loose stroke adjustment screw on the air-operated valve positioner and a sticking pilot valve within that same positioner. The sticking pilot valve may have been caused by excessive wear or dirt build-up. This degradation in the material condition of the valve's positioner is attributed to an inadequate preventative maintenance program/schedule for the Feedwater Regulating Valves.

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

III. SAFETY SIGNIFICANCE

There were no actual safety consequences as a result of this event. All systems required to limit an overfeeding event of a steam generator, as enumerated in FSAR sections 15.1.2 and 15.0.8, remained operable throughout the event. Additional features available to protect the unit from an over-feeding event are the overtemperature and overpower delta-T reactor trips. These features remained available, and were not challenged on the March 12, 1999 trip. No safety limits were exceeded and the event neither initiated nor exacerbated any radiological releases.

This report is being submitted pursuant to the criteria of 10 CFR 50.73(a)(2)(iv) for an unplanned automatic actuation of the Reactor Protection System (RPS) and the unplanned, automatic Engineered Safety Features (ESF) actuations as follows:

A) The automatic trip of the turbine and feedwater isolation signal from the "hi-hi steam generator level trip signal" (P-14);
B) The automatic reactor trip, which is required following a turbine trip with the plant power above the P-7 setpoint (approximately 10 % power);
C) The automatic start of the two motor driven auxiliary feedwater pumps, indirectly due to the feedwater isolation signal (i.e., loss of both running main feed pumps on the feedwater isolation signal);
D) The automatic start of the turbine driven auxiliary feedwater pump on the low-low steam generator levels in the 'A' and 'B' steam generators, due to the main feed isolation signal and the expected water level 'shrink' following the trip.

IV. CORRECTIVE ACTIONS

Corrective actions that have been taken:

1. Replaced the positioners on all three Feedwater Regulating Valves, including a verification that the stroke adjustment screw on each is not loose.
2. Determined that similar air operated valve positioners do not have a potential for similar failures which may jeopardize plant safety or reliability.

Planned corrective actions:

3. Review and revise, as applicable, the preventative maintenance schedules for Bailey air operated valve positioners (by 8/31/99).

V. SIMILAR EVENTS

This is the first trip of the Harris Plant involving a 'hi-hi' steam generator water level. Harris Plant LER 97-001-00 describes a reactor trip on low steam generator water level, due to degraded equipment on the positioner of a major valve in the main feedwater system. This LER is distinguishable from that event in that the manufacturer and make of the valves are very dissimilar. Corrective actions from that event had no effect on the FRVs for steam generator water level control.

Investigations following the trip did not reveal any industry operating experience which would be directly on point with this occurrence.