REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR: 8710160270 DOC. DATE: 87/10/13 NOTARIZED: NO DOCKET #
FACIL: 50-250 Turkey Point Plant, Unit 3, Florida Power and Light C 05000250

AUTH. NAME	AUTHOR AFFILIATION
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WOODY, C. O.	Florida Power & Light Co.

RECIP. NAME	RECIPIENT AFFILIATION

SUBJECT: LER 87-023-00: on 870913, safety injection sys actuation & subsequent reactor trip occurred. Caused by component failure along w/personnel error. Response team formed to investigate causes & initiate corrective action. W/871013 ltr.

DISTRIBUTION CODE: IE22D COPIES RECEIVED: LTR 1 ENCL 1 SIZE: 6
TITLE: 50.73 Licensee Event Report (LER), Incident Rpt, etc.

NOTES:

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	PD2-2 LA	1	1	PD2-2 PD	1	1
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INTERNAL:	ACRS MICHELSON	1	1	ACRS MOELLER	2	2
	AEOD/DOA	1	1	AEOD/DSP/NAS	1	1
	AEOD/DSP/ROAB	2	2	AEOD/DSP/TPAB	1	1
	ARM/DCTS/DAB	1	1	DEDRO	1	1
	NRR/DEST/ADS	1	0	NRR/DEST/CEB	1	1
	NRR/DEST/ELB	1	1	NRR/DEST/ICSB	1	1
	NRR/DEST/MEB	1	1	NRR/DEST/MTB	1	1
	NRR/DEST/PSB	1	1	NRR/DEST/RSB	1	1
	NRR/DEST/SGB	1	1	NRR/DLPQ/HFB	1	1
	NRR/DLPQ/QAB	1	1	NRR/DOEA/EAB	1	1
	NRR/DREP/RAB	1	1	NRR/DREP/RPB	2	2
	NRR/~~DRIS~~/SIB	1	1	NRR/PMAS/ILRB	1	1
	~~REG FILE~~ 02	1	1	RES DEPY GI	1	1
	RES TELFORD, J	1	1	RES/DE/EIB	1	1
	RGN2 FILE 01	1	1			
EXTERNAL:	EG&G GROH, M	5	5	H ST LOBBY WARD	1	1
	LPDR	1	1	NRC PDR	1	1
	NSIC HARRIS, J	1	1	NSIC MAYS, G	1	1

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NRC Form 366
(9 83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Turkey Point Unit 3	0 5 0 0 0 2 5 0	1 OF 0 5

TITLE (4) Unit 3 Safety Injection and Reactor Trip Due to Failed High Steam Flow Channels and an Actual Low Average Reactor Coolant Temperature After a Load Reduction for Turbine Testing

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)
									N/A	0 5 0 0 0
0 9	1 3	8 7	8 7	— 0 2 3	— 0 0	1 0	1 3	8 7	N/A	0 5 0 0 0

OPERATING MODE (9): 2

POWER LEVEL (10): 0 0 5

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)	
Randall D. Hart, Licensing Engineer	3 0 5	2 4 6 - 6 5 5 9

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

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS
X	W I	P S V	T 0 2 0	Y	X	J C	F T	R 3 6 9	Y
X	J C	F T	R 3 6 9	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 fifteen single-space typewritten lines)* (16)

On September 13, 1987, Unit 3 experienced a safety injection (SI) system actuation and subsequent reactor trip from approximately 5% power. On September 12, 1987, 1 of 2 high steam flow protection channels on the 3A steam generator (SG), FT-3-475, failed low and was taken out of service as per procedure. On September 13, 1987, a load reduction was commenced for Unit 3 to perform a turbine overspeed test as per operating procedure (OP) 8004.1, Turbine Generator Overspeed Trip Test. The test was begun with the low reactor coolant system (RCS) average temperature (Tave) input to the SI logic for a SI signal on low Tave on 2 out of 3 RCS loops coincident with high steam flow on 2 out of out of 3 SGs, alarmed on the A, B and C RCS loops. Upon completion of the test a high steam flow transmitter for the 3C SG, spiked, began cyclic operation and subsequently failed which locked in the high steam flow signal for the 3C SG and completed the logic for a SI actuation and subsequent reactor trip. The cause of the trip was component failure along with personnel error. An event response team was formed to determine root cause and corrective actions. A review was performed to assess proper plant operation which determined that the plant response to the transient was as expected. After completion of necessary reviews and equipment repairs the unit was returned to service on September 22, 1987.

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

EVENT:

On September 13, 1987, while Unit 3 was at approximately 5% power, a safety injection (SI) system actuation and subsequent reactor trip occurred. At 2045 on September 12, 1987, 1 of 2 high steam flow protection channels on the 3A steam generator (SG), FT-3-475, failed low and was taken out of service as per procedure. The protective functions fed by this flow transmitter were placed in the tripped mode.

At 0810 on September 13, 1987, a load reduction was commenced for Unit 3 to perform a turbine overspeed test as per operating procedure (OP) 8004.1, Turbine Generator Overspeed Trip Test. At 0829, the alarms for the reactor coolant system (RCS) low average temperature (Tave) input to the SI logic came in on the A, B, and C RCS loops. The SI logic is satisfied with high steam flow on 1 out of 2 steam flow channels in 2 out 3 SGs coincident with low Tave on 2 out of 3 RCS loops or low SG pressure on 2 out of 3 SGs. At this time, it was decided to continue with the test since the decrease in Tave had been stopped and the trend was starting to increase. At 0831, the generator was taken off the line and the test begun. At 0835, the turbine was tripped on overspeed as per OP 8004.1 and the high steam flow transmitter for the 3C SG, FT-3-494 began cyclic operation. After cycling several times, the output fuse of the high steam flow comparator, FC-3-494, which locked in the high steam flow signal for the 3C SG and completed the logic for a SI actuation which also resulted in a reactor trip. No actual SI flow was delivered to the RCS. Upon receipt of the SI and reactor trip the Operations personnel entered emergency operating procedure (EOP) 3-EOP-E-0, Reactor Trip or Safety Injection. At 0838, they transitioned to 3-EOP-ES-1.1, SI Termination. At 0839, it was discovered that solenoid valve (SV) SV-3-6275B-1 , 3B SG blowdown isolation bypass valve had a dual position indication and was declared out of service. The unit was subsequently stabilized in hot standby and an event response team was formed to review the event for root cause and corrective actions.

CAUSE OF EVENT:

Investigation of this event revealed several contributing factors to the root cause.

a) An investigation into the cause of the failure of FT-3-475 revealed that the leads at the transmitter were rolled. The leads were returned to their correct configuration and a loop check was performed to verify proper operation. The investigation of the rolled leads identified that FT-3-475 was worked on by Construction as a part of plant change/modification (PC/M) 87-093, Acceptance Criteria and Installation Details for Raychem Splices. The electrical splices for FT-3-475 were worked as per the PC/M and turned over to the Start Up Department as a partial turnover. The post modification testing was done by the Instrumentation and Control (I&C) Department on May 19, 1987. Subsequent inspection by Construction Quality Control personnel identified a bend radius that did not meet the acceptance criteria of the PC/M. A nonconformance report (NCR) was issued for engineering review and determination of appropriate corrective actions. The NCR was dispositioned and the repairs

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done by construction. During this work the leads were at the transmitter were rolled. After the work was completed, no turnover was made to the Start Up Department. Since no turnover was made to Start Up, no post modification testing was done and this condition was not discovered until the unit entered a mode where the transmitter was required to work. If any other transmitter had experienced a similar incident, it would have exhibited a similar response.

b) Another contributing factor was the rapid shutdown of the unit in preparation for the turbine overspeed test. This resulted in reactor power being greater than turbine load causing the reactor operator to borate the reactor to bring down reactor power. This resulted in Tave decreasing to below the low Tave SI setpoint of 543 degrees Fahrenheit on the three loops.

c) The low Tave by itself would have not resulted in the SI system actuation. However, once Tave was stabilized, it was decided to commence the turbine overspeed test even though the low Tave signal input to SI was alarmed along with one half of the rest of the signal on high steam flow. In this condition if another high steam flow transmitter on another SG alarmed, the logic for a SI actuation would be completed and the SI system would actuate. At this time the Plant Supervisor-Nuclear decided to start the test because he felt that Tave was stabilized and increasing and the fact that the turbine overspeed test had to be completed within 15 minutes of reducing generator load below 70 megawatts. This is done to limit cool down of the turbine generator in order to simulate as close as possible to actual load conditions and is a requirement to maintain the warranty on the turbine generator. Also, if the test could not be performed within 15 minutes, the unit would have to return to 30% power and hold for 8 hours before the test could be performed again. This resulted in a perceived necessity to perform the test in a rapid manner.

d) The failure of FT-3-494 which completed the SI logic resulted from vibration in the sensing lines for the transmitter. This vibration caused the relays to begin cycling which resulted in a blown output fuse in the high steam flow comparator after several cycles. The blown fuse locked in the last input to the SI system actuation logic and resulted in the initiation of the SI system and reactor trip.

ANALYSIS OF EVENT:

A post trip review was performed to assess the proper operation of safety related equipment. The review verified that the safety related equipment that started upon initiation of the event operated as expected. The malfunction of SV-6275B-1 was determined to be the result of bad wires going into the reed switch for the open light indication. FT-3-494 cycled several times before blowing the output fuse in FC-3-494 and locking in the signal. The signal did not lock in during the cycling due to the high rate of speed (< 60msec). The cycling was too fast for the next stage of protection/safeguards relays to pick up.

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In addition to the sensing line vibration concern it was discovered that the output fuse that blew in FC-3-494 was a 1/4 amp fuse instead of a 3/8 amp fuse as required. The fuse size did not have any significant effect on the fuse burn out. I&C performed an inspection of the other fuses in the reactor protection and safeguards circuitry to verify proper fuse size. Discrepancies that were found were corrected and further evaluations of the discrepancies are being conducted to determine their significance.

This reactor trip was compared with a similar trip that had occurred on June 27, 1986 and it was determined that system responses were as expected. The only differences were the power level and turbine generator status at the time of the trip. The post trip review also established that the transient behavior of pertinent plant parameters for the RCS and SGs responded as expected for a reactor trip of this kind. Specifically, the RCS pressures and temperatures were determined to have followed an expected pattern based upon the conditions leading up to the transient. Based on the above, the health and safety of the public were not affected.

CORRECTIVE ACTIONS:

1) An event response team was formed to investigate the cause of the SI actuation and reactor trip to identify root causes and initiate corrective actions.

2) Administrative Site Procedure (ASP) ASP-8, Corrective Action, will be revised to require that any discrepancy report and Nonconformance report requiring work that includes specific testing requirements, affects a system pressure boundary, or mechanically or electrically alters a system after acceptance shall be routed to the Start Up Department for appropriate testing requirements.

3) This event will be reviewed by the Training Department to determine appropriate training requirements and methods.

4) OP 8004.1 will be revised to include precautions to describe the possible results of performing this test with reactor protection and/or safeguards circuits alarmed.

5) A non-conformance report (NCR), NCR-87-211, was written to have engineering evaluate the problem with the vibration in the sensing lines and provide a method of repair. The interim repair was to have the high and low pressure lines wrapped together with ty-raps to help dampen the vibration. This was done for both transmitters on each steam generator for unit 3.

6) In addition to the ty-raps implemented in corrective action 5 above, a design equivalent engineering package (DEEP) 87-328 is currently being implemented on unit 3. This DEEP will implement modifications to the main steam flow transmitter tubing and supports to reduce the vibration problem. This DEEP is intended to be a more permanent interim corrective action. The long term solution to the vibration problem is currently under study by engineering.

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7) An administrative procedure is currently being reviewed that will provide guidance for the administrative control of fuse replacement.

8) A post trip review was completed which verified that the plant response to this event was as expected for a reactor trip of this kind. Following completion of the reviews of this event and completion of the required corrective actions, the Unit was returned to service at 2248 on September 22, 1987.

ADDITIONAL DETAILS:

The SG blowdown isolation valve is manufactured by Target Rock, model number 300525-1. The steam flow transmitters were manufactured by Rosemount, model number 1153 DD6.

Similar Occurrences: None

P. O. BOX 14[illegible], JUNO BEACH, FL 33408-0420

FPL

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1987 SEP -6 A 9:58

OCTOBER 1 3 1987
L-87-407
10 CFR 50.73

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

Gentlemen:

Re: Turkey Point Unit 3
Docket No. 50-250
Reportable Event: 87-23
Date of Event: September 13, 1987
Unit 3 Safety Injection and Reactor Trip Due to Failed
High Steam Flow Channels and an Actual Low Average
Reactor Coolant Temperature After a Load Reduction for Turbine Testing

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,

R. J. Acosta for

C. O. Woody
Group Vice President
Nuclear Energy

COW/SDF/gp

Attachment

cc: Dr. J. Nelson Grace, Regional Administrator, Region II, USNRC
Senior Resident Inspector, USNRC, Turkey Point Plant

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