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FACIL:50-251 Turkey Point Plant, Unit 4, Florida Power and Light C 05000251
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RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 93-003-00:on 930816,Turbine Trip/RT occurred due to personnel error in which high pressure feedwater heaters were not returned to svc correctly.Counseled involved operators & revised procedure.W/930907 ltr.

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L-93-224
10 CFR 50.73

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

Gentlemen:

Re: Turkey Point Unit 4
Docket No. 50-251
Reportable Event: 93-003-00
Turbine Trip/Reactor Trip Due to High Steam Generator Level

The attached Licensee Event Report 251/93-003-00 is being provided in accordance with 10 CFR 50.73 (a)(2)(iv).

If there are any questions, please contact us.

Very truly yours,

T. F. Plunkett
Vice President
Turkey Point Nuclear

TFP/CLM/cm

enclosure

copies:

Stewart D. Ebneter, Regional Administrator, Region II, USNRC
Tom P. Johnson, Senior Resident Inspector, USNRC, Turkey Point

IE22
1/1

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
TURKEY POINT UNIT 4	05000251	1 OF 4

TITLE (4) TURBINE TRIP/REACTOR TRIP DUE TO HI-HI STEAM GENERATOR LEVEL

EVENT DATE (5)			LER NUMBER(6)			RPT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MON	DAY	YR	YR	SEQ #	R#	MON	DAY	YR	FACILITY NAMES	DOCKET # (S)
08	16	93	93	003	00	09	07	93		

OPERATING MODE (9)	1	10 CFR 50.73(a)(2)(iv)
POWER LEVEL (10)	28	

LICENSEE CONTACT FOR THIS LER (12)

C. L. Mowrey, Licensing OEF Engineer/Analyst	TELEPHONE NUMBER
	305-246-6204

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	NPRDS?	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	NPRDS?

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

ABSTRACT (16)

On August 16, 1993, Turkey Point Unit 4 was in Mode 1 at 28% power. The two high pressure feedwater heaters were being placed in service. The restoration of the feedwater heaters was not done smoothly, and feed flow to the steam generators was severely curtailed. The loss of flow was recognized, an extra condensate pump and feed pump were started, and steam generator levels began to recover. Simultaneously, both feedwater heaters were completely restored to service in an attempt to aid in the recovery of feed flow. As a result steam generator levels rose faster than the reactor operators could control them. At 2233, the turbine tripped on high-high level in the 4C steam generator, which generated an automatic reactor trip.

The cause of the event was personnel error in that the high pressure feedwater heaters were not returned to service correctly. Contributors were an inadequate pre-evolution briefing, inadequate knowledge of the operation of the heater inlet valves, inadequate communications, and existing equipment failures which required both high pressure feedwater heaters to be valved out.

The operators involved were counselled, the procedure is being revised, management expectations for briefings are being emphasized, and operators are being trained on the operation of the three-way inlet valves.

I. DESCRIPTION OF THE EVENT

On August 16, 1993, Turkey Point Unit 4 was in Mode 1 at 28% power. The unit was being returned to service following a shutdown several days earlier to repair a small leak on the pressurizer [AB:pzr] spray line. About one hour prior to shift change, the Nuclear Watch Engineer (NWE) (licensed senior operator) and the Turbine Operator (TO) (non-licensed utility operator) were restoring the high pressure feedwater heaters [SJ:hx] to service. Each heater is restored using a separate section of procedure 4-OP-081.1, Feedwater Heater Extraction Steam, Vents, and Drains Valve Alignment. Both trains had been removed from service, and the 6A high pressure feedwater heater was being restored. The NWE was called away from the evolution to investigate another condition, and the TO continued the evolution. To save time, he initiated restoration of the 6B high pressure feedwater heater simultaneously.

The inlet valve to each high pressure feedwater heater is a three-way valve called the 6A(6B) FW HTR Condensate Normal/Bypass Valve [SJ:v]. The valve directs flow from the discharge of the main feedwater pumps [SJ:p] either through the heater (Normal), around the heater (Bypass), or some portion to each path. The procedure calls for this valve to be cracked off its seat (assuming the inlet valve is initially positioned to bypass full flow around the heater), until the heater is filled and vented, and heater inlet and outlet temperatures have equalized. Then the procedure directs that the outlet valve be opened, and the inlet valve be moved to its "Normal" position.

The design of the inlet valve is such that it is difficult to determine when it is "just cracked off its seat," in fact, it appears that the TO may have changed the valve position from full bypass to full normal, and that what he thought was "seat breakaway" resistance was actually the resistance of the valve seating in the full Normal position. The result of this mispositioning would not normally be noticed, since the opposite train would be either fully bypassed or fully restored. In this case, since both trains were being restored at the same time, both inlet valves were repositioned at the same time. Consequently, all feed flow was directed into the still-isolated high pressure feedwater heaters.

The loss of feed flow occurred at about 2225, and was recognized immediately by the RCOs, who noted dropping steam generator [SB:sg] level, dropping feed flow and dropping feed header pressure. Believing that the cause may be a fault in the running main feedwater pump, they started a second condensate pump [SD:p] and a second main feedwater pump, and manually signalled all three Main Feed Regulating Valves [SJ:lcv] to fully open.

Meanwhile the NWE had returned to the high pressure feedwater heater area, and an oncoming Assistant Nuclear Plant Supervisor went to the heater area and relayed information that the unit had lost feed flow. Recognizing that their evolutions had affected the feed flow, the NWE and the TO restored both feedwater heaters to service rapidly, allowing steam generator levels to recover.

The combination of an extra feedwater pump, an extra condensate pump, both trains of feedwater heaters, and full demand on all three Main Feed Regulating valves, was sufficient to cause steam generator level to rise faster than the RCOs could respond. The 4C steam generator reached its trip setpoint of 80% narrow range level, and a turbine trip and automatic reactor trip occurred at 2233.

All safety-related equipment responded as expected, and the event was reported to the NRCOC at 0016 on August 17, 1993, in accordance with 10CFR50.72(b)(2)(ii).

II. CAUSE OF THE EVENT

Immediate Cause

The immediate cause of the turbine trip was a high-high level in the 4C steam generator. The turbine trip automatically generated a reactor trip since power level was greater than 10%.

Root Cause

The root cause of the event was cognitive personnel error by a utility non-licensed operator in that the high pressure feedwater heaters were not properly restored to service. Several factors contributed to the root cause: (1) an adequate pre-evolution briefing was not conducted, in part because of recent successful performances of the evolution, and as a result, the differences between those successful performances and this ultimately unsuccessful performance were not recognized, (2) the Control Room operators were not sufficiently aware of field evolutions, specifically that both high pressure feedwater heaters were being valved in at the same time, (3) the Turbine Operator's knowledge of the operation of the heater inlet valve was deficient, as evidenced by the operator's fully opening these valves when the procedure required the valves to be only cracked open, and (4) repeated unsuccessful attempts to repair the 6A and 6B heater thermal relief valves caused the feedwater system to be in an abnormal configuration, i.e., both high pressure feedwater heaters were bypassed during low power operation.

III. ANALYSIS OF THE EVENT

A reactor trip due to a turbine trip at power is a previously analyzed event. As a result of this analysis, plant procedures are developed to provide operator guidance in responding to these transient conditions and to assure that the plant is stabilized in a safe condition. The unit was stabilized in Mode 3 in accordance with these approved plant procedures. All safety related equipment operated per design. Therefore the health and safety of the public was not affected.

IV. CORRECTIVE ACTIONS

1. The need to perform adequate pre-evolution briefings is being stressed to Control Room supervisors. The briefings will address the need for communication, and the expected plant responses for the actions to be performed. The NPS will determine when briefings are required for each evolution. This action will be completed by September 30, 1993.

2. The operators involved were counselled.

3. Operating procedure 4-OP-081.1 has been revised to provide better direction on sequential and/or simultaneous performance of sections. These improvements are also reflected in the corresponding Unit 3 procedure.

4. Equipment problems which contribute to the need to remove the high pressure feedwater heater from service have been identified, and repairs are being expedited.

5. Licensed and non-licensed operators are being trained on the operation of three-way valves, like those used on the inlets to the high pressure feedwater heaters. This information is being incorporated into the operator requalification training program, and into the system descriptions used to conduct initial training.

V. ADDITIONAL INFORMATION

Previous events involving procedure violations resulting in mispositioned valves were reported in LERs 250/92-005 and 250/93-003. In the first of these, a procedure was available but not used, resulting in two of three main steam isolation valves [SB:isv] being declared inoperable. In the second, the procedure was used, but steps were performed prematurely, resulting in the isolation of both trains of containment spray [BE] while still in Mode 4.

EIIS Codes are shown in the format [EIIS SYSTEM: IEEE component function identifier, second component function identifier (if appropriate)].