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JUL 1 3 2004
L-2004-158
10 CFR § 50.73

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

Re: Turkey Point Unit 4
Docket No. 50-251
Licensee Event Report: 2004-002-00
Automatic Reactor Trip Due to Low Steam Generator Level
and Steam/Feedwater Flow Mismatch

The attached Licensee Event Report 251/2004-002-00 is being submitted pursuant to the requirements of 10 CFR 50.73(a)(2)(iv)(A) to provide notification of the subject event.

If there are any questions, please call Mr. Walter Parker at (305) 246-6632.

Very truly yours,

Terry Jones

Terry O. Jones
Vice President
Turkey Point Nuclear Plant

SM
Attachment

cc: Regional Administrator, USNRC, Region II
Senior Resident Inspector, USNRC, Turkey Point Nuclear Plant

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NRC FORM 366 (7-2001) | U.S. NUCLEAR REGULATORY COMMISSION | APPROVED BY OMB NO. 3150-0104 | EXPIRES 7-31-2004

LICENSEE EVENT REPORT (LER)

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

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
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4. TITLE
Automatic Reactor Trip Due to Low Steam Generator Level and Steam/Feedwater Flow Mismatch

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
5	14	2004	2004	-002	-00	07	13	2004	FACILITY NAME	DOCKET NUMBER

9. OPERATING MODE	1
10. POWER LEVEL	100

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

	20.2201(b)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(B)		50.73(a)(2)(ix)(A)
	20.2201(d)		20.2203(a)(4)		50.73(a)(2)(iii)		50.73(a)(2)(x)
	20.2203(a)(1)		50.36(c)(1)(i)(A)	X	50.73(a)(2)(iv)(A)		73.71(a)(4)
	20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)		73.71(a)(5)
	20.2203(a)(2)(ii)		50.36(c)(2)		50.73(a)(2)(v)(B)		OTHER
	20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)		Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iv)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)		
	20.2203(a)(2)(v)		50.73(a)(2)(i)(B)		50.73(a)(2)(vii)		
	20.2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)		
	20.2203(a)(3)(i)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)		

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
Ronald L. Everett, Licensing Engineer	(305) 246 - 6190

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
-B	-SJ	-FC	-H015	-YES					

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

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

On May 14, 2004, at about 1729 hours, an automatic reactor trip occurred from 100% reactor power after the 4A Steam Generator (S/G) Feedwater Flow Control Valve FCV-4-478 closed unexpectedly. This resulted in an automatic reactor trip actuated by steam flow/feedwater flow mismatch coincident with 4A S/G low-level. The Reactor Operator (RO) identified the condition causing the "S/G A Level Deviation" annunciation and placed FCV-4-478 in manual control. The RO then attempted to manually trip the reactor; however, the reactor automatic trip initiated first. All three turbine driven auxiliary feedwater (AFW) pumps started as expected, supplying the S/Gs.

The spurious closure of FCV-4-478 was caused by the failure of a capacitor in the Hagan feedwater flow controller, which caused the controller output to drive to zero closing the feedwater regulating valve. This caused the steam flow/feedwater flow mismatch coincident with the 4A steam generator low level. An Engineering Evaluation of the failed component determined that the capacitor failure mode was a short circuit and the capacitor was being utilized in a circuit for which it was exposed to an operating voltage above its ratings. The root cause of the event was inadequate implementation of a plant design change to eliminate an originally supplied undersized capacitor design. Immediate corrective actions included replacement of the Hagan Flow Controller FC-4-478 and the associated Hand/Auto Station FC-4-478F, followed by automatic transfer testing and Post Maintenance Testing. Controller power loss manual/auto fail-over tests were also successfully performed on FC-4-488 and FC-4-498 (in the other two control loops). Equipment operated as expected during the event and all parameters were within the envelope of the Updated Final Safety Analysis Report (UFSAR) analysis for Loss of Normal Feedwater. As a result, the event did not compromise the health and safety of the public.

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

Event Description:

On May 14, 2004, Turkey Point Unit 4 was in Mode 1 operation at approximately 100 percent reactor power. At the time of the event, the Steam Dump to Condenser Valves (SDTC) [EIIS: SB, V] had been placed in manual to comply with the pre-requisites for a pre-planned preventative maintenance activity on PI-4-447 [EIIS: SB, PI]. This instrument monitors Turbine [EIIS: TA] first stage pressure and provides an input into SDTC control logic. At 1729 hours, an automatic reactor trip occurred from 100% reactor power, after the 4A Steam Generator (S/G) Feedwater Flow Control Valve FCV-4-478 [EIIS: SJ, FCV] failed to the closed position. Since the SDTC valves were in manual, the valves did not open in response to the reactor trip and were operated manually by the Reactor Operator (RO). The reactor coolant system (RCS) [EIIS: AB] cold and hot leg temperatures increased; however, more significantly, it (SDTC not being in automatic) delayed the reduction in RCS temperatures. This in turn increased the duration that the RCS average temperature remained above 554 degrees F, such that receipt of the feedwater [EIIS: SJ] isolation signal was delayed. This resulted in a level increase in the 4B and 4C S/Gs. The steam dump to atmosphere (SDTA) [EIIS: SB, RV] valves did actuate to limit the increase in S/G pressure, such that the S/G safety valves [EIIS: SJ, RV] did not actuate. The operation of the SDTC in manual and SDTA in automatic, in addition to the heat sink provided by the 4B and 4C S/Gs, significantly reduced the magnitude of this transient. All three turbine driven AFW pumps [EIIS: BA, P] started as expected, supplying auxiliary feedwater to the S/Gs.

The first indication of a disruption in feedwater flow was the actuation of annunciator (ANN) C 6/1 "S/G A Level Deviation" [EIIS: BJ, ANN] (actuates at 5% deviation from programmed level). The RO identified the condition causing the actuation of ANN C 6/1 and placed FCV-4-478 in manual control. However, the rate of inventory reduction within the steam generator did not allow the RO enough time to restore the water level in the 4A S/G. The operator was attempting to initiate a manual reactor trip, when the automatic reactor trip was actuated by a steam flow / feedwater flow mismatch coincident with 4A S/G low-level (less than 10% narrow range level).

Upon the reactor trip actuation, all control rods [EIIS: AA, ROD] inserted as expected. The Turbine Stop valves [EIIS: SB, SHV] closed and the Main Steam Isolation Valves (MSIVs) [EIIS: SB, SHV] were manually isolated in response to the reactor trip signal. A turbine trip resulted from the reactor trip. The auxiliary feedwater system automatically initiated. All equipment operated as expected and all parameters were within the envelope of the UFSAR analysis for Loss of Normal Feedwater.

The event is reportable in accordance with 10 CFR 50.73 (a)(2)(iv)(A), due to the automatic actuation of the reactor protection system (10 CFR 50.73(a)(iv)(B)(1)), MSIV closure (10 CFR 50.73(a)(2)(iv)(b)(2)), and AFW system actuation (10 CFR 50.73(a)(2)(iv)(B)(6)).

Cause of the Event

An Event Response Team was formed to investigate the cause of the rapid closure of FCV-4-478 and to ensure that corrective actions to prevent recurrence were identified and implemented prior to Unit 4 return to service. The team determined that at the time of the reactor trip, I&C Maintenance personnel were performing a calibration of turbine first stage pressure transmitter PT-4-447 [EIIS: SB, PT] . This is significant, because turbine first stage pressure is an input variable to the feedwater flow control valve logic. However, for FCV-4-478, the steam generator level control program input is selectable from either PT-4-446 or 447; PT-4-446 had been appropriately selected at the time of the reactor trip.

Inadvertent interaction with turbine first stage pressure calibration was eliminated as a potential contributing factor after review of real-time data which confirmed that PT-4-446 remained stable through the event and decayed in response to the reactor trip. A review of the applicable maintenance procedure was also performed which confirmed no interaction between the feedwater control circuit and the PT-4-447 work underway was possible (i.e., with S/G level program selected from PT-4-446). Thus, work on PT-4-447 was not a contributing factor to the cause of the event.

Analysis of the feedwater flow signal from real-time data and Emergency Response Data Display System found that 4A feedwater regulating valve FCV-4-478 had closed in about 8 seconds. The fast close solenoids were inspected and found to be energized, indicating that the failure was related to the valve control components, not the fast closure circuit.

Further troubleshooting found that FCV-4-478 would stroke correctly when controlled in Manual, but the Man/Auto Station, FC-4-478F, would not switch to Auto operation. Visual inspection of Controller FC-4-478 found its power supply fuse [EIIS: SJ, FU] blown. Initial bench testing of the failed controller found that an internal component, C21, was shorted (zero ohms across its terminals). By design, a failure of the controller forces the Man/Auto station into manual operation. However, the shorting of C21 would first cause the controller output to fail to zero (driving the valve shut), before blowing the fuse. Hagan controller FC-4-478 and Man/Auto Station FC-4-478F were removed to the Hagan repair bench for testing. Recently refurbished and calibrated modules were installed for FC-4-478 and FC-4-478F.

C21 is a capacitor [EIIS: SJ. CAP] in the controller internal DC power supply. It serves as the filter for the unregulated DC source that supplies the -15 VDC regulator [EIIS: SJ, RG]. Capacitor C21 was removed from the circuit board [EIIS: SJ CBD] and inspected. There was no apparent physical damage. Internal resistance was found to be zero ohms (shorted); whereas, an intact capacitor would exhibit near infinite ohms (capacitors rarely fail by internal short). The manufacturer's markings indicated that the capacitor rating is 100µf, 50 VDC @85°C, 40 VDC @ 125°C, part number 600D107G050DJ4, manufactured by United Chemi-Con.

Subsequent to Unit 4 startup, further testing of the controller found that the DC voltage being applied to C21 is 56 VDC. Circuit analysis shows that the expected voltage is 56 VDC. Testing of various controllers has found this voltage to range from 54VDC to 57VDC. Capacitor C21 is therefore undersized for its application. It should also be noted that this capacitor is rated at 40VDC, but marked for allowed 50 VDC operation under de-rated conditions (lower temperature). Based on these findings, and additional investigations, an Event Timeline and Causal Factor Analyses were performed for this event. An Engineering Evaluation of the failed component determined that the capacitor failure mode was a short circuit and the capacitor was being utilized in a circuit for which it was exposed to an operating voltage above its ratings. This analysis indicates that the Root Cause of the event was inadequate implementation of a plant design change that identified the requirement for replacement of an originally supplied under-rated 50-Volt capacitor in a circuit that generated 56 VDC.

The following provides a summary and findings of the analysis:

1. Westinghouse/Hagan original design installed a 50 VDC rated capacitor in a circuit that had a nominal operating voltage of 56 VDC.

2. <u>Spring 1984</u> - I&C Maintenance personnel identify that Hagan controller capacitor C21, a 50 VDC capacitor, has approximately 56 VDC applied to it. As a result, the useful life of the capacitor would be reduced. The Hagan schematic showed a 75 VDC capacitor for C21, while the parts list showed a 50 VDC capacitor. PC/M 84-073 was issued in response to a request to upgrade C21 capacitor to 100µf 75 VDC. The implementation vehicle for the PC/M was a controlled work order. CPWO 84-073 was issued to allow substitution of C21 with Sprague number 137D107X0075W2 on an as-fail basis. However, the PC/M called for the replacement of C21 in all Hagan controllers with no allowance for replacement on an as-fail basis. Additionally, the technical manual parts list was not changed by this PC/M. No parts were ordered or set up in stores and the CPWO was not time bound.

3. Summer 1993 – With only 3 controllers upgraded on an as-fail basis, a change request notice (CRN) was issued against PC/M 84-073 to close it. The CRN erroneously allowed either 50 VDC or 75 VDC capacitors to be installed in C21. Both the schematic and parts list were changed. The CRN supporting analysis stated that the Sprague 50 VDC was acceptable to operate under 56 VDC continuous voltage, based on the capacitor's surge rating. This analysis misinterpreted the vendor input.

Root Cause Summary

The Root Cause of the event was the failure of the control system for FCV-4-478, due to the premature failure of capacitor C21 in the Hagan controller from a continuous over-voltage condition. C21 capacitors were identified in 1984 as being undersized for the voltage found in the circuit. PC/M 84-073 was generated to replace these capacitors in all Hagan model 124 controllers; however, this PC/M was never fully implemented.

PC/M 84-073 prescribed the replacement of capacitor C21 in all Hagan controllers with a higher rated capacitor and specified a 100µf, 75 VDC capacitor. However, the PC/M did not change the Technical Manual parts list, even though the initiating request specifically stated the parts list was in error. The PC/M implementing work order (work order CPWO 84-073) did not require the replacement of under-rated capacitor C21, but instead allowed the substitution of upgraded C21 to all Hagan controller power supply circuit boards on an as fail basis. Consequently, the PC/M had no required completion schedule. Had PC/M 84-073 been implemented completely and in a timely manner, this event would not have been expected to occur due to the premature failure of under-rated capacitor C21.

Analysis of the Event

This condition is reportable under 10 CFR 50.73 (a)(2)(iv)(A) and (B) as an event that resulted in manual or automatic actuation of the reactor protection system, initiation of the auxiliary feedwater system (AFW), and manual isolation of the MSIVs. The May 14, 2004 event, a reactor trip due to a low steam generator level in S/G 4A, in conjunction with a steam/feedwater flow mismatch, is an event bounded by a previously analyzed event in the UFSAR, Loss of Normal Feedwater. Closure of FCV-4-478 initiated a rapid decrease in steam generator 4A inventory The mass of water in the 4B and 4C S/G increased, which increased narrow range level well above 60%. The Reactor Operator manually controlled the steam dump to condenser valves, providing adequate heat removal and preventing actuation of a S/G safety relief valve and automatic actuation of Safety Injection. The Reactor Operator attempted to manually trip the reactor; however, the reactor automatically tripped due to the steam flow/feedwater flow mismatch (feedwater flow 0.665 E6 lbm/hr less than steam flow), coincident with 4A S/G low-level (10% narrow range level).

Analysis of Safety Significance

Loss of feedwater is an analyzed event that is provided for by the plant design and procedures. Plant instrumentation designed to monitor for this event actuated and performed as required, placing the reactor in a safe condition. Although such trips are a challenge to the automatic safety systems, these challenges are well within the design parameters of these systems and the overall expected plant response.

The UFSAR analysis assumes a loss of normal feedwater to all steam generators due to the loss of the feedwater pumps or valve malfunction. In the May 14, 2004 event, only the "A" steam generator reached a low level condition. It should be noted that this event never compromised the ability of FCV-4-478 to perform its Safety or Quality Related functions. Upon failure, the valve attained its Safety Related (closed) position.

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Because normal feedwater flow and AFW flow were available for removing stored and residual heat, the event did not compromise the health and safety of the public. A scoping risk assessment yielded a conditional core damage probability of less than 1.0E-6. The risk associated with the event was not significant.

Corrective Actions/Failed Components Identified

1. An Item Equivalency Evaluation Design Change has been issued, changing the rating value of capacitor C21 to 100µf, 100 VDC. This also changes the vendor technical manual to show C21 as 100µf, 100 VDC, both schematically and on the parts list.

2. To address the extent of condition, capacitor C21 in all Hagan controllers (including FCV-4-478) in various systems are being replaced with capacitors rated at 100µf, 100 VDC. An Operability Evaluation of the remaining applications of the under-rated capacitors has been performed. The under-rated capacitors are not installed in any application required to perform a safety function. The higher rated (100 VDC) capacitors were selected for their superior design life characteristics (rated lifetime of 1,000,000 hours). Replacement of Unit 3 Feedwater controller capacitors with 100 VDC rating was completed during a recent downpower.

3. To address generic implications, the closed PC/Ms of that timeframe which correct an equipment deficiency and those that are not yet fully implemented will be reviewed to determine impact on plant safety and reliability.

Similar Events

A review of other reactor trip events in recent years was performed. At Turkey Point, two somewhat similar events were identified. On January 27, 2003, a manual reactor trip occurred on Unit 3 due to low steam generator level on the 3C steam generator. The low steam generator was caused by momentary loss of instrument air compressors, leading to the main feedwater control valves drifting closed. In addition, on July 30, 1997, an automatic reactor trip of Unit 3 occurred, due to closure of the B Main Steam Isolation Valve. The reactor tripped automatically on low-low level in the 3B S/G due to a relay failure in the MSIV actuator circuitry. The root cause of these events was not the same as for this event.