CATEGORY 1

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9904070235 DOC.DATE: 99/03/30 NOTARIZED: NO DOCKET #
FACIL:50-270 Oconee Nuclear Station, Unit 2, Duke Power Co. 05000270

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RECIP.NAME	RECIPIENT AFFILIATION

SUBJECT: LER 99-001-00:on 990228,reactor trip occurred.Caused by equipment faiure.Failed fuse replaced & revised maint processes as required.With 990330 ltr.

DISTRIBUTION CODE: IE22T COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 10
TITLE: 50.73/50.9 Licensee Event Report (LER), Incident Rpt, etc.

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March 30, 1999

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

Subject: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287
Licensee Event Report 270/1999-01, Revision 0
Problem Investigation Process No.: 2-099-0771

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a) (1) and (d), attached is Licensee Event Report 270/1999-01, concerning a reactor trip on high Reactor Coolant System pressure due to an equipment failure.

This report is being submitted in accordance with 10 CFR 50.73 (a) (2) (iv). This event is considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

Je22

W. R. McCollum, Jr.

Attachment

770022

cc: Mr. Luis A. Reyes
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Mr. M. A. Scott
NRC Resident Inspector
Oconee Nuclear Station

NRC FORM 366 (4-95)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Oconee Nuclear Station, Unit Two	05000 270	1 Of 8

TITLE (4)
Equipment Failure Results In A Reactor Trip

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(S)
										05000
02	28	1999	1999	- 01	- 00	03	30	1999		05000

OPERATING MODE (9): N
POWER LEVEL (10): 098

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)	
J.E. Burchfield, Regulatory Compliance Manager	(864)	885-3292

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
X	TG	FU	Gould Inc	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 fifteen single-space typewritten lines)* (16)

On February 28, 1999, at 1544 hours, Unit 2 was at 100% power. A small Reactor transient occurred when Electro Hydraulic Control (EHC) System normal electrical power was lost and a swap over to backup power occurred. The loss of EHC system normal electrical power was due to the failure of a fuse. Reactor power stabilized at 98% following the transfer of EHC electrical power. At approximately 2040 hours, during troubleshooting and root cause investigation, there was a spurious reapplication of power through the failed fuse. The reaction of the EHC system caused Unit 2 Turbine control valves to close and the Reactor tripped on high Reactor Coolant System pressure. All primary and secondary parameters responded as expected and operations personnel appropriately stabilized the unit at hot shutdown conditions. The root cause of the reactor trip is equipment failure. The fuse failed due to an internal open circuit fault at one end of the fuse rather than an overcurrent condition. A detailed laboratory analysis of the failed fuse is being performed. Corrective action includes replacing the failed fuse, having the failed fuse analyzed, and revising maintenance processes as required.

This event is considered of no significance with respect to the health and safety of the public.

EVALUATION:

Background

The Electro-Hydraulic Control (EHC) [EIIS:TG] System is a General Electric Mark 1 solid state analog control system of late 1960's technology. The system is divided into three major functional control areas: speed control, load control, and flow control. Monitoring circuits are provided to annunciate conditions of the system. Protection circuitry is imbedded in each control section to protect the turbine generator from mechanical and electronic failures in the system. The system is supplied power from various sources to support the control and protection functions. Power sources include:

125 VDC for supply of electronic trip inputs.

60 Hz 120 VAC from breaker 2KX1 for supply to the following circuits:

- EHC cabinet cooling fans
- Fast Acting Solenoid Valves
- Valve Test Solenoid Valves
- Monitor Panel meter lamps (alarm circuit source)
- Load Reference Motor Circuits
- +30 VDC normal power supply
- -22 VDC normal power supply
- +24 VDC normal logic power

60 Hz 120 VAC from Integrated Control System [EIIS:JA] Transfer Bus power to the Load Reference Motor (LRM) Circuits. This allows LRM movement in automatic and manual with the loss of 2KX1.

420 Hz 120 VAC from the Permanent Magnet Generator (PMG) for supply to the following circuits:

- +30 VDC backup power supply
- -22 VDC backup power supply
- +24 VDC backup logic power

Two DC power busses, +30 VDC and -22 VDC, supply power for the EHC electronics. These busses are referenced to the EHC cabinet ground.

A third, +24 VDC, logic bus provides power for all EHC internal relay logic, including Monitor Circuits, Alarm and Trip logic, Speed Control logic, Valve Test logic, Chest/Shell Warming logic, and Stage Pressure Feedback logic. This bus is floating with respect to ground, and is monitored for grounding by a detection circuit.

These three DC power busses are normally supplied by the 60 Hz AC power source from breaker 2KX1 through the three 60 Hz normal power supplies. If one or more of the normal supplies fails, the respective bus may be powered by the 420 Hz 120 VAC source from the PMG through the backup power supply. The normal and backup power supplies are auctioneered through high current capacity diodes. The DC supplies are deliberately adjusted so that they do not carry current when the normal supplies are available.

Description of Event

On February 28, 1999, at 1544 hours, Unit 2 was operating at 100% power. The Unit 2 Control Room received various alarms associated with the Integrated Control System (ICS) and Electro Hydraulic Control (EHC) System. Operators took appropriate actions for the transient condition and Reactor power was stabilized at 98%. Operators determined that the EHC System had switched to the backup power supply.

The Shift Technical Advisor requested assistance from engineering and maintenance support for troubleshooting and repair of the EHC System. The Operations Shift Manager notified the appropriate management and a Failure Investigation (FIP) team was organized.

At approximately 1900 hours, after taking shift turnover, the Unit 2 operations shift conducted a Pre-Job Briefing. They discussed the problems experienced with the Unit 2 EHC system, the possible consequences, and the actions to take while the FIP team was investigating the transient.

The goal of the FIP investigation team was to:

1) Determine the affected signals in the EHC
2) Determine that the ICS and plant responses were normal
3) Develop methods for restoring the system to normal without causing a unit trip.
4) Determine the root cause of the transient.

From an initial investigation, it was known that a loss of 60 Hz AC power had occurred to the EHC, and the 120 Volt AC breaker 2KX1 had not tripped. Since the breaker contains an internal overload fuse, this was immediately suspected as the faulty device.

At approximately 2000 hours, data gathering began by recording control room EHC indications and Monitor Panel indications. There were two Instrument and Control technicians involved in taking measurements. Technician A was performing the work while Technician B observed. A technical expert for ICS and EHC was observing the overall process. At approximately 2039 hours, the three EHC DC power supply voltages had been measured and recorded. Immediately after the measurement of the +30 VDC bus was completed, the technical expert observed the Monitor Panel lights "flicker". Immediately following that, the technical expert observed the "Backup Overspeed Acceleration Amplifier Out of Negative Saturation" lamp illuminated, and relays were heard actuating. The Backup Overspeed Acceleration Amplifier initiation caused the Main Turbine [EIIS:TA] Control Valves and Intercept Valves to close. The Reactor tripped from 98% power at 2040 hours, approximately ten seconds after the Monitor Panel lights were observed.

In the Unit 2 Control Room, the Main Steam [EIIS:SB] Press hi/low alarm actuated. The Control Room operations team observed the Post Accident Monitoring (PAM) [EIIS:IP] indications for the 2A and 2B SGs and noted pressure increasing rapidly. The Reactor Protection System (RPS) tripped the reactor on high Reactor Coolant System (RCS) pressure. All full length control rods fully inserted into the core shutting down the reactor.

Operators confirmed that the Reactor and Turbine had tripped and monitored the unit for proper operation. They implemented the Emergency Operating Procedures (EOP) and as normally occurs after a Reactor trip, a second High Pressure Injection (HPI) [EIIS:BG] pump started for additional makeup flow to the RCS.

As expected, when 2HP-26 (2A High Pressure Injection Valve) was opened, HPI pump 2B automatically started. It was secured and placed into automatic after approximately 1 minute 30 seconds. Reactor Coolant Pump Seal flow had not recovered and HPI pump 2B automatically restarted on low seal flow (< 30 gpm). The pump ran for 2 minutes and 30 seconds before being secured and placed in automatic. The time for recovering makeup flow is consistent with previous trips.

All primary and secondary parameters responded as expected. Pressurizer level decreased from a pre-trip level of 227 inches to a post trip minimum of 62 inches before stabilizing at 151 inches. Immediately prior to the trip Reactor Coolant System (RCS) [EIIS:AB] pressure increased from approximately 2155 psig to 2332 psig. Post trip RCS decreased to 1783 psig before stabilizing at 2172 psig. Pre-trip RCS average temperature peaked at 580 F. Following the trip, RCS average temperature decreased and stabilized at approximately 553 F.

Immediately prior to the trip, Steam Generator (SG) 2A and 2B levels decreased from 166.5 inches and 162.3 inches respectively to approximately 122 inches. Both SGs maintained a normal minimum level of approximately 25 inches following the trip. Prior to the trip, SG pressures were at a steady state of 927 psig for the 2A and 932 psig for the 2B SG before increasing to 1093 and 1097 respectively. The post trip peak pressure was 1137 psig for the 2A SG and 1119 psig for the 2B SG. The 2A and 2B SGs subsequently stabilized at 960 psig and 961 psig respectively. The letdown storage tank (LDST) level dropped to 55 inches and valve 2HP-24 (suction from Borated Water Storage Tank) was opened for about 2 minutes.

There were no Engineered Safeguard actuations and Main Feedwater was maintained following this Reactor trip.

Following the Reactor trip, the FIP team metered the overload fuse for breaker 2KX1 and confirmed that it had failed. The fuse was radiographed and found open on one end. This fuse had not failed due to overcurrent, since the internal bus bar was intact. On March 1, 1999, other troubleshooting efforts did not identify any faults in the EHC circuitry.

The FIP team verified that there were no inappropriate actions associated with the troubleshooting that could have resulted in a Reactor trip.

The FIP team analysis of the pre-trip transient was reviewed and various circuits were verified for possible causes of the 2KX1 breaker fuse failure. The FIP team concluded that none of the EHC circuits had experienced other component failures. The visual observation of the Monitor Panel Lights by the technical expert and the physical evidence provided by the radiograph indicated that the fuse had momentarily re-energized.

On March 2, 1999, at 0230 hours, all the post trip items had been satisfied and the approval for re-start was granted. At 0654 hours, the reactor was critical and at 1008 hours reactor power was 15%.

At 1100 hours, operators identified that a Main Steam bypass valve, 2MS-19, was closed and had not operated. The valve was isolated and repaired. A Problem Investigation Process Report was initiated for this item.

On March 3, 1999, at 0632 hours, Unit 2 was at 100% power.

Conclusion

The root cause of this event is an equipment failure. The initiating transient was caused by failure of the 60 Amp overload protection fuse in breaker 2KX1. The power transfer from the normal Electro Hydraulic Control (EHC) system power to the back-up source caused the transient. The unit tripped when the failed fuse spuriously re-energized after backup electrical power was in service. The re-energizing of the fuse resulted in a condition affecting the speed control circuitry of the EHC system. The EHC speed control upset caused the Main Steam Stop Valves to close and the Reactor tripped due to high Reactor Coolant System pressure. The only failed equipment causing this event was the 60 Amp Class G (current limiting) protection fuse in breaker 2KX1. The fuse did not fail due to overcurrent but failed due to an internal open circuit fault at one of the end caps. A detailed analysis of the failed fuse is being performed.

A review of LERs and operating experience within the past two years indicates that there have not been any reactor trips associated with equipment failures of this type. There have been industry events in the past where fuses intermittently failed resulting in transients and or reactor trips.

The failed fuse is a Gould AMPTRAP Class G fuse, type AT1 CC60, rated 60 Amps. This model fuse is now obsolete. The failed fuse had been in a spare breaker and was placed in the 2KX1 location in 1995, when the original breaker was replaced. The replacement type identified by the manufacturer is a Gould AG60.

This event did not result in personnel injuries, radiation overexposures, or releases of radioactive materials.

CORRECTIVE ACTION:

Immediate:

1. Operators took appropriate actions to stabilize the unit at hot shutdown conditions.

Subsequent:

1. The overload protection fuse in breaker 2KX1 was replaced.

2. The Electro Hydraulic Control (EHC) system circuitry was checked for problems.

Planned:

1. Perform a detailed failure analysis on the failed fuse.

2. After the detailed failure analysis is completed, evaluate fuse PM program to determine appropriate changes to prevent recurrence.

Planned corrective actions 1 and 2 are considered to be NRC Commitment Items. These are the only NRC Commitment items contained in this LER.

SAFETY ANALYSIS:

Unit 2 Reactor tripped due to a failed fuse that spuriously re-energized in the Electro Hydraulic Control (EHC) power supply system. With the backup source supplying power, the re-energizing of the normal supply caused an upset in the EHC speed control circuitry. This resulted in the Main Steam Control Valves closing and the Reactor Coolant System pressure increasing to the trip setpoint.

The Reactor Protective System operated as designed and tripped the reactor. The unit post trip response was acceptable. No Engineered Safeguards System or Emergency Feedwater actuations were either required or received after the trip.

The health and safety of the public was not compromised by this event.