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SUBJECT: LER 89-002-00:on 890306,reactor trip due to turbine/
generator trip.W/890405 ltr.

W/8 ltr.

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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)
Oconee Nuclear Station, Unit 3	0 5 0 0 0 2 8 7	1 OF 0 7

TITLE (4)
Reactor trip due to turbine/generator trip

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAMES	DOCKET NUMBER(S)
0 3	0 6	8 9	8 9	— 0 0 2 —	0 0	0 4	0 5	8 9		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 1 0 0

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	
H. R. Lowery, Oconee Safety Review Group	8 0 3	8 8 5 - 3 0 3 4

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

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 March 6, 1989 at 0657 hours while operating at 100% Reactor Power, the Unit 3 Main Turbine (MT) tripped, resulting in an anticipatory Reactor trip. The MT trip was initiated by a generator lockout which was due to a loss of generator excitation. The reason for the loss of excitation could not be determined. Additionally, three pipe supports were damaged following the trip as a result of a water hammer in the Main Steam Turbine Bypass line. The immediate corrective action was to stabilize the unit at hot shutdown. Subsequent corrective actions included determining the cause of the trip and implementing a plan for the repair of the three pipe supports. The actual root cause of the trip could not be determined, therefore this event is classified as unknown.

IE22

NRC Form 366 (9-83)

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
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TEXT (If more space is required, use additional NRC Form 366A's) (17)

BACKGROUND

The Oconee Nuclear Station switchyard [EIIS:FK] is divided into a 230 kV portion and a 525 kV portion. The 230 kV and 525 kV switchyards are connected to each other through the autobank transformer [EIIS:XFMR], thereby allowing the units to supply power to either of the switchyards. Units 1 and 2 normally supply power to the 230 kV switchyard and Unit 3 normally supplies power to the 525 kV switchyard. The 230 kV switchyard supplies power to the Jocassee Pumped Storage Hydro Station whenever it is necessary to operate the Hydro Station as a pump to pump water back to the lake.

Dispatchers notify Control Room Operators (CRO) when the 230 kV and the 525 kV switchyards need to be separated based on system power requirements. Dispatchers also specify to the CROs the required reactive load prior to separating switchyards, the positioning of the automatic voltage regulator either in automatic or manual mode, and the acceptable voltage differential between the plant-side bus and the grid-side bus.

The automatic voltage regulator monitors the output voltage of the main generator and maintains the voltage at the level specified by the Control Room Operator.

Switchyard line reactors are applied to long distance 525 kV transmission lines to partially compensate for the large charging currents generated by the transmission lines. The reactors help control the voltage conditions of both the line and the line terminal. In some instances the reactors also help minimize overvoltages that can occur during switching operations.

EVENT DESCRIPTION

On the morning of March 6, 1989, Unit 3 was operating at 100 % power. Power Circuit Breakers (PCB) had previously been opened to separate the grid-side bus and the plant-side bus of the 525 kV switchyard while the Jocassee Hydro Station was pumping back to the lake for later peak reserve. Therefore, Unit 3 was supplying power to the 230 kV switchyard via the autobank transformer. The Dispatcher requested the Units 1 and 2 Control Room Operators (CRO) to reduce reactive load, MVARs. The Dispatcher also requested the Unit 3 CRO to increase reactive load to 180 MVARs, in preparation for closing Unit 3 into the grid-side of the 525 kV switchyard. The Unit 3 CRO noticed that reactive load on Unit 3 increased to 180 MVARs when Units 1 and 2 reactive load was decreased. The Unit 3 CRO informed the Dispatcher of the Unit 3 reactive load increase. The Dispatcher told the Unit 3 CRO that no additional changes were necessary and to expect a mild swing in reactive load upon tieing the 525 kV plant-side and grid-side buses together. At 0657 hours, PCB-55 was closed, thereby tieing the 525 kV plant-side and grid-side buses together. Approximately five seconds after PCB-55 was closed, a loss of excitation on the Unit 3 generator [EIIS:TB] occurred, followed by a generator lockout and a Main Turbine [EIIS:TA] and Reactor [EIIS:RCT] trip. Following the Turbine and Reactor trip, the Unit was stabilized at hot shutdown. The Main Feedwater [EIIS:SJ] Pumps did not trip, and consequently no actuation of the Emergency Feedwater System [EIIS:BA] occurred. In general the plant post-trip response was as expected. The average Reactor Coolant System (RCS) [EIIS:AB] temperature stabilized at about 550 degrees Fahrenheit. RCS pressure ranged from 2150 psig prior to the trip, to a minimum of about 1830 psig, and to a maximum of 2200 psig. After initial fluctuations, the RCS pressure averaged 2155 psig. Pressurizer level decreased from the initial pre-trip value of 220 inches to a minimum of 70 inches immediately after the trip and then was maintained at 160 inches by starting the "3B" High Pressure Injection [EIIS:BQ] Pump. Pressure in the "3B" Steam Generator (SG) [EIIS:SG] was initially controlled at approximately 1025 psig before controlling steadily at 970 psig. Pressure in the "3A" SG was initially controlled between 980 and 1005 psig until pressure dropped, concurrent with the drop in the "3B" SG, to 925 psig and increased to 950 psig. The control of the "3A" SG pressure is attributed to the manual cycling of Main Steam Bypass valves by the CRO to lower header pressure to reseat a Main Steam Relief valve, 3 MS-8. The level in both SGs dropped to 25 inches and controlled smoothly.

Additionally, following the trip, a water hammer in the Main Steam Turbine Bypass [EIIS:SQ] line to the "A" Condenser [EIIS:SQ] caused a twelve to eighteen inch deflection in the line. Three pipe supports were damaged and insulation on the pipe was damaged as a result of the water hammer. Subsequent investigation indicated that the most likely point of water accumulation was at an orifice in a drain line. A plan for the repair of these pipe supports concurrent with plant restart was implemented.

Immediately following the Turbine and Reactor trip, the Main Feedwater Block Valves (3FDW-31 and 40) came out of automatic mode and remained open. These valves were in automatic mode prior to the trip and they should have remained in automatic mode. The valves also should have closed after the Reactor trip. Because of training and awareness of this problem with the valves, the CRO expected the valves to be out of automatic mode and out of their required position. The CRO properly closed the valves. While the performance of the Main Feedwater Block Valves was not desirable, it was expected. The above-mentioned problems were previously identified and corrective actions assigned in Licensee Event Report 269/89-01.

Initially, lightning strikes on the transmission lines, concurrent with the closure of PCB-55, were suspected as having caused the trip. The Reactor was returned to critical at 1215 hours and two of the damaged piping supports were repaired prior to placing the Main Turbine on line at 2253 hours that night.

There was no apparent RCS leakage induced by this trip and no actuation of either Engineered Safeguards Systems [EIIS:JE] or Pressurizer relief valves occurred during this incident.

CONCLUSIONS

The Main Turbine and Reactor tripped five seconds after Power Circuit Breaker (PCB) 55 was closed. When PCB-55 was closed, a loss of excitation on the Unit 3 generator occurred, followed by a generator lockout, which caused the Main Turbine and Reactor trip. A review of switchyard data indicated that the lightning strikes did not occur at the same time as the closure of PCB-55 and did not cause the trip. The root cause of this event could not be determined, therefore the root cause of this event is classified as unknown. It is noted that a task force of Transmission, Design Engineering and Nuclear Production representatives has been formed to determine the cause of the trip. The task force has outlined additional testing and investigation into the cause of the unit trip to be conducted, but much of this testing must be performed with the unit at both low power levels and off line. Although various adjustments to the Unit 3 automatic voltage regulator have been verified as being appropriate, testing of the automatic voltage regulators during each units next refueling outage will be conducted. An evaluation of the voltage differential between the plant-side and grid-side buses, which was 24 kV at the time of the trip, revealed that a lower differential would have been preferable, and that a 24 kV differential, combined with not utilizing the switchyard line reactors, could have contributed to the trip. Also, it would have been preferable to have placed the switchyard line reactors into service prior to closure of PCB-55, failure to do so could have contributed to the trip. It is concluded that the voltage differential between the plant and grid-side buses should be kept to a minimum prior to closing switchyard tie-breakers (PCBs) and that switchyard line reactors should be placed in service prior to closing switchyard tie breakers (PCBs).

It is concluded that the damage to three Main Steam Turbine Bypass (MSTB) piping supports was caused by a deflection of the piping due to a water hammer in the line. It is suspected that the MSTB drain line was obstructed, thereby allowing water to collect in a portion of the MSTB piping, which led to the water hammer. Repairs to two of the piping supports began shortly after the trip and were completed at 2140 hours on March 6. Repairs to the third pipe support were completed on March 9. It is also suspected that the water hammer cleared the drain line of the obstruction. Thus, presence of an obstruction in the line prior to the trip could not be verified. The drain line was verified as being clear after the trip, so that additional condensation in the MSTB piping is expected to drain to the Condenser, however, the potential for the drain line to become obstructed still exists. It is noted that Design Engineering has generated a Station Problem Report, which suggests a means of resolving the drainage problem. It is concluded that in order to reduce the probability of a similar problem in the future, a long-term resolution should be pursued.

After the Unit 3 trip, the Main Feedwater (FDW) Block Valves, 3FDW-31 and 3FDW-40, came out of automatic mode and remained open. Because of training and awareness of this problem with the valves, the Control Room Operator (CRO) expected the valves to be out of automatic mode and out of their required position. The CRO properly closed the valves. Investigation from an earlier incident (reference Licensee Event Report 269/89-01) revealed that a brief power loss to the momentary contact switch for the Main Feedwater Block Valves caused the loss of a seal-in circuit relay. The loss of this relay allowed the Main Feedwater Block Valves control to come out of automatic mode. The loss of this relay is an expected response until a modification to correct the problem is implemented.

The Main Steam Relief Valve which had to be reseated by lowering Turbine Header pressure was identified as 3MS-8. Additional investigation was performed by Mechanical Maintenance regarding the valve's performance and it was determined that additional valve maintenance is not required.

Several other unit trips occurred in the past year. Two of the trips, LER 287/88-06 and LER 270/89-03, were due to unknown root causes, however, the events were unrelated to this incident, therefore this event is considered non-recurring. None of the corrective actions associated with these trips could have prevented this trip. LER 287/88-06 detailed two Unit 3 trips that occurred on November 14, 1988. One trip was due to an unknown cause and the other was due to a ground fault in the Steam Generator (SG) level trip signal monitor. Corrective actions for the second trip addressed testing the SG high level trip signal monitors that were in use and those that were in stock prior to use. LER 270/89-03 detailed a Unit 2 trip that occurred on February 5, 1989, during testing of the Master Trip Solenoid test switch. The cause of the trip could not be determined. No personnel injuries, radiation exposures, or releases of radioactive material resulted from this unit trip. Consequently, the health and safety of the public were not compromised as a result of this event.

CORRECTIVE ACTIONS

Immediate

The immediate corrective action was to stabilize the unit at hot shutdown.

Subsequent

1. The three piping supports which were damaged during the transient were repaired.

2. Operations generated Information Notices for each unit to identify specific switchyard-related details that Control Room Operators should discuss with Dispatchers prior to tieing the 525 kV plant and grid-side buses together.

3. Station Problem Report, I.D.# 890311, was initiated by Design Engineering to resolve problems with the failure of the Main Steam Turbine Bypass piping to drain properly.

4. Mechanical Maintenance determined that additional maintenance on Main Steam Relief Valve, 3MS-8 is not necessary.

Planned

1. Nuclear Station Modification 2804, as identified in Licensee Event Report 269/89-01, ensures that the Main Feedwater Block Valves do not come out of automatic mode upon the transfer of power from the Main Transformer to the Startup Transformer, will be implemented on Units 2 and 3. The modification was implemented on Unit 1 during its 1989 refueling outage.

2. Additional testing will be conducted by Nuclear Production, Design Engineering and Transmission Department to determine the cause of the trip and to correct anomalies identified as a result of the investigation.

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

SAFETY ANALYSIS

Following the Turbine and Reactor trip, the unit was stabilized at hot shutdown. Emergency Feedwater [EIIS:BA] was not actuated and the Integrated Control System [EIIS:JA] responded properly. The Operations Control Room personnel safely controlled the unit following the trip. No actuation of Engineered Safeguards [EIIS:JE] systems or Pressurizer relief valves occurred, and no Reactor Coolant System leakage was induced as a result of this trip. Following the trip, both Main Feedwater Block Valves transferred from automatic to manual control and remained open due to the seal-in relay losing power when the unit's electrical loads transferred from the Main Transformer to the startup transformer. Because this valve response was expected, the Control Room Operator quickly and properly closed the valves. This response did not adversely affect the plant post-trip response. Emergency systems were available to assist Operations personnel in controlling the plant, however, the systems were not required to be used and were not activated. All plant parameters were within expected values and no safety limits were exceeded. All systems responded correctly and no abnormal plant responses occurred as a result of the trip. The health and safety of the public were not jeopardized as a result of this event.

Duke Power Company
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Hal B. Tucker
Vice President
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DUKE POWER

April 5, 1989

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

Subject: **Oconee Nuclear Station**
Docket Nos. 50-269, -270, -287
LER 287/89-02

Gentlemen:

Pursuant to 10CFR 50.73 Sections (a) (1) and (d), attached is Licensee Event Report (LER) 287/89-02 concerning a Unit 3 reactor trip on March 6, 1989.

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,

Hal B. Tucker/RWR

Hal B. Tucker

PJN

Attachment

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