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FACIL:50-269 Oconee Nuclear Station, Unit 1, Duke Power Co. 05000269

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RECIP.NAME	RECIPIENT AFFILIATION

SUBJECT: LER 92-004-01:on 920508,reactor tripped from 14% full power on RPS anticipatory trip signal.Caused by mgt deficiency & low discharge pressure on main feedwater pump. Task-specific procedure developed.W/920715 ltr.

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DUKE POWER

July 15, 1992

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

Subject: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287
LER 269/92-04, Revision 1

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a)(1) and (d), attached is Revision 1 to Licensee Event Report (LER) 269/92-04, concerning a unit trip. This revision covers additional information discovered after the initial report was submitted.

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,

J W Hampton

J. W. Hampton
Vice President

/ftr

Attachment

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NRC FORM 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

ESTIMATED BURDEN PER RESPONSE TO COMPLY WTH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Oconee Nuclear Station, Unit 1	0 5 0 0 0 2 6 9	1 OF 0 7

TITLE (4) Reactor Trip Results From Low Main Feedwater Pump Dischrge Pressure Due to Management Deficiency

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 5	0 8	9 2	9 2	— 0 0 4 —	0 1	0 7	1 5	9 2		0 5 0 0 0
										0 5 0 0 0

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

OPERATING MODE (9): N

POWER LEVEL (10): 0 1 4

[ ] 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	
S. G. Benesole, Safety Review Manager	8 0 3	8 8 5 - 3 5 1 8

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
F	S J	P S P	X 0 0 0	Yes					
F	B A	X C V	V 0 3 0	Yes					

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)

ABSTRACT

On May 8, 1992 at 0342:23 hours, Unit 1 reactor tripped from 14 percent full power on a Reactor Protective System anticipatory trip signal due to low discharge pressure on the Main Feedwater Pump (MFDWP). The low discharge pressure occurred when operators were attempting to decrease a high hotwell level, which diverted flow from the suction of the MFDWP. After the trip, the Emergency Feedwater (EFDW) System actuated due to the low MFDWP discharge pressure. Once the MFDWP was verified to be operating, the EFDW Pumps were secured. The two root causes identified for this event were management deficiency, less than adequate training given and lack of a task specific procedure. Corrective actions include Operator training to inform Operators of the hotwell level oscillations, correct methods of reducing hotwell level, and development of a task specific procedure.

BACKGROUND

The main condenser is designed to condense turbine exhaust steam for reuse in the steam cycle. The main condenser also serves as a collecting point for various steam cycle vents and drains to conserve condensate which is stored in the hotwell. The hotwell has an emergency high level alarm (72 inches), High level alarm (69 inches), Low level alarm (57 inches), and Emergency low level alarm (10 inches). The condenser also serves as a heat sink for the Turbine Bypass Valves (TBVs) [EIIS:SO] which are capable of passing 25 percent of rated main steam flow.

The TBVs are designed to dump Main Steam [EIIS:SB] load directly to the main condenser during startup and shutdown operation, thereby creating an artificial load on the reactor.

The Condensate Steam Air Ejectors (CSAEs) remove air and noncondensable gasses from the main condenser to maintain proper Condenser vacuum.

The Condensate System [EIIS:SD] originates at the condenser hotwell. The Hotwell Pumps and Condensate Booster Pumps increase system pressure to that required for the Main Feedwater Pump (MFDWP) net positive suction head. The Upper Surge Tank provides a surge volume for the Condensate System. (See Attachment 1)

The MFDWP increases the Feedwater System [EIIS:SJ] pressure to provide adequate feeding of the Steam Generators.

The Reactor Protective System (RPS) [EIIS:JC] consists of four identical protective channels, each terminating in a trip relay within a reactor trip module. The coincidence logic in all reactor trip modules actuate when any two of the four protective channels trip. The RPS monitors Reactor Coolant System (RCS) [EIIS:AB] parameters related to safe operation and trips the reactor to protect against fuel rod cladding damage. It also assists in protecting against exceeding RCS pressure limits by providing an anticipatory trip on low MFDWP discharge pressure.

The Emergency Feedwater [EIIS:BA] will actuate on loss of both Main Feedwater Pumps (MFDWPs). The actual initiating conditions are low discharge pressure (<800 psig) on both MFDWPs or low of hydraulic oil pressure on both MFDWPs. MFDWPs will trip on high discharge pressure.

The Auxiliary Steam System [EIIS:SA] consists of a header which is supplied by Main Steam and each unit's header is normally cross-connected to the other units. When a unit is starting up the Auxiliary Steam header is normally supplied by another units main steam to supply various steam loads.

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

EVENT DESCRIPTION

On May 7, 1992 at 2330 hours, Unit 1 was at Hot Shutdown following a Unit trip due to a Generator Lockout. The Hotwell Level was observed at 67 inches by the Control Room Operator (CRO). The CRO identified this to the Control Room Senior Reactor Operator (CR SRO).

At 0145 hours on May 8, 1992, Unit 1 was returned to criticality. Reactor Power was increasing and the Condensate and Feedwater Systems were aligned utilizing 1B Main Feedwater Pump (MFDWP) to maintain minimum Steam Generator level. All steam being produced was bypassing the Main Turbine [EIIS:TA] via the Turbine Bypass Valves to the condenser. Unit 1's Auxiliary Steam header, being supplied by another unit, was supplying steam to the Condensate Steam Air Ejectors (CSAEs), 'E' Heaters, MFDWP and various steam seals and exhausting into Unit 1's condenser.

At 0325 hours, the hotwell high level alarm (setpoint 72 inches) was received. The hotwell level was fluctuating between 73 and 78 inches, and trending upward. The CRO reviewed the Alarm Response Manual to determine the appropriate actions to be taken. The CR SRO and Shift Manager were concerned with the possibility of flooding the CSAEs suction lines due to high hotwell level. At approximately 0330 hours, the Shift Supervisor was notified of the high hotwell level by the CR SRO. The CR SRO, Shift Manager, and Shift Supervisor discussed the need and the method to reduce the hotwell level.

At 0340 hours, the CR SRO, Shift Manager, and Shift Supervisor decided on a method to reduce hotwell level, which only involved opening two valves in the Condensate System. This included opening 1C-124 (Condensate Recirc to Upper Surge Tank) and then opening 1C-128 (Condensate Recirc Control) to divert condensate to the Upper Surge Tank and then to the Condensate Storage Tank. After completing this lineup it would be transferred to another unit. The CR SRO stated that he had performed this method on other occasions. The CR SRO stated that actions in the Alarm Response Manual would not solve the high level, because the Alarm Response procedure (1SA6/C-12) did not address the unit's specific operating condition.

At approximately 0342 hours, the CR SRO told CRO to verify that 1C-128 (Condensate Recirc Control) was in the closed position. After verifying 1C-128 closed, the CR SRO instructed the RO to open 1C-124. Upon opening 1C-124, 1B MFWDP discharge pressure decreased to approximately 800 psig, causing Reactor Protective System Channels A, B, C, and D Feedwater Pump Anticipatory Trip to initiate a Reactor and Main Turbine Trip at 0342:16 hours. The CRO immediately closed 1C-124.

At 0342:23 hours, 1A and 1B Motor Driven Emergency Feedwater Pumps (MDEFDWPs) started on low Feedwater Pump discharge pressure. 1B MFDWP did not trip and the CRO secured the MDEFDWPs at 0343:06 hours, after verifying proper operation of the 1B MFDWP. The automatic control of 1-FDW-315 (Emergency Feedwater Loop A throttle valve) was disabled due to the failure of a solenoid valve. This resulted in no Emergency Feedwater flow to A Steam Generator. This was not identified during the Post-Trip Review. The Turbine Driven Emergency Feedwater Pump did not actuate due to a time delay that allows the pump to reset automatically if the automatic initiation signal is present for less than fifteen seconds.

All full length control rods [EIIS:ROD] fully inserted into the core and the reactor was shutdown.

Following the reactor trip, the Reactor Coolant System (RCS) average temperature decreased from 580 degrees F to approximately 555 degrees F. RCS pressure decreased from approximately 2145 psig to 1985 psig. Pressure then slowly increased to 2130 psig. Pressurizer [EIIS:VSL] level reached a minimum of 136 inches and stabilized at approximately 150 inches. Steam Generator (SGs) pressures increased to a maximum of 1009 psig and then decreased to a minimum of 892 psig on both A and B SGs before leveling off at approximately 1000 psig. SGs levels decreased to a minimum of 18 inches for approximately 14 seconds on both SGs before the 25 inch post trip setpoint was maintained.

During a routine inspection of equipment on May 8, 1992 at 0730 hours, a leak was discovered on the impulse line connected to the 1A MFDWP suction line.

CONCLUSIONS

The root cause of this event is Management Deficiency, lack of 'task specific' procedure and less than adequate training given. When the Emergency High HW level alarm was received the Alarm Response Manual was referenced. It was determined by Operations personnel that the Alarm Response Manual did not provide the proper guidance to reduce HW level during this condition. Operators were concerned with the HW level trending upward and extending past the level instrumentation range (0 to 7 feet) and flooding the suction line of the Condensate Steam Air Ejectors. The Operators felt a need to reduce HW level, realizing they would be at this power level for two hours, because they were waiting for the completion of shell warming of the Main Turbine [EIIS:TA]. A decision was made to divert a portion of Condensate flow to the Upper Surge Tank (UST) and then to the Condensate Storage Tank, where it could be pumped to another unit. The volume between 1C-124 and 1C-128 is large. The Operators were not aware that the piping was empty due to evaporation of the water to the UST, via leakage through 1C-128. Upon opening 1C-124 the void in the piping was filled, reducing the suction pressure of the Main Feedwater Pump, thus decreasing the pump discharge pressure. The flow path utilized by the Operators to lower HW level was performed because of a lack of understanding on the proper method to reduce HW level. Additionally a procedure did not exist to reduce HW level under this operating condition.

**LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION**

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

Response of the primary system to the trip was normal. Reactor Coolant System inventory, pressure, and temperatures were all maintained within the normal post-trip range. The immediate response of the secondary system was also normal. Both steam generators' pressure and level were maintained at or near their proper setpoints.

A review of events over the last two years, indicates that this is not a recurring problem.

The leak discovered on the impulse line (1/2 inch, carbon steel, ASTM A106, grade B, seamless, schedule 40) for the 1A Main Feedwater Pump was corrected under work request 37321C by replacing the damaged section with a new section of piping. The probable cause of the failure was due to the pressure surge during the Feedwater transient. Engineering is currently evaluating the cause of the piping material failure. A search for the piping material manufacture was performed and the manufacturer could not be determined. This piping is Duke Class G (Non-Safety) and was installed during the initial construction of the plant.

A solenoid valve (SV) failure disabled the automatic control of FDW-315, the Emergency Feedwater Loop A throttle valve. The SV is normally energized but is required to operate to the de-energized position upon Emergency Feedwater actuation to permit automatic control. The failure of this valve and the violaton of the Technical Specification will be addresed in Licensee Event Report 269/92-05.

The equipment failure of 1A Main Feedwater Pump suction line instrumentation piping and 1-SV-200 is NPRDS reportable. The manufacturer and Model number for the piping material is unknown. The SV was a Valcor V-70900-21-3 and the serial number is 1495. There was no release of radioactive material or exposure to radiation involved. This event did not involve any personnel injuries.

<u>CORRECTIVE ACTIONS</u>

Immediate

1. The CRO Closed 1C-124

2. Operations personnel took appropriate actions per the Emergency Operating Procedure to bring the unit to stable conditions.

Subsequent

1. Enclosure 3.22 (Control of High Hotwell Level) was added to OP/0/A/1106/02 (Condensate and Feedwater System) as a written method to reduce high Hotwell level.

2. The Alarm Response Manual for Hotwell Level Emergency High Statalarm (1SA6 / C-12) was revised to reference OP/0/A/1106/02 (Condensate and Feedwater System) enclosure 3.22 (Control of High Hotwell Level).

Planned

1. Operator training will be conducted to inform Operators of the Hotwell level oscillations and the correct method of reducing Hotwell level.

SAFETY ANALYSIS

Low Main Feedwater Pump (MFDWP) discharge pressure is an anticipated transient and is described in Section 10.4 of the Final Safety Analysis Report. Low MFDWP discharge initiates a reactor trip and starts the Emergency Feedwater (EFDW) System to provide decay heat removal. In this event all the systems and equipment operated as designed to mitigate the consequences of low MFDWP discharge pressure. Instrumentation detected the low MFDWP discharge pressure, initiated the Main Turbine and Reactor trips, and provided the start signal to EFDW System. Both Motor Driven Emergency Feedwater Pumps (MDEFDWPs) started as required. The MFDWP did not trip, after verifying proper operation of MFDWP the Operators secured the MDEFDWPs. The health and safety of the public was not compromised by this event.

NRC FORM 366A
(6-89)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

APPROVED OMB NO. 3150-0104
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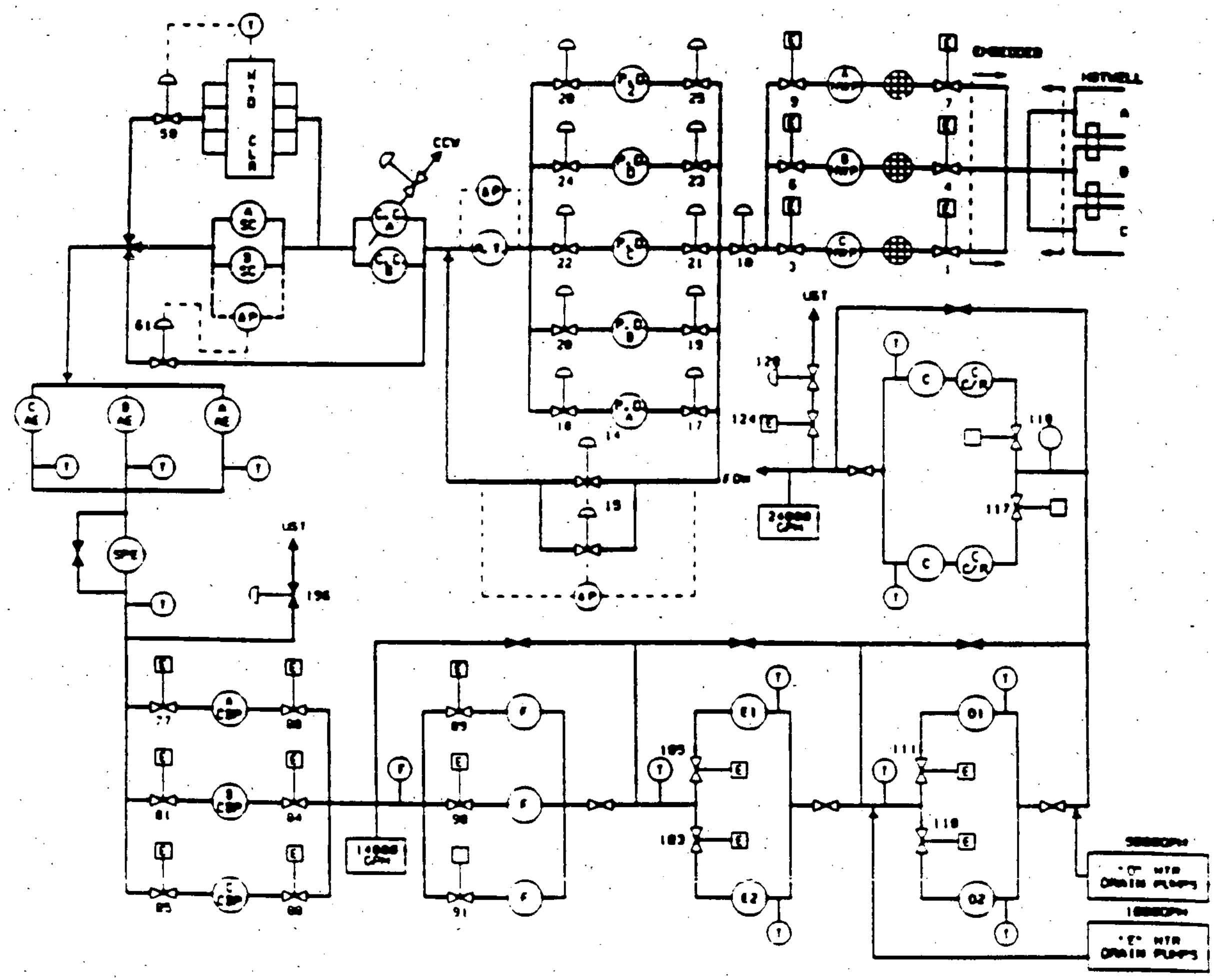
TEXT (If more space is required, use additional NRC Form 366A's) (17)

DUKE POWER COMPANY

ATTACHMENT 1

CONDENSATE SYSTEM ARRANGEMENT



NRC Form 366A (6-89)