CATEGORY 1

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ACCESSION NBR:9611050342 DOC.DATE: 96/10/24 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 96-04-00:on 960924,secondary drain line repture results in a Manual Reactor Trip occurred. Caused by deemed appropriate to prevent recurrence. Main Steam block valves, 2MS-76 & 79 were closed. W/961024 ltr.

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TITLE: 50.73/50.9 Licensee Event Report (LER), Incident Rpt, etc.

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CATEGORY 1 DOCUMENT

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DUKE POWER

October 24, 1996

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

Subject: Oconee Nuclear Station Unit
Docket Nos. 50-269, -270, -287
Licensee Event Report 270/96-04
Problem Investigation Process No.: 2-096-1828

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a) (1) and (d), attached is Licensee Event Report 270/96-04, concerning a secondary drain line rupture resulting in a manual reactor trip. Upon completion of the investigation a LER supplement will be issued to fully identify the cause(s) and the corrective actions deemed appropriate to prevent recurrence.

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
Oconee Nuclear Site

IE22 /1

Attachment

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NRC Resident Inspector
Oconee Nuclear Station

NRC FORM 366
(4-95)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

APPROVED OMD NO. 3150-0104
EXPIRES: 04/30/98
ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Oconee Nuclear Station, Unit 2	05000 270	1 0f 7

TITLE (4)

Secondary Drain Line Rupture Results In A Manual 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
09	24	96	96 -	04 -	00	10	24	96		05000

OPERATING MODE (9): N

POWER LEVEL (10): 60

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)	
Lanny V. Wilkie, Safety Review Manager	(864)	885-3518

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)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
X YES (*if yes, complete EXPECTED SUBMISSION DATE)*	NO		12	10	96

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

On September 24, 1996, at approximately 1642 hours, while operating at 60% full power, Unit 2 experienced a drain pipe rupture on the secondary side of the plant. At 1643 hours, the resulting steam release from the pipe rupture was isolated by closing the Main Steam supply block valves to the Second Stage Reheat system and manually tripping the reactor. Investigation into the cause(s) of the pipe rupture has not yet been completed. Upon completion of the investigation a LER supplement will be issued to fully identify the cause(s) and the corrective actions deemed appropriate to prevent recurrence.

Background:

Moisture Separator/Reheaters generate large quantities of saturated or near saturated liquid which is routed either to the feedwater heater drain [EIIS:SN] system or to the condenser [EIIS:COND].

Each pair ('A' and 'B') of the Moisture Separator/Second Stage Reheaters (SSRH) [EIIS:RHTR] drains into an identical SSRH drain system. The drains flow to 'A' and 'B' Second Stage Reheater Drain Tanks (SSRHDT), respectively, which act as a reservoir. Downstream of the SSRHDTs the drain line divides, one branch going to the condenser via dump valves 2HD-25 (SSRH Drain '2A' Dump To Condenser) and 2HD-26 (SSRH Drain '2B' Dump To Condenser). The second branch drains directly to the 2A1 and 2A2 Feedwater Heaters and includes block valves 2HD-91 (2A SSRHDT Level Control Inlet Block) and 2HD-94 (2B SSRHDT Level Control Inlet Block), level control valves 2HD-92 (2A SSRHDT Level Control) and 2HD-95 (2B SSRHDT Level Control), and block valves 2HD-93 (2A SSRHDT Level Control Outlet Block) and 2HD-96 (2B SSRHDT Level Control Outlet Block).

Basically, if the route to the Feedwater Heaters is aligned, the drains will take this route and be controlled by the low level controller on the drain tank. If this route is isolated, the drain tank level will rise until it reaches the high level controller and drains will be automatically directed to the condenser.

The purpose of Operations procedure, OP/2/A/1106/14 (Moisture Separator Reheater), is to provide direction for the operation of the Moisture Separator Reheaters. The purpose of OP/2/A/1106/14, Enclosure 3.6 (Abnormal Operating Conditions-First Stage and Second Stage Reheater Drain Operation With First and Second Stage Reheaters In Service) is to provide guidance for the isolation and restoration of the associated drain system, during power operation, in the event that the need arises.

EVENT DESCRIPTION:

On September 23, 1996, Unit 2 was escalating in power during a forced outage startup. The Moisture Separator Reheater procedure, OP/2/A/1106/14 (Moisture Separator Reheater), Enclosure 3.6 (Abnormal Operating Conditions - First Stage and Second Stage Reheater Drain Operation With First and Second Stage Reheaters In Service) was entered to dump the Second Stage 2A and 2B drains to the condenser. To achieve this configuration, procedure steps 2.3 and 2.4 of Enclosure 3.6 were performed. Step 2.3 required the closing of valves 2HD-91 and 2HD-94. Step 2.4 required verification that valves 2HD-25 and 2HD-26 were open.

On September 24, 1996, the Operations Shift personnel received verbal instructions from the Operations Unit Coordinator regarding the initial conditions to reopen 2HD-91 and 2HD-94.

At approximately 1448 hours, Unit 2 power escalation was stopped to investigate level control problems with the 2B Second Stage Reheater Drain Tank (SSRHDT). Two separate alarms had been received, one of which indicated a high level in the 2B SSRHDT and the other indicated that valve 2HD-26 was closed.

At approximately 1500 hours, the Operations Shift personnel initiated a work request for the investigation and repair of the SSRHDT level control problems. Further evaluation of the problem by the Operations Shift personnel resulted in resuming the power escalation.

At approximately 1600 hours, the Instrument and Control (I&C) technicians, investigating the SSRHDT level control problems, informed the Operations Shift personnel that level control problems were most likely due to valves 2HD-91 and 2HD-94 being closed. Shortly afterwards, the Control Room Senior Reactor Operator and Unit Supervisor made the decision to begin the Second Stage Reheater 2A and 2B drains feed forward activity by opening valves 2HD-91 and 2HD-94. The Operations Shift personnel conducted a pre-job briefing, using Enclosure 3.6 of OP/2/A/1106/14, with the Non-Licensed Operators (NLO) who were assigned to open valves 2HD-91 and 2HD-94. During this briefing, it was emphasized to the NLOs to open these valves slowly.

At approximately 1630 hours, the NLOs left the Control Room to begin the feed forward activity.

At approximately 1640 hours, the NLOs began opening valves 2HD-94 and 2HD-91 as instructed during the pre-job briefing.

At approximately 1642 hours, with Unit 2 at approximately 60% full power, a rupture of a SSRH line, downstream of 2HD-94, occurred. The release of steam resulted in burn injuries to seven employees (five NLOs and two I&C technicians). There were no radioactive releases, radioactive exposures to employees, or contamination of employees.

At approximately 1643 hours, the Unit 2 Control Room received an emergency phone call identifying the steam release and personnel injuries. Control Room personnel took immediate action to isolate the steam release by closing Main Steam block valves 2MS-76 (MS to 2A1 and 2A2 SSRHs) and 2MS-79 (MS to 2B1 and 2B2 SSRHs) and then, manually tripping the reactor.

Valve 2HP-120 (RC Volume Control) automatically opened to provide additional make-up flow to the Pressurizer [EIIS:PZR]. This diverted some of the flow from the Reactor Coolant Pump (RCP) seals. The low seal flow circuit automatically started the 'A' High Pressure Injection (HPI) [EIIS:BG] Pump to provide additional flow to the RCP seals. This is a normal function of the HPI makeup system and is separate from the Engineered Safeguards function.

Specific post-trip parameters remained within acceptable limits. Reactor Coolant System (RCS) [EIIS:AB] pressure decreased to 1932 psig after the reactor trip from 2150 psig. RCS pressure then slowly returned to 2151 psig. Pressurizer inventory remained on scale between a high of 220 inches and a low of 111 inches post trip before increasing and stabilizing at 114 inches. RCS average temperature was approximately 579 F before the trip then decreased to approximately 550 F post trip. Immediately following the trip, the 2A and 2B Steam Generator (SG) pressures reached a post trip high of approximately 1091 and 1087 psig, respectively. The SG pressures then decreased to 905 and 902 psig, respectively.

At approximately 1644 hours, the Medical Emergency Response Team was dispatched to provide medical aid to the injured employees.

At approximately 1653 hours, Site Assembly was conducted to aid in the identification of all injured employees and to help to prevent any further injuries. The Operations Shift Manager requested partial activation of the Emergency Response organization (Technical Support Center and Operational Support Center).

At approximately 1700 hours, a Failure Investigation Process Team was established to determine the root cause(s) for the pipe failure.

At approximately 1704 hours, the first ambulance arrived at the Site and advanced medical care was started.

At approximately 1735 hours, the Technical Support Center and Operational Support Center were fully staffed and declared operational.

An Event Investigation Team was established at the request of the Site VP to provide a non-site perspective as to the cause(s) of the event and lessons learned.

At approximately 1753 hours, the last injured employee was transported from the Site to Oconee Hospital.

At 2040 hours, a Notification of Unusual Event (NOUE) was declared and terminated. The NOUE was declared as a conservative action after the assessment of the collateral damage to plant equipment adjacent to the steam release area.

CONCLUSIONS:

The manual reactor trip was a proper and conservative response to the pipe rupture.

The Failure Investigation Team's preliminary conclusion was that a severe water hammer was the immediate cause of the pipe rupture. This team also concluded that erosion/corrosion was not a factor in the piping failure.

The Event Investigation Team's investigation will establish the root cause(s) for the pipe rupture event. Their conclusions will be provided in a supplement to this LER.

A review of events for the past two years indicates that the manual tripping of a reactor at Oconee Nuclear Station is not a recurring problem.

There were no equipment failures that were NPRDS reportable.

CORRECTIVE ACTIONS:

Immediate:

1) Main Steam block valves, 2MS-76 and 79 were closed and the reactor was manually tripped to isolate the steam supply to the Second Stage Reheat drain system.

Subsequent:

1) A Site Assembly was initiated to account for all Site personnel.

2) The Technical Support Center and Operational Support Center were activated to provide additional support to the operating shift.

3) A Failure Investigation Process Team was established to investigate the cause(s) of the pipe rupture and inspect for similar conditions.

4) An Event Investigation Team was established to evaluate the actions leading up to the pipe rupture and the actions in response to the pipe rupture.

Planned:

1) Unit 2 will remain off line until necessary repairs have been made.

2) Other corrective actions addressing the EIT and FIP investigations will be provided via LER supplement.

SAFETY ANALYSIS:

At approximately 1643 hours, the reactor was manually tripped, as a conservative action, to aid in the steam isolation for personnel protection.

Following the reactor trip, plant response was as designed with no safety system actuations. The consequences of a steam line break accident are analyzed in the Updated Final Safety Analysis Report (UFSAR), section 15.13, "Steam Line Break Accident". The UFSAR accident scenario is the double ended rupture of a thirty four inch main steam line between the reactor building and a turbine stop valve with the unit operating at rated power. The rupture that occurred in the heater drain system was well within the bounds of this analysis and did not challenge the plant from a nuclear safety perspective. No equipment important to nuclear safety was affected by the pipe rupture.

Prior to the reactor trip seven employees were injured due to the steam release from the pipe rupture. All seven individuals were transported from the Site, via ambulance, to receive professional medical care. This event did not result in any personnel exposures, contamination, or release of radioactive materials. This event had no impact on the health and safety of the public.