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FACIL: 50-270 Oconee Nuclear Station, Unit 2, Duke Power Co. 05000270

AUTH. NAME	AUTHOR AFFILIATION
NORTH, P. J.	Duke Power Co.
TUCKER, H. B.	Duke Power Co.

RECIP. NAME	RECIPIENT AFFILIATION

SUBJECT: LER 87-003-00: on 870408, nonisolable leak observed from cracked pipe on reactor vessel level instrumentation sys. Caused by const error. Weld overlay performed, stiffener designed & attached. W/870508 ltr.

DISTRIBUTION CODE: IE22D COPIES RECEIVED: LTR 1 ENCL 1 SIZE: 5
TITLE: 50.73 Licensee Event Report (LER), Incident Rpt, etc.

NOTES: AEOD/Ornstein: 1cy. 05000270

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EXTERNAL:	EG&G GROH, M	5	5	H ST LOBBY WARD	1	1
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APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Oconee Nuclear Station, Unit 2	0 5 0 0 0 2 7 0	1 OF 0 4

TITLE (4)
Reactor Shutdown Due To Non-Isolable Leak In The Reactor Coolant System

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 NAMES	DOCKET NUMBER(S)
0 4	0 8	8 7	8 7	0 0 3	0 0	0 5	0 8	8 7		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 0 0

[ ] 20.402(b)	[ ] 20.405(c)	[ ] 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)	[X] 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	
Philip J. North - Licensing	7 0 4	3 7 3 - 7 4 5 6

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
B	A B			NO					

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 April 8, 1987, at 0600, Unit-2 was at 240 degrees F and 170 psig in order to check Decay Heat Cooler efficiency. Personnel entered the Reactor Building and observed water leaking from a cracked pipe on the Reactor Vessel Level Instrumentation System (RVLIS). The leak was considered non-isolable; therefore an Unusual Event was declared, and the unit was taken to cold shutdown conditions.

The root cause of the event was determined to be an error in construction of the branch connection resulting from a misinterpretation of the modification drawing. This increased the possibility of the pipe being exposed to its natural frequency and invalidated the seismic design of the branch connection. Subsequent corrective action included repair of the crack by weld overlay and inspection of similar taps. A similar event occurred on Unit 3 on March 31, 1987 (see LER 287/87-04).

In the event that this line had sheared at full power, the resulting Small Break LOCA would be within the bounds of FSAR Small Break LOCA analysis. In addition, the Reactor Building would contain any radiological release. There were no releases of radioactivity outside containment, as such, the health and safety of the public was not affected.

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☆U.S.GPO:1984-0-454-481/18759

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Oconee Nuclear Station, Unit 2	0 5 0 0 0 2 7 0	8 7	— 0 0 3	— 0 0	0 2 OF 0 4

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

BACKGROUND

The Reactor Vessel Level Instrumentation System (RVLIS) was installed on Unit-2 during its latest refueling outage. RVLIS is used by the operators to determine the level of water covering the core during accident conditions. One of the level transmitter taps is welded to the 12" Decay Heat System suction line where it comes off of the Reactor Coolant System (RCS). There are no isolation valves between the RCS and where this crack developed.

The Oconee Pipe Specification, which references the ASME Pipe Specification, Section XI IWA 7400, requires only visual inspection for welding done to pipes less than 1 inch outside diameter. In addition to the pipe specification, the Reactor Building is toured at operating system pressure to look for any leaks prior to unit startup after an outage.

DESCRIPTION OF INCIDENT

In early 1986, the type drawings used for piping installations was changed from a piping drawing that shows full pipe diameters to one line drawings of computerized isometrics showing only the centerline of the piping. No training concerning this change was given to any personnel associated with implementation of modifications.

To expedite the RVLIS installation and reduce exposure to workers, part of the modification was pre-fabricated in the construction area. The pipe that cracked was welded to the coupling and an isolation valve for a level transmitter per the isometric drawing. The piping section was cut 6" longer than the isometric drawing had called for due to misinterpretation of the drawing dimensions by the pipe fitter. This piping section welded to the Decay Heat System suction piping on September 11, 1986. The Quality Assurance Inspectors met all Section XI requirements by inspecting before and after weld. However, they also misinterpreted the pipe length from the drawings.

During Unit-2 heatup on October 15, 1986, an inspection was made in the Reactor Building per procedures at normal operating pressure to ensure that any welds done during the recent refueling outage were not leaking. No visible leaks were observed.

On April 6, 1987 the unit was taken off-line and brought to 240 degrees F and 170 psig to determine the efficiency of the Decay Heat Coolers. While at 240 degrees F and 170 psig, maintenance personnel entered the Reactor Building to measure for a pipe support. At 0600 on April 8, 1987,personnel observed water coming from a welded connection. Control Room personnel were immediately notified of the situation. It was concluded that this section of pipe could not be isolated thus at 0607 an unusual event was declared. At 0708 the unit was taken to cold shutdown. All appropriate notifications were made. At 0110 on April 9, 1987 the weld was successfully repaired. At 1040 the unusual event was closed out.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Oconee Nuclear Station, Unit 2	0 5 0 0 0 2 7 0	8 7	— 0 0 3	— 0 0	0 3 OF 0 4

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

CAUSE OF OCCURRENCE

The root cause of this event was determined to be a construction deficiency. The leak was due to a crack in the heat affected zone of the pipe to coupling weld of one of the RVLIS level transmitters. The pipe was approximately 1 inch in diameter with a minimum of .219 inch thick wall (Schedule 160). The crack propagated circumferentially about 180 degrees around the pipe. The exact mode of failure of the level instrument pipe cannot be determined without removal and examinations of the affected pipe section. However, it is likely that the failure of this pipe was due to a weakening of the pipe wall due to stress induced by natural frequency vibration. Vibration was induced because the extra length of pipe put the pipe section outside the seismic stress design basis.

In addition, there were three contributing factors to this event.

1. Appropriate personnel were not given adequate training in determining pipe dimensions from the newly issued computerized isometric drawings. The personnel involved in this event had been trained on the job and were deemed qualified by management from prior performance. It is the responsibility of Management to ensure that adequate training is given to all appropriate personnel when a program change goes into effect.

2. The pipe fitter who measured and cut the pipe per the isometric drawing did not account for the radius of the connecting pipe, thereby making the pipe section too long.

3. The possibility of a misinterpretation in the new computerized isometric drawings was created due to an effort to provide easy to read installation drawings and a lack of specific training. The coupling is shown on the isometric in such a way that if the centerline designation is overlooked, a wrong measurement could result.

Previous pipe failures were caused by steam erosion on the secondary side of the plant. Therefore, there is no similarity between this pipe failure and the others that have occurred. Although the root cause of this event is the same as the event reported in LER 287/87-04, this is not considered a recurring event. Due to the time frame of the two events, the planned corrective actions for Unit 3 had not yet been fully implemented.

CORRECTIVE ACTIONS

The immediate corrective action was to bring the reactor to cold shutdown conditions.

Subsequent corrective actions were to:

o Perform a weld overlay over the section of pipe that was cracked.

- Design and attach a stiffener to the section of pipe for added restraint.
- Take vibration data prior to installation of the restraint to analyze the possible cause of the failure.
- Inspect similar taps in Unit-2 and the accessible tap in Unit 1.
- Perform a Dye Penetrant Test on the repaired line.
- A Task Force was formed to resolve possible discrepancies since the change to computerized isometric drawings.

Planned corrective actions are for the Task Force to:

- Develop training for craft personnel, management, and all other personnel who install modifications to interpret the correct dimensions of piping installation from isometric drawings.
- Inspect all modifications that were installed with isometric drawings that have the possibility of similar consequences.
- Perform safety analyses and initiate changes to modifications where appropriate.

ANALYSIS OF OCCURRENCE

Consequences of this event were mitigated in the early stages of the pipe failure due to early detection by personnel. This event is bounded by the Small Break LOCA analysis in the Oconee FSAR (Section 15.14.4.3) and is within the bounds of the automatic equipment in the plant to reestablish the water inventory to the primary system to keep the core covered.

The FSAR Small Break LOCA analysis addresses a break from $.0007ft^2$ to $.5ft^2$. The calculated break size if this line had sheared, is approximately $.00373ft^2$. Thus, the Emergency Core Cooling System has the capacity to keep the core covered to prevent damage to primary system components if this pipe would have completely sheared off at full power. In the event of a Small Break LOCA, Reactor Building integrity would have contained any radiological release within the building.

There was no release of radioactivity outside of the containment building as a result of this event, therefore, the health and safety of the public were not affected.

DUKE POWER COMPANY
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HAL B. TUCKER
VICE PRESIDENT
NUCLEAR PRODUCTION

TELEPHONE
(704) 373-4531

May 8, 1987

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

Subject: Oconee Nuclear Station, Unit 2
Docket No. 50-270
LER 270/87-03

Gentlemen:

Pursuant to 10CFR 50.73 Sections (a)(1) and (d), attached is Licensee Event Report (LER) 270/87-03 concerning a reactor shutdown due to a non-isolable leak in the Reactor Coolant System.

This report is submitted in accordance with §50.73(a)(2)(i). 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

Hal B. Tucker

PJN/70/sbn

Attachment

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