Previous Report Date 07/03/85

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)
Browns Ferry - Unit 3	0 5 0 0 0 2 9 6	1 OF 0 7

TITLE (4): Mismatch of Reactor Water Level Indicators

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 2	1 3	8 5	8 5	0 0 6	0 2	0 1	3 1	8 6		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N
POWER LEVEL (10): 0 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)	[X] 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)
Stephen B. Jones	205 729-2538

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)

During the startup of unit 3 on February 13, 1985, a mismatch between the GEMAC water level indicators developed while at low pressure and subsequently returned to normal. The situation was misdiagnosed, and the appropriate technical specifications were not implemented. The unit was shut down on March 9, 1985, and a number of investigations were conducted to determine the cause of the event. The net effect of the incident was a temporary introduction of a nonconservative error on the affected leg's instruments.

Since the event, training has been received by operators, plant management and shift technical advisors to enable them to more rapidly diagnose water level indication problems during startup operations. Also, the need for initiation of conservative action on discovery of off normal situations was stressed.

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

Unit 1 was in normal operation at 95 percent power, unit 2 was in a refueling outage, and unit 3 was in a startup from cold shutdown. Only unit 3 was affected by this event.

On February 13, 1985, at 2130, unit 3 was returning to service from a cold shutdown. The unit had been critical since 2058, and reactor pressure was approximately 35 psig. At this time, the unit operator noticed the "B" GEMAC water level indicator, LI-3-60, was indicating approximately 17 inches less than "A" and "C" GEMACs, LI-3-53 and LI-3-206. At 2136, a half scram was initiated by LIS-3-203D. The unit operator immediately raised the water level which cleared the half scram. The operator had five water level indicating instruments to show the water level in the reactor vessel (see Figure). Level indicators LI-3-53 and LI-3-206 are GEMAC instruments in channel "A," and they share a common reference leg. Level indicators LI-3-60 is a GEMAC instrument and is in channel "B." Yarway instruments LI-3-46A and LI-3-46B are in channel "A" and "B" respectively, and they have separate reference legs from the GEMAC instruments. When viewing the five level indicators, the operator initially concluded that LI-3-60 was in error since there were four other instruments indicating levels very close to each other at 37 inches and 40 inches. This conclusion caused subsequent problems because it was a misinterpretation of the information provided by the control room indicators. The technical specification requirement for this condition specifies that all operable rods are to be inserted within four hours. Due to the misinterpretation, rod movement was continued. By 2230, all three control room indicators were indicating the same level. Operability checks on LIS-3-203C and LIS-3-203D verified operability of redundant low water level scram switches. The event was red phoned to NRC on February 15, 1985, following review of the circumstances by management and engineering.

Browns Ferry unit 3 operated until March 9, 1985, at which time TVA removed the unit from service to conduct further investigations. Following shutdown, special test number ST-8502 was conducted in an effort to duplicate the operating conditions and level mismatch that had occurred. The large mismatch observed on February 13 was not reproduced during the performance of the test, although during the unit cooldown, a small level mismatch was observed. The level mismatch increased from one to four inches with reactor pressure approximately 60 psig, then level indication converged. On subsequent repressurization and depressurization, the mismatch was not duplicated.

Visual inspections of level sensing lines, welds, valves, and instruments were made both inside the drywell (BD) and outside the drywell. No leaking lines were initially discovered. Minute leakage at valve packings was noted but this leakage was not sufficient to have caused the mismatch experienced, and it is believed that this leakage existed only at operating pressure.

Liquid penetrant non-destructive testing was conducted on selected sensing lines and welds. During preparatory cleaning of the "A" channel reference leg line (X-28a) at penetration X-28 outside the drywell, a leak was discovered. No leak had been observed at this location during two previous visual inspections. The leak resulted from a crack in the 304 stainless steel sensing piping. From initial inspection of the crack, metallurgical personnel thought the crack was a fatigue crack. The leakage rate from the crack was determined to be approximately one drop every 2 minutes and 17 seconds with the reactor at atmospheric pressure (measured 13.5 milliliters in 3.5

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

hours). This leakage would have drained some water from the "A" channel reference leg which would not have been automatically made up via the condensing pot when the reactor was at low temperature. Such leakage could have caused or contributed to the level mismatch. The liquid penetrant examinations of the welds and sensing lines adjacent to the X-28 and X-29 penetrations revealed no other cracks or leaks.

Prior to any line repairs, the inservice inspection group performed ultrasonic testing of the GEMAC associated sensing lines both inside and outside the drywell. No bubbles were found in the sensing lines outside of the drywell. A bubble about 8-inches long and approximately 1/8 to 1/4-inch thick was found near penetration X-28 inside the drywell. No other bubbles were found.

During a time span on March 11 and March 12, 1985, with the reactor in cold shutdown, reactor water level (JB) indicators LIS-3-53 and LIS-3-206 developed a 19.5-inch indicated level variance compared with LIS-3-60 in a period of approximately 17 hours. Repair to the leak on the "A" channel reference leg had not been performed at this time. Ultrasonic testing in the drywell revealed that level in "A" reference leg was low by an amount appropriate to the indicated mismatch. The leakage rate from the cracked line was measured and calculations were made to determine if the crack leakage was the cause of the indicated level variance. It was concluded that the leak was the cause of the level variance.

The cracked portion of the line was removed along with an attached line (X-28a-1) which was no longer in use. The instruments which had been connected by the X-28a-1 line were associated with low pressure cooling injection (BO) loop selection logic which had previously been removed. Two pipe couplings and a short section of pipe taken from the removed line adjacent to the crack were welded in place. Subsequent liquid penetrant testing revealed that the short section of replacement pipe was also cracked. This cracked pipe was removed and replaced with another piece of piping taken from the removed line further away from the leaking cracked portion. This section of pipe tested satisfactorily.

The section of line containing the crack was polished and microscopically inspected onsite by metallurgical personnel. The inspection results revealed the crack to be transgranular stress corrosion instead of fatigue cracking as previously thought. The crack failure mode indicated halogen contamination was likely to exist, thus contaminant swipe checks were made at penetration X-28 and X-29 and associated piping outside the drywell. These tests showed chloride to be the major contaminant present.

Swipe checks were also taken inside the drywell from the X-28 and X-29 penetrations to the constant head pot for both "A" and "B" channel water level sensing lines. The chloride levels found inside and outside the drywell were only slightly above the expected background levels except adjacent to the X-28 penetration outside the drywell. The outside portion of penetration X-28 is located in the reactor water cleanup (RWCU) (CE) heat exchanger room. Significant levels of chloride contamination were found in this room. During inspection of unit 1 lines, pitting of the line just below the condensing chamber was observed. These lines were then inspected on unit 3 and swipe checks were taken. Results of checks for both units revealed acceptable levels of chlorides. It is believed the source of chlorides at the X-28 penetration was due to paint on the instrument lines from the painting of the drywell

penetration. The fire retardant in the paint is the actual source of the chlorides. Additionally, a steam leak from the RWCU heat exchanger was observed spraying in the vicinity of the instrument lines during the inspection at power. It is believed the temperature and humidity resulted in leaching of chlorides from the paint chips and resulted in conditions highly favorable for transgranular stress corrosion cracking. The chloride levels at the other locations are believed to be due to non-specific sources (e.g., perspiration, concrete dust, etc.).

Cleaning of the instrument lines, both inside and outside containment, was performed using stainless steel brushes, demineralized water, and methanol. Following cleaning, swipe tests were again performed to determine if decontamination efforts were effective. All areas were at satisfactory levels except the X-28 penetration transition piece and sensing lines located in the immediate area of the penetration external to the drywell. The lines and transition piece were again cleaned and swipe tests made. The results after additional cleaning revealed acceptable chloride levels. After chloride concentrations were reduced, quality control inspectors performed liquid penetrant dye testing of all sensing lines from X-28 penetration to the wall of the RWCU heat exchanger room. This area was selected because this was the location of the leak, and was also the only area where levels of chloride contamination significantly above background levels were detected. The examination revealed all lines were satisfactory. An attempt to assess the potential for undetected leakage in the area of the sensing lines was made by taking swipes and measuring the contamination levels on the wall and floor area below the sensing lines. No contamination gradient indicative of preexisting leakage was detected.

Special Test ST-8503 was prepared and initiated to determine if there was any evidence of post-repair leakage or communication between the reference and variable legs (via equalizing valves, differential pressure transmitter diaphragms, etc.). Leakage at these locations may be proven or discounted by observing a level discrepancy trend and sequentially isolating the instruments at the panel and monitoring the discrepancy trend for change. Following the sensing line repair, all associated instrumentation was returned to service and was monitored for leakage until March 27, 1985, when the special test was discontinued. No discrepancies were noted during the test or since return to service of the instruments to date.

The observed level mismatch was most likely caused by a loss of water in the "A" instrument reference leg. This investigation has revealed two probable causes of the loss of water in the "A" reference leg.

One possible cause of the level mismatch was reference leg leakage via the transgranular stress corrosion cracking that existed adjacent to the X-28 penetration in the reactor water cleanup heat exchanger room. Leakage from the crack may have varied in quantity due to external forces applied to the line from thermal expansion, film coating over the interior surface of crack passage, pressure changes, or unknown reasons. The transgranular cracking was caused by chloride contamination.

The more probable cause, as indicated by supporting calculations which may have been enhanced by the above listed cause, is the potential for the presence of air bubbles in the "A" reference leg. During the refueling outage preceding this event, the "A" reference leg was drained when the vessel level was lowered to accomplish jet pump

instrument nozzle repairs. In this maintenance process, water level was lowered to a point below the range of control room level indicators, and temporary level instrumentation was connected to the "below core plate" sensing line and the drained "A" reference column. When vessel level was returned to normal, the "A" reference leg was backfilled through an instrument drain line at panel 25-51 using a hose and demineralized water pressure. Due to the number and character of restrictions to flow when backfilling, in conjunction with high points which have been determined to exist in horizontal runs, a bubble may have been trapped in the sensing line. Additionally, it was established that the reactor was maintained at negative pressures (via the main condenser) for several days prior to the November 20, 1984 (discussed below) and February 13, 1985 startups. This, in conjunction with the previously listed cause, could potentially contribute to the introduction of air in the horizontal runs of the reference leg. Upon startup, the bubble could have been compressed in a horizontal run or escaped, thus causing a decrease in water level in the reference leg. The inability to reproduce the level mismatch during shutdown, and the post shutdown ultrasonic examination of the sensing line lend more credibility to escape, rather than compression, of a bubble in the sensing line.

An experience review survey was conducted for related Browns Ferry events and available BWR data. An Licensee Event Report (LER) review indicated two similar occurrences at Browns Ferry:

- In August 1977, during startup of unit 2, "B" reference column instruments read high by 20 inches. Sensing lines and valves checked for leakage. The reference leg was backfilled.

- In May 1981, during startup of unit 3, LI-3-53 and LI-3-206 failed upscale, and a shutdown commenced. The reference leg was backfilled, and the instruments were brought back into agreement.

A review of unit 3 was made for startups since the refueling outage end in November 1984. There had been six startups on unit 3, with four from cold conditions. Review of logs and recordings indicated an essentially identical but less pronounced event occurred during startup on November 20, 1984. This was attributed to a bubble in the reference column. It is noted this was the first startup to pressure after a lengthy refueling outage.

Industry data search yielded 22 other BWR events related to water level instrument problems. Several of the events appear to be directly analogous in that mismatches were observed during shutdown or startup situations. For those events for which a cause could be ascertained, leaking fittings or valves were implicated. During power operation, level events involving mismatches appear abruptly and involve offscale readings. At low pressures or shutdown, mismatches occur gradually as observed at Browns Ferry. Details of these events are included in the engineering report on this matter.

Concerning safety ramifications of the event, the following observations may be made.

- Shift personnel promptly restored and maintained water level throughout the situation.

- Redundant reactor protection system instrumentation was operable.

- The mismatch introduced about a 30-inch error in sensed level in the affected leg. Assuming a single failure in unaffected leg, this error in actuation setpoint would be of low consequence considering the reactor power and pressure.

Selected primary containment isolation system logic is also on these instrument columns and the same conclusions as above are pertinent. High pressure coolant injection (BG) and reactor core isolation cooling (BN) trip logic (high water level) were also affected. These systems are, however, inoperable at low reactor pressure. Experience review indicates that the same scenario is unlikely at rated pressure since failures are abrupt and automatic functions will occur prior to operator intervention.

As explained earlier, the root cause of the event has not been explicitly determined. Reference leg leakage or bubble formation is strongly suggested. The net effect was introduction of a temporary nonconservatism in the instrumentation setpoints. Analysis of the operator action also points out the need for additional training in diagnosing water level instrumentation problems at off-rated condition.

The reactor water level instruments are calibrated to be most accurate for certain conditions of pressure and temperature. The calibration conditions used for level instruments are those corresponding to rated pressure (1000 psig) and rated temperature (546°F). The method for compensating for off-rated conditions varies from instrument to instrument. For the two of concern here, the compensation for the Yarways is a temperature compensation using heat clamps and the compensation for the GEMACs is an electronic pressure compensation. These methods of compensation result in water level indications at off-rated conditions which are not exactly indicative of actual level but the variances are predictable. The operator training program traditionally has stressed level instrument response and malfunctions under emergency conditions. Simulator exercises for the diagnosis of level indicator problems will be reviewed and expanded to embrace a general set of postulated malfunctions. This action will be completed in time for the next training rotation (April 1986). This additional training is particularly appropriate in that the experience search indicates this type event is not particularly unusual and is also consistent with the importance placed on water level instrumentation. The reported events at low power have relatively minor safety consequences; however, this additional training may assist operators in avoiding more serious conditions under more unfavorable circumstances.

In the interim, training has been provided to operators, shift technical advisors, and plant management (plant manager, plant superintendents, engineering group supervisor, compliance supervisor, reactor engineering supervisor, shift technical advisor unit supervisor) to enable them to more rapidly diagnose water level indication problems that may occur during startup and normal operation and take appropriate technical specification action. The need to take conservative action promptly in the face of unusual or irregular occurrences was also stressed.

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As discussed in the text, the exact cause of the event has not been definitely identified. Corrective actions for the probable causes have been the repair of the cracked sensing line and preparation of a special test that requires ultrasonic testing prior to the next unit 3 startup to verify that the reference legs to the scram Bartons and GEMACs are filled to the condensing chambers. Also, the level indicators will be carefully observed during the initial pressurization to rated conditions.

Details of the engineering report and supporting material are available at the site.

Responsible Plant Section - N/A

Previous Events -BFRO-50-296/81027, 50-260/7711, 50-260/81014

TENNESSEE VALLEY AUTHORITY
Browns Ferry Nuclear Plant
P.O. Box 2000
Decatur, Alabama 35602

January 31, 1986

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

Dear Sir:

TENNESSEE VALLEY AUTHORITY - BROWNS FERRY NUCLEAR PLANT UNIT 3 - DOCKET NO. 50-296 - FACILITY OPERATING LICENSE DPR-68 - REPORTABLE OCCURRENCE REPORT BFRO-50-296/85006 R2

The enclosed report provides additional details concerning the mismatch of reactor water level indicators. This report is submitted in accordance to 10 CFR 50.73 (a)(2)(iv) and (vii).

Very truly yours,

TENNESSEE VALLEY AUTHORITY

Robert L. Lewis

Robert L. Lewis
Plant Manager
Browns Ferry Nuclear Plant

Enclosures
cc (Enclosures):

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