TENNESSEE VALLEY AUTHORITY

CHATTANOOGA, TENNESSEE 37401

400 Chestnut Street Tower II

August 28, 1985

Mr. James M. Taylor, Director
Office of Inspection and Enforcement
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Taylor:

This is in response to J. Nelson Grace's July 22, 1985 letter to H. G. Parris which transmitted the Proposed Civil Penalty Action: EA 85-51 (IE Inspection Report Nos. 50-259/85-13, 50-260/85-13, and 50-296/85-13) for Browns Ferry Nuclear Plant. Our response to the violation is provided in the enclosure. On August 21, 1985, I discussed with Dave Verrelli of your staff an extension to August 28 for submitting this response.

Fees in response to the proposed civil penalty of $112,500 are being wired to the NRC, Attention: Office of Inspection and Enforcement. This fee is being sent in accordance with your letter to H. G. Parris dated August 5, 1985 regarding EA 84-136 which recommended we reduce the fees from $150,000 to $112,500

If you have any questions, please call R. E. Alsup at FTS 858-2725.

To the best of my knowledge, I declare the statements contained herein are complete and true.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

J. A. Domer

J. A. Domer, Chief
Nuclear Licensing Branch

Enclosure
cc (Enclosure):
U.S. Nuclear Regulatory Commission
Region II
ATTN: Dr. J. Nelson Grace, Regional Administrator
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

Mr. R. J. Clark
Browns Ferry Project Manager
U.S. Nuclear Regulatory Commission
7920 Norfolk Avenue
Bethesda, Maryland 20814

IE-14
1/1

ENCLOSURE
RESPONSE
NOTICE OF VIOLATION AND PROPOSED IMPOSITION
OF CIVIL PENALTIES: EA 85-51 (INSPECTION REPORT NOS.
50-259/85-13, 50-260/85-13, AND 50-296/85-13)
J. NELSON GRACE'S LETTER TO H. G. PARRIS
DATED JULY 22, 1985

Item 1

Technical Specification 3.1 requires that for the reactor protection system there be two operable or tripped trip systems for each function (Table 3.1.A). If two instrument channels for the reactor low water level trip system are not operable for both trip systems, the appropriate actions shall be taken, including the initiation of insertion and completion of insertion of all operable control rods within four hours.

Contrary to the above, this requirement was not met on Unit 3 for the reactor low water level trip system in that two reactor water level instruments (LIS-3-203 A, B) were inoperable on February 13, 1985. At this time, sufficient redundant water level indication existed which should have alerted the licensee that it was in an action statement condition. The licensee did not initiate and complete insertion of operable control rods within four hours as required by the Technical Specification.

This is a Severity Level II violation (Supplement I).
(Civil Penalty -$100,000)

1. Admission or Denial of the Alleged Violation

   TVA admits the violation.

2. Reasons For the Violation

   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 indicator 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 movements were 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.

Although sufficient redundant water level indication existed at the time to diagnose a nonconservative error in water level instruments LIS-3-203 A and B, operator training was not sufficient to ensure correct diagnosis within the timeframe that the condition existed. Previous operator training had emphasized diagnosis at rated conditions and operator actions for the significant water level errors which occur under accident and degraded conditions and which affect adequate core cooling. The complexity introduced by the calibration of the various instruments at off-rated conditions impeded diagnosis until after the condition had returned to normal.

3. Corrective Steps Which Have Been Taken and Results Achieved

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 samll 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 intially 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 to be approximately one drop every 2 minutes and 17 seconds with the reactor at atmospheric pressure (measured 13.5 milliliters in 3.5 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 to1/4 inche thick was found near penetration X-28 inside the drywell. No other bubbles were found.

During a time span between 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 of 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-23a-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 sesnsing 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 instsrument 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 dusts, 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 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 lsocation 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-85-3 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 be 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 poentially 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 essesntially 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 appears 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 an 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 cooling injection (BG) and reactor core isolation cooling (BF) 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

Operators, plant management, and shift technical advisors received training to enable them to more rapdily diagnose water level indication problems with emphasis on calibration of the various instruments at off-rated conditions and its effect on comparison between instruments. The need for conservative action on discovery of off-normal situations was also stressed.

4. <u>Corrective Steps Which Will Be Taken to Avoid Further Violations</u>

The reactor water level instruments are calibrated to be most accurate for certain conditions of pressure and temperatured. The calibration conditions used for level instruments are those corresponding to rated pressure (1000 psig) and rated temperature ($546^{o}F$). 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 instrumentresponse 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 (January 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. Further, this training should preclude the confusion which led to the miscommunication and lack of aggressive action on the part of the operators in their resolution of the reactor water level indication discrepancies. 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.

The classroom training described in item 3 will also be repeated before the next unit startup.

5. <u>Date When Full Compliance Will Be Achieved</u>

Full compliance will be achieved on completion of the described training. This should be accomplished by April 15, 1986.

Item 2

10 CFR Part 50, Appendix B, Criterion XVI requires that measures be established to assure that conditions adverse to quality be promptly identified and corrected. These measures must assure that the cause of the condition is determined and corrective action is taken to preclude repetition.

Contrary to the above, on November 20, 1984 reactor vessel water level instrument problems were identified in that the Shift Engineer and Assistant Shift Engineer logs indicated that the GEMAC B narrow range instrument (LI-3-60) was reading approximately 11 inches lower than GEMACs A and C (LI-3-53 and LI-3-206 respectively). The cause of this discrepant condition was not determined and corrective actions were not taken to preclude repetition in that on February 13, 1985, the same instruments were similarly providing inconsistent readings.

This is a Severity Level III violation (Supplement I).
(Civil Penalty - $50,000)

1. Admission or Denial of the Alleged Violation

   TVA admits this violation.

2. Reasons For the Violation

   Due to the training deficiency outlined in response to violation 1, plant personnel and management did not correctly diagnose that the earlier event affected the vessel level instruments in a nonconservative manner and that a degraded condition existed. Failure to fully recognize the degraded condition led directly to failure to investigate and take appropriate corrective action.

3. Corrective Steps Which Have Been Taken and Results Achieved

   After the similar event of February 13, 1985, operators, management, and shift technical advisors received training as outlined in our response to Violation 1. This training has resulted in improving the plant staff's ability to correctly diagnose water level instrument anomalies which will cause appropriate identification of degraded conditions and initiation of corrective actions.

4. Corrective Steps Which Will Be Taken to Avoid Further Violations

   Refer to Violation 1.

5. Date When Full Compliance Will Be Achieved

   Refer to Violation 1.

Item 3

10 CFR 50.72 requires each nuclear power reactor licensee to notify the NRC as soon as practical and in all cases within one hour of the occurrence of any event during operation that results in the condition of the nuclear power plant being seriously degraded.

Contrary to the above, two low water level instrument channels, one in each Reactor Protection System trip system, were inoperable during a reactor startup on February 13, 1985. This seriously degraded condition was not reported to the NRC until approximately forty-three hours after the event occurred when a one-hour report was filed.

This is a Severity Level IV violation (Supplement I).

1. Admission or Denial of the Alleged Violation

   TVA admits this violation.

2. Reasons For the Violation

   The event occurred between approximately 2100 and 2230 on February 13, 1985. The entire nature of the instrumentation problem was not recognized at that time. Investigation and reportability was referred to plant management and engineering. An investigation was begun early on February 14, and it was determined during that day that instruments LIS 3-203 A and B were nonconservative. The NRC resident inspector was briefed on the progress of the investigation at 0700 and again at 1200 on that same day. However, due to the attention focused on the investigation, reporting pursuant to 10 CFR 50.72 was overlooked until the following day.

3. Corrective Steps Which Have Been Taken and Results Achieved

   On discovery that an Emergency Notification System (ENS) report for the event had been unintentionally neglected; an immediate report was initiated to the Commission in accordance with 10 CFR 50.72 requirements. In addition, responsibility has been placed with the Shift Technical Advisor unit supervisor to review each potentially reportable event, as documented by a Licensee Reportable Event Determination (LRED). Further, the supervisor will ensure each LRED is completed and dispositioned for ENS reportability in a timely manner and in accordance with 10 CFR 50.72 requirements.

4. Corrective Steps Which Will Be Taken to Avoid Further Violations

   No further corrective action is required.

5. Date When Full Compliance Will Be Achieved

   Full compliance has been achieved.