UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30323

Report Nos.: 50-259/85-13, 50-260/85-13, and 50-296/85-13

Licensee: Tennessee Valley Authority
500A Chestnut Street
Chattanooga, TN 37401

Docket Nos.: 50-259, 50-260 and 50-296 License Nos.: DPR-33, DPR-52, and DPR-68

Facility Name: Browns Ferry 1, 2, and 3

Inspection Conducted: February 20 - March 1, 1985

Inspectors: W. H. Rulland for — 3/26/85
G. Paulk — Date Signed

B. Wilson, for — 3/21/85
J. Munro — Date Signed

P. Wagner — 3/21/85
P. Wagner — Date Signed

P. Wagner for — 3/21/85
K. Poergner — Date Signed

Approved by: D. P. Johnson for — 3/21/85
C. A. Julian, Section Chief — Date Signed
Division of Reactor Safety

SUMMARY

Scope: This routine, unannounced inspection entailed 84 inspector-hours on site in the areas of reactor vessel level indication anomalies observed during the Unit 3 startup of February 13, 1985, and Unit 1 Control Rod 34-03 maintenance and testing beginning February 20, 1985.

Results: Of the two areas inspected, two apparent violations were found. (Failure to meet Technical Specification 3.1, paragraph 5.c; failure to follow procedures/inadequate procedures, paragraph 6).

REPORT DETAILS

1. Persons Contacted

Licensee Employees

*J. Coffey, Site Director
*G. Jones, Power Plant Superintendent
*J. Pittman, Assistant Power Plant Superintendent
*R. Hunkapiller, Operations Group Supervisor
*J. Thompson, Instrument Maintenance
*R. Burns, Instrument Maintenance Supervisor
*J. Ratliff, Engineering
*J. Carlson, Quality Assurance
*E. Cornelius, Mechanical Maintenance
*P. Border, Quality Assurance
*S. Jones, Compliance
*A. Burnette, Assistant Operations Group Supervisor
*J. Duke, Operations
*J. Wolcott, Engineering
*G. Henry, Engineering
*F. Poppell, Engineering
*B. Morris, Compliance Supervisor
*D. Mims, Engineering Group Supervisor

Other licensee employees contacted included operations, maintenance and engineering personnel.

NRC Resident Inspectors

*G. Paulk
C. Patterson
C. Brooks

*Attended exit interview

2. Exit Interview

The inspection scope and findings were summarized on March 1, 1985, with those persons indicated in paragraph 1 above. The licensee did not identify as proprietary any of the materials provided to or reviewed by the inspectors during this inspection.

3. Licensee Action on Previous Enforcement Matters

This subject was not addressed in the inspection.

4. Unresolved Items

Unresolved items are matters about which more information is required to determine whether they are acceptable or may involve violations or

deviations. New unresolved items identified during this inspection are discussed in paragraph 5.c.

5. Reactor Vessel Water Level Discrepancies

a. Chronology of Events

An NRC inspection was conducted to determine the circumstances surrounding the inoperability of two Unit 3 reactor protection system (RPS) reactor water level instruments (LIS-3-203 A, B) during a reactor startup on February 13, 1985, and the actions taken by licensee personnel in response to available reactor water level instrumentation. Based on document reviews and interviews with licensed operators, Shift Technical Advisors, cognizant engineers, maintenance personnel, supervisors, and management personnel, the following chronology was developed.

Date	Time	Event
2/13/85		With a unit startup in progress and the reactor critical, operators observed a discrepancy in the three narrow range reactor water level instruments in the control room.
	2130	Unit 3 operator called the instrument shop and asked if the Bartons were operable. Bartons are the reactor water level switches that provide an RPS scram on low level.
	2136	Unit 3 received a half scram, RPS B2 logic, from LIS-3-203 D. Unit 3 operator immediately raised reactor water level.
	2138	The half scram was reset. The operator informed the instrument shop about the half scram and inconsistent water level indication.
	2300	The Shift Engineer contacted the instrument shop to determine if level switches were operable.
	2315	The instrument maintenance supervisor was called by the instrument maintenance foreman to inform him of the water level problems.

	2330	The cognizant engineer was contacted by instrument maintenance personnel. All GEMAC narrow range indicators were again reading consistently and indicated normal reactor water level.
2/14/85	0045	A portion of Surveillance Instruction (SI) 4.1.A-7 was begun on level indicator switches LIS-3-203 C and LIS-3-203 D to verify that the scram set points of the instruments were within the required tolerance.
	0235	SI 4.1.A-7 on LIS-3-203 C and LIS-3-203 D was completed. The instruments were found within the required accuracy with no adjustments necessary.

b. Details

On February 13, 1985, Unit 3 received a half scram, RPS B2 logic, from LIS-3-203 D (D Scram Barton Level Indicator Switch). Approximately six minutes before the half scram, a licensed operator had contacted the instrument maintenance shop and had reported that the A and C narrow range GEMAC reactor water level indicators (LI-3-53 and LI-3-206 respectively) were indicating approximately 40 inches of water and the B narrow range GEMAC (LI-3-60) was indicating approximately 10 inches of water. None of the operations personnel interviewed were able to recall when the B GEMAC began to drift abnormally low. GEMAC A and C share a common reference leg along with the A and B scram Barton level indicator switches (LIS-3-203 A and LIS-3-203 B respectively). The B GEMAC shares a common reference leg with the C and D scram Barton level indicator switches (LIS-3-203 C and LIS-3-203 D respectively). Scram Bartons A and C provide input to Trip System A of the Reactor Protection System (RPS) and scram Bartons B and D provide input to Trip System B. The logic of the RPS is a one-out-of-two taken twice arrangement.

Browns Ferry Failure Investigation Report No. FI-85-29 indicated the following plant parameters at the time of the half scram (approximately 2136): A and C narrow range GEMAC - approximately 37 inches, B narrow range GEMAC - approximately 10 inches, Yarway A and B level indicators (LI-3-46 A, B) - approximately 40 inches, reactor pressure - approximately 40 psig, reactor temperature - approximately 286°F.

The half scram was reset by raising reactor water level in manual control. Interviews with licensed reactor operators indicated that the control rod withdrawal was continued within approximately five minutes after the half scram was reset. After the half scram was reset, a licensed operator checked local level indications in the reactor building and reported to the control room operator that scram Bartons

C, D, and LIS-3-185 (ADS Blowdown Permissive) were all indicating 15 to 17 inches. These instruments are on the same common reference leg as the B GEMAC narrow range indication available in the control room. The operator also checked the other local reactor water level indicators off the opposite reference leg and reported those levels to be indicating greater than 33 inches. This operator also stated to the inspectors that the two local Yarway water level indicators were reading between 30 and 40 inches. The Yarway level indicators, both in the control room and locally, come off totally independent reference legs than the narrow range GEMACs and the scram Barton level indicator switches.

The Browns Ferry Failure Investigation Report and operator interviews confirmed that control room operators were aware of what Yarway levels were at the time of the event. Therefore, another means of confirming what instruments were correctly indicating actual water level was available. The licensed operators stated to the inspectors that they believed the B GEMAC level indicator was reading erroneously and that the A and C GEMACs were indicating closest to what actual reactor water level was at the time prior to and following the half scram. Several operators stated that since Yarway level indicators were reading approximately 40 inches and, therefore, appeared consistent with the A and C narrow range GEMACs, then this was further evidence that the B GEMAC narrow range level was reading incorrectly. It was based on these observations and apparently incorrect conclusion that the startup continued.

After the half scram was reset, operations personnel contacted instrument maintenance. The instrument mechanics who were interviewed had no recollection of operations personnel requesting any type of investigation into the level indicator problem. However, operations personnel stated that the instrument mechanics were requested to investigate the level instruments soon after the half scram was reset. Because of this breakdown in communication, no troubleshooting or testing of the instruments occurred until the beginning of the next shift (approximately two and a half hours after the half scram). At approximately 2300, a licensed operator contacted the instrument shop to inquire about level instrument operability. By this time, all GEMAC narrow range indicators were again reading consistently. Instrument maintenance personnel informed the operator that they had not done any troubleshooting or corrective maintenance on any of the level indicators or associated switches and transmitters since their assistance was not requested. At approximately 2315, the instrument maintenance supervisor was contacted by the instrument maintenance foreman and informed of the reactor water level problem. This problem was also referred to the level instrument cognizant engineer at approximately 2330. Based on the Failure Investigation Report, the cognizant engineer stated the conclusion that the B GEMAC level was the correct instrument and that the A level constant head pot had lost part of its reference leg. By this time, however, the reference leg had

reestablished itself. A visual check of accessible reference leg piping and valves showed no observed leakage.

Following further discussions with the Operations Group Supervisor and Assistant Plant Manager for Operations, instrument mechanics performed part of SI 4.1.A-7, "Reactor Protection System Reactor Water Level," on LIS-3-203 C and LIS-3-203 D to verify that the instruments were within the required tolerance. The major concern was to determine if a full scram would have occurred if LIS-3-203 C had tripped. Both instruments were verified to be within required tolerance. A full scram apparently did not occur because the scram setpoint of LIS-3-203 C was approximately three tenths of an inch below the trip setpoint of LIS-3-203 D and this setpoint was not reached due to the operator responding promptly to raise water level in response to the half scram.

c. Conclusions

The licensee failed to correctly determine which instruments were providing correct reactor water level indication. Yarway level indications in the Control Room (LI-3-46 A, B) were observed by the operators to be reading approximately 40 inches at the time of the half scram. The Yarway instruments are calibrated for correct indication at operating temperature and pressure and normally read high during startup. These Yarway instruments come off different reference legs than both the GEMAC narrow range instruments (A, B, and C) and the scram Barton level indicator switches (LIS-3-203 A, B, C, D). At approximately 40 psig and normal water level (approximately 33 inches), Yarway levels should be indicating very high in the indicating range or still be greater than full scale of 60 inches. This effect is explained in Browns Ferry System Lesson Plan No. 3 which discusses how drywell temperature causes a zero shift due to changes in the fluid density in the reference leg. Thus the fact that Yarway level indications were reading 40 inches at 40 psig and a drywell temperature of 90°F indicated that actual vessel water level had to be significantly lower than the 37 inches as observed on the A and C narrow range GEMACs. Therefore, it appears that GEMAC B was indicating actual water level and was consistent with the Yarways due to the zero shift effect. The inspectors verified the Yarway response by calling the Browns Ferry Simulator and having the instructor input the same plant conditions. Therefore, operators should have observed the Yarway instruments to read upscale during the startup had reactor vessel level been normal.

Based on the available Yarway indications, GEMAC B narrow range level indication, local level indications, and the receipt of a half scram on low reactor water level, operators had sufficient information to conclude that A and C narrow range GEMACs were reading erroneously high. Since these instruments are on a common reference leg, it should also have been concluded that LIS-3-203 A and LIS-3-203 B were inoperable due to the erroneous high reactor water level indication.

Technical Specification 3.1 for the reactor protection system states that there shall be two operable or tripped trip systems for each function. If the minimum number of operable instrument channels per trip system cannot be met for both trip systems, the licensee shall initiate insertion of operable rods and complete insertion of all operable rods within four hours. Sufficient information existed to conclude that two of the level switches, one in each trip system, were inoperable during the startup. The licensee did not shutdown but continued power escalation after the half scram was reset. This is a violation (296/85-13-01). At the exit interview conducted on March 1, 1985, the licensee made a commitment to the NRC to investigate and determine the root cause of the faulty level indication on Unit 3 at the earliest opportunity. A potential cause of the anomaly is a leak in the reference leg inside the drywell and thus inaccessible with the unit in operation. The licensee further agreed to provide a written evaluation of the problem including the cause(s) and the corrective actions taken prior to the following startup.

During a review of records and interviews with selected personnel, the inspectors noted that a similar reactor water level instrument problem had occurred on November 20, 1984. Records indicated that the cause of the level differences may have been air trapped in the instrument sensing lines. Shift Engineer and Assistant Shift Engineer logs indicated that the B GEMAC 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). As seen in the event of February 13, 1985, water levels eventually indicated approximately the same levels without any corrective action having been taken. The licensee committed to the NRC at the exit interview conducted on March 1, 1985, to review and investigate this event, and any other analogous events. A final determination and accompanying report will be made concerning the findings and any violations of Technical Specifications 3.1 (RPS) and/or 3.2 (Protective Instrumentation). This will remain as an unresolved item (259, 260, 296/85-13-02).

The inspectors expresssed concern that the Licensee Reportable Event Determination (LRED), a written evaluation of the reportability of the event, and subsequent phone report to the NRC were not completed until 1600, February 15, 1985, approximately 43 hours after the half scram. The LRED was initiated by the Shift Technical Advisor on first shift, February 14. However, interviews with selected supervisory personnel indicated that due to a breakdown in communications, no effective action was taken on the LRED from approximately 1200, February 14, until the next morning when it was realized that no individual had taken the responsibility for ensuring the LRED was completed. All information required to complete the LRED was available immediately after the half scram was reset.

The inspectors also expressed the following observations. (1) There was an apparent lack of effective communications as exemplified by the confusion as to whether or not instrument maintenance personnel were

told to investigate the level indication problem, and also the breakdown in communications associated with processing the LRED. (2) There is an apparent training weakness among operators and Shift Technical Advisors concerning the use of available Control Room indications for evaluation of correct plant conditions with respect to reactor vessel water level. (3) The inspectors observed a reluctance by some operators on shift to feel responsible for the correct interpretation of Technical Specification situations which may require reports to the NRC. For example, the Shift Technical Advisor was tasked to initiate an LRED even though he is not licensed and may not be as familiar with Technical Specifications as licensed operators.

6. Unit 1 Control Rod 34-03 Maintenance and Testing

Control Rod 34-03 was declared inoperable at 0445 on January 26, 1985, during a startup of Unit 1. The rod would not move from position "00" when given a withdrawal signal and the cause was initially diagnosed as being seal failure. Rod 34-03 was electrically deenergized and the startup continued as allowed by the Technical Specifications. Unit 1 is at end of cycle so under normal conditions all control rods would have been withdrawn to position "48". With rod 34-03 inoperable and fully inserted, the other three symmetric rods in the group were also left fully inserted to maintain a symmetrical pattern.

On the morning of February 20, 1985, following a request by nuclear engineering to restore the rod to operability, mechanical maintenance requested that the operating shift attempt movement of Control Rod 34-03 so they could observe Control Rod Drive (CRD) flows and pressures associated with the rod. Based upon these observations, maintenance determined that directional control valve 40D for Control Rod 34-03 was possibly stuck open. A maintenance work request was initiated to replace the directional control valve. Directional control valve 40D was replaced and control rod 34-03 was declared operable at 1440 on February 20, 1985, after testing motion by withdrawing the rod to notch position "06" and reinserting the rod to notch position "00". At 1520 on February 20, 1985, Control Rod 34-03 and its three companion rods were withdrawn to position "48".

Upon subsequent review it was determined that Mechanical Maintenance Instruction (MMI)-28, Control Rod Drive Hydraulic Unit Module (Repair, Removal and Replacement), required that CRD insert and withdrawal timing tests be performed as part of the post maintenance test requirements. This testing was conducted at approximately 0930 on February 22, 1985, and resulted in a withdrawal time of 32 seconds and an insert time of 84 seconds. The timing of Rod 34-03 was adjusted and insert and withdrawal times of 53 and 41 seconds respectively were achieved. Refueling Test Instruction (RTI)-5, Control Rod Drive System, was the procedure used to time and adjust the insert and withdrawal times of Control Rod 34-03. RTI-5 has acceptance criteria of 50±10 seconds for rod insert and withdrawal speeds.

Within this area, the following examples of failure to follow procedures or inadequate procedures were identified:

a. Step 6.3 of Mechanical Maintenance Instruction (MMI)-28, Control Rod Drive Hydraulic Unit Module (Repair, Removal and Replacement), requires that functional and post maintenance testing to be done will be specified on the Maintenance Request (MR) when performing maintenance on a HCU component.

   Step 6.3 of MMI-28 further requires that the maintenance foreman will ensure that all tests that he is responsible for are performed and signed off.

   Contrary to the above, MR A126652 to replace directional control valve FSV 85-40D did not specify all functional and post maintenance testing required by MMI-28 nor did the responsible foreman ensure this testing was performed. As a result, Control Rod 34-03 was declared operable prior to completion of the post maintenance functional testing requirements of MMI-28.

b. Data Sheet 28-4 of MMI-28 which is required for documentation of completed maintenance on directional control valves was inadequate in that it did not contain blocks to record the insert and withdraw times nor did it contain a verification signoff that the required post maintenance testing had been completed.

c. MMI-28 and Operating Instruction (OI)-85, Control Rod Drive System, require that rod insert and withdrawal times be 48±3 seconds.

   Contrary to the above, on February 22, 1985, rod withdrawal and insertion times of 41 and 53 seconds respectively for Control Rod 34-03 were accepted as satisfactory. RTI-5, Control Rod Drive System, however specifies an acceptance criteria of 50±10 seconds.

d. OI-85 requires that if control rod drive pressure is increased to initially move a rod from fully inserted, then it should be returned to normal before a rod passes the 02 position.

   Contrary to the above, on February 22, 1985, rod 34-03 was withdrawn past notch position 02 with drive water pressure approximately 50 psi above normal.

   The above examples are a violation (259/85-13-03) of Technical Specification 6.3.A that requires detailed written procedures, including applicable check off lists, shall be prepared, approved and adhered to for systems and corrective maintenance operations which could have an effect on the safety of the reactor.