UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30303

Report Nos.: 50-259/84-15, 50-260/84-15, and 50-296/84-15

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 at Browns Ferry site near Decatur, Alabama

Inspectors: Ross Butcher for — 5/16/84
G. L. Pauik — Date Signed

Ross Butcher for — 5/16/84
C. A. Patterson — Date Signed

Approved by: [signature] — 5/17/84
F. S. Cantrell, Section Chief — Date Signed
Division of Reactor Projects

SUMMARY

Inspection on March 26 - April 25, 1984

Areas Inspected

This routine inspection involved 140 inspector-hours on site in the areas of operational safety, surveillance testing observation, reportable occurrences, plant physical protection, maintenance observation, regulatory performance improvement and Unit 1 drywell temperature problems.

Results

Of the seven areas inspected, there were 8 violations identified. There were five violations in the area of operational safety: one violation with two examples for misaligned system valves on the control air compressor (Unit 1) and drywell delta pressure control air compressor (Unit 2); one violation of TS 6.3.A.1 for incorrect valve checklist on the control air system; one violation for inadequate tag removal on the core spray system; one violation for inadequate control air drawings (first example); and one violation for incorrect drawing updating to the current revision in the Technical Support Center. In the area of fire protection there was one new violation and a second example of a violation noted in the operational safety area. One violation for inadequate fire protection surveillance procedures and a second example of inaccurate drawings in

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the fire protection area. There were two violations in the area of maintenance observation; one violation for failure to execute an implemented monorail system, underhung crane, forklift, mobile crane, and overhead hoists testing program; one violation for failure to have an adequate work plan or procedure for fuel rack removal from the Unit 2 fuel pool.

REPORT DETAILS

1. Persons Contacted

   Licensee Employees

   G. T. Jones, Power Plant Superintendent
   J. E. Swindell, Assistant Power Plant Superintendent
   J. R. Pittman, Assistant Power Plant Superintendent
   L. W. Jones, Quality Assurance Supervisor
   W. C. Thomison, Engineering Section Supervisor
   A. L. Clement, Radwaste Supervisor
   D. C. Mims, Engineering and Test Unit Supervisor
   J. R. Smith, Chemical Unit Supervisor
   A. L. Burnette, Operations Supervisor
   R. Hunkapillar, Operations Section Supervisor
   T. L. Chinn, Plant Compliance Supervisor
   C. G. Wages, Mechanical Maintenance Section Supervisor
   T. D. Cosby, Electrical Maintenance Section Supervisor
   R. E. Burns, Instrument Maintenance Section Supervisor
   J. H. Miller, Field Services Supervisor
   A. W. Sorrell, Supervisor, Radiation Control Unit BFN
   R. E. Jackson, Chief Public Safety
   R. Cole, QA Site Representative Office of Power

   Other licensee employees contacted included licensed reactor operators and senior reactor operators, auxiliary operators, craftsmen, technicians, public safety officers, quality assurance, quality control, and engineering personnel.

2. Management Interviews

   Management interviews are conducted on April 11 and 27, 1984, with the Power Plant Superintendent and/or Assistant Power Plant Superintendents and other members of his staff. The licensee was informed of eight violations identified during this report period. The licensee had no comment on the violations cited.

3. Licensee Action on Previous Enforcement Matters

   This area was not inspected during this period.

4. Unresolved Items

   There was one new unresolved item as noted in paragraph 11.

5. Operational Safety

The inspectors were kept informed on a daily basis of the overall plant status and any significant safety matters related to plant operations. Daily discussions were held each morning with plant management and various members of the plant operating staff.

The inspectors made frequent visits to the control rooms such that each was visited at least daily when an inspector was on site. Observations included instrument readings, setpoints and recordings; status of operating systems; status and alignments of emergency standby systems; purpose of temporary tags on equipment controls and switches; annunciators alarms; adherence to procedures; adherence to limiting conditions for operations; temporary alterations in effect; daily journals and data sheets entries; and control room manning. This inspection activity also included numerous informal discussions with operators and their supervisors.

General plant tours were conducted on at least a weekly basis. Portions of the turbine building, each reactor building and outside areas were visited. Observations included valve positions and system alignment; snubber and hanger conditions; instrument readings; housekeeping; radiation area controls; tag controls on equipment; work activities in progress; vital area controls; personnel badging, personnel search and escort; and vehicle search and escort. Information discussions were held with selected plant personnel in their functional areas during these tours. In addition a complete walkdown which included valve alignment, instrument alignment, and switch positions was performed on control air system, drywell control air system, and the fire protection system.

During a routine operational tour of the Unit 1 reactor building on April 5, 1984, the inspector noted that the drywell control air compressor return path filter station was not aligned in accordance with Operating Instruction 32-A requirements. Filter bypass valve 32-2525 was mispositioned in the open position. The shift engineer was informed and he took immediate action to correct the valving error. Failure to have the system aligned as required by plant operating instructions and drawings was identified as a violation of 10 CFR 50, Appendix B, Criterion V to the Plant Manager at the exit meeting of April 11, 1984. (259, 260/84-15-01).

During a routine operational tour of the Unit 2 reactor building on April 10, 1984, the inspector noted that the drywell delta-pressure control air compressor temperature regulating station was not correctly aligned. The temperature regulator bypass valve 2-24-876 was open vice shut as required by the system operating instruction (OI 24). The master valve check list in the control room indicated the valve was shut vice its as-found open position. Failure to have the system aligned as required by the operating instructions was identified to the Plant Superintendent as the second example for misaligned valves during this inspection. (259, 260/84-15-01).

The inspector made a followup inspection of control air system valves for all units. A random check of other control air valve positions for Unit 1 revealed that nine out of fifteen valves identified by valve identification tag numbers could not be found on the valve checklist for Operating Instruction 32 or 32A, Control Air/Drywell Control Air. This check was not a comprehensive check of the entire system and is only a partial check of the reactor building, but it is indicative of the inadequate procedure checklist. The nine valves identified for Unit 1 are as follows:

1-32-1421	Control Air to Drywell N_2 Makeup Valve FCV 76-18
1-32-1422	Control Air to Drywell N_2 Makeup Valve FCV 76-17
1-32-1423	Supply to 1-32-1421 and 1-32-1422
1-32-1424	FCV-64-18 Drywell Ventilation Supply
1-32-1425	Drain Valve
1-32-1228	Control Air to Suppression Chamber Vacuum Relief Valve FSV-64-21
1-32-2145	I lation to Panel 25-172
1-32-1336	Valve to Removed Panel
1-32-1255	Vent Damper to SBGT FSV-64-36

Examples for the other units are listed below:

2-32-1755	HPCI Control Air
3-32-2276	HPCI Control Air
3-32-2224	Containment Inerting Control Air
3-32-2225	Containment Inerting Control Air

This is a violation of Technical Specification 6.3.A.1 in that a valid checkoff list was not in the plant operating procedures as required. The Plant Superintendent was notified of this violation in an exit meeting on April 11, 1984. (259/260/296/84-15-02).

During a routine operational safety tour of the Unit 1 reactor building on April 5, 1984, the inspector noted a clearance tag (83-1260) on core spray valve FCV-75-9. A search of clearance records indicated the clearance had been cleared on August 30, 1983, and the system returned to service. The tag was not removed from the valve handwheel as required by Browns Ferry Standard Practice 14.25 (Plant Clearance Procedures). The system had been placed in service by an Assistant Shift Engineer, ASE, not licensed at Browns Ferry, but acting in that capacity for the Unit 1 refueling outage. A similar occurrence of a tag not being removed from a system upon clearing a clearance was brought to the licensee's attention in I.E. Report 83-60. In that case the tag was also cleared by a person acting in the capacity of an ASE but not licensed at Browns Ferry. The Plant Superintendent was informed that this was a violation of 10 CFR 50, Appendix B, Criterion V on April 10, 1984. (259/84-15-03).

Investigation into the plant drawings for the Control Air System in the reactor building, 47W847-9, 10, 11 for all three units, revealed that the plant drawings do not reflect the system in the plant. On Unit 1 the drawing 47W847-9 contained three examples of differences as listed below:

- Valve 1-32-1278 to PC-68-106 is not shown on the drawing.
- Isolation valve to FCV-68-106 is labeled as 1-32-1279 in the plant and is not identified as such on the drawing.
- Isolation valve to FCV-70-1 is not identified on the drawing the same as the valve identification tag 1-32-2554.

On unit two, differences exist between the plant labeled valves and the drawing as listed below:

	Plant	Drawing
Isolation to PCV 68-105	32-1278	32-2121
Isolation to PC 68-105	32-1279	32-2122
Spare	32-1894	No number
Isolation to PCV 68-106	No label	32-2133
Isolation to PC 68-106	32-1397	32-2132
Spare	32-1781	32-2139

One valve was missing from the unit three drawing between valves 2121, 2122 and 696, 2322. Valve 2133 on the drawing is not labeled in the plant.

In general the plant drawings for the control air system do not reflect the control air system piping or valve identification. Examples of errors are easily identified. The examples provided are not a comprehensive audit of the system but serve to indicate the problem. Additionally, this audit was restricted to the reactor building and other problems may exist elsewhere in the plant.

This is a violation against 10CFR50, Appendix B, Criterion V, on instructions, procedures, or drawings. The Plant Superintendent was notified of this violation at an exit meeting on April 11, 1984. (259/260/296/84-15-04).

Browns Ferry Standard Practice 2.5 requires as-constructed copies of drawings to be stamped "ADVANCE COPY" in order to notify the user to check in the advance copy file stick when the system has been changed and the as-constructed drawing has not been updated. A review of drawings in the control room for all three units, the Technical Support Center (TSC), and the shift engineer office did not indicate the proper "ADVANCE COPY" designation on 47W847-14 although advance copies were available. As-constructed drawing 47W847-9 in the TSC was not stamped "ADVANCE COPY" although an advance copy existed.

The inspector informed the licensee of the above items. The licensee had discovered a similar problem with the "ADVANCE COPY" methodology and control and was taking prompt action to correct the deficiency. Therefore, in keeping with the current enforcement policy, although this item would typically be identified as a violation, the licensee is credited with discovery and prompt action to correct the problem and this item will not be carried as a violation.

A review of drawing 47W847-10 in the Technical Support Center (TSC) indicated the File Copy in the TSC was incorrect. Revision 2 was in the TSC control drawing file vice the required revision 3. The Plant Superintendent was informed that this item was a violation of 10 CFR 50, Appendix B, Criterion VI. (259/260/296/84-15-08).

6. Surveillance Testing Observation

The inspectors observed and/or reviewed the below listed surveillance procedures. The inspection consisted of a review of the procedure for technical adequacy, conformance to Technical Specifications, verification of test instrument calibration, observation on the conduct of the test, removal from service and return to service of the system and a review of test data.

S.I. 4.11.D.1/2/3	Fire Protection Surveillance
S.I. 4.11.E.1	Fire Protection Surveillance
S.I. 4.11.A.1.g.	Reactor building hydraulic performance test.
S.I. 2	Operator daily logs

More details on the fire protection surveillances are addressed in paragraph 7.

7. Fire Protection System

During this inspection period, the inspectors reviewed the fire protection system operations to assure that regulatory requirements were being satisfied. Several deficient areas were noted as listed below:

a. The inspector reviewed Technical Specification criteria as related to Fire Protection Technical Specification (T.S.) 4.11.D. TS.4.11.D. requires a monthly walk-through by the Safety Engineer be made to visually inspect the plant fire protection systems. Browns Ferry does not have an organizational title called the "Safety Engineer" due to various organizational changes. Instead, the current title is "Safety Engineering Supervisor" for the person who fills the "Safety Engineer" job capacity. The Safety Engineering Supervisor has not been making the monthly walk-through, however, fire protection assistants to the Safety Engineering Supervisor have been conducting the inspection walk-through. Until Technical Specifications are revised, the plant manager committed to have the monthly walk-through conducted by the Safety Engineering Supervisor. This item will remain open until the Technical Specifications are revised. (259/84-15-05).

b. The inspector reviewed Technical Specification Table 3.11.A to determine procedural compliance and accuracy. Table 3.11.A lists the fire protection system hydraulic requirements. The inspector verified station 3 A/B/C for the cable tray fixed spray system. On the three stations verified, all three were listed in error, not in accordance with the design bases as identified in the system post-modification test 13-1. The surveillance flow requirements were incorrectly listed on two out of three stations checked and the residual pressure requirements were inaccurate on all three stations checked. Additionally, several cable tray fixed spray systems installed in the plant are not included in Table 3.11.A and have had no operational surveillances run on them since original installation. This item will remain open until evaluated by the licensee. (259/84-15-06).

c. The inspector reviewed the criteria used to meet the Technical Specification requirements of surveillance 4.11.A.1.g. for testing of the reactor building hydraulic performance verification. A review of records (Fire Recovery Plan, pg. 41) and post modification testing (PT-13-1) indicates that cable tray fixed spray flow requirements are satisfied only if one 1½" hose connection is used simultaneously. Therefore, the fire hose flow rate must be added in to the cable tray nozzle's flow requirements to assure Technical Specification flow requirements are met. S.I. 4.11.A.1.g was inadequate as written since the design criteria flow rate requirements were not addressed in the procedure. This is a violation of T.S. 6.3.A.6. (259/260/296/84-15-07).

d. The following discrepancies in plant drawings, pressure switch setpoints and annunciation were found in the fire protection area:

   (1) Browns Ferry Instrument Tabulation (Drawing 47B601-026, page 40) gives the setpoint of pressure switch PS-26-44 as 120 psi for the header pressure. As constructed drawing 45N644-1 gives the setting as 100 psi. Instrument and calibration data card states that the setpoint change to 100 psi was made on June 27, 1983. No basis or reason for this change can be found.

   (2) Design change request 1581 R1 dated September 23, 1978, gives the setting of pressure switch PS-26-44A as 50 psig, but a setting of 60 psi is shown on drawing 45N644-1 and 35N731-9. Additionally, two temporary alteration control forms 3-76-079 and 3-76-080 dated May 31, 1976 related to the design change request 1581 R1 are still outstanding after eight years of plant operation.

   (3) Annunciation for "Fire Protection Water Supply On" supplied from PS-26-44 was changed to "Raw Service Water Pressure Low" supplied from PS-26-44A. Logic diagram 47W611-26-13 incorrectly shows the alarm being supplied from PS-26-44. Also, the control diagram for the annunciator system, 47W610-55-2, incorrectly shows the title and pressure switch numbe for the annunciator as PS-26-44, "Fire Protection Water Supply On .

(4) The installation of PS-26-44A is not correctly reflected in plant drawings. Flow diagram 47W836-1 shows an isolation valve for PS-26-44 but no valve for PS-26-44A. A valve is installed in the system. Panel drawing 47W600-51 does not show PS-26-44A on panel 25-139.

The licensee has been requested to provide a clarification of the intended operational state of the fire protection system. Currently, one fire pump runs continuously to maintain fire header pressure adequate. The reasons for TACF 3-76-079 and 3-76-080 and DCR 1581 R1 should be reviewed to determine their current applicability to system operation. Additionally, the interaction of the annunciators to the operational scheme should be addressed. The various drawing discrepancies will be included as a second example to the violation on drawing errors. (259/260/296/84-15-04).

8. Maintenance Observation

During the report period, the inspectors observed the below listed maintenance activities for procedure adequacy, adherence to procedure, proper tagouts, adherence to Technical Specifications, radiological controls, and adherence to quality control hold points.

MMI 125 Monorail systems, underhung cranes, and overhead hoists inspection and testing.

MMI 130 Mobile cranes and forklifts inspection, testing and preventative maintenance.

Removal of Unit 2 fuel pool fuel racks.

Unit 3 Outage maintenance

MMI 122 Flush fire protection system.

During the review of Mechanical Maintenance Instruction (MMI) 125, (Inspection Testing, and Maintenance of Monorail Systems, Underhung Cranes, and Overhead Hoists) and MMI130, (Mobile Cranes and Forklifts, Inspection, Testing, and Preventative Maintenance), the inspector noted several deficiencies as listed below:

a. MMI 125 requires a periodic inspection of monorail systems, underhung cranes, and hand chain-powered overhead hoists to be conducted on idle (over six months) equipment. No evidence was available for review to indicate this inspection was being scheduled or completed as required.

b. MMI 125, Appendix 2, requires frequent (not defined) inspections be conducted on hand-powered overhead hoists. The hooks are to be checked to ascertain the hook throat opening was not more than 15% greater than normal throat opening. The procedure did not specify the normal throat opening and no evidence the inspection had even been conducted was available for review. Several mechanical engineers/technicians interviewed did not know what the normal throat opening would be for various size hooks. The procedure specifically deleted any data sheet requirements.

c. MMI 130 requires wire rope inspections to include a check for proper rope reeving. The reeving of individual cranes was not listed in the procedure or known by mechanical craft personnel.

d. MMI 130 data sheet 7, monthly wire rope inspection, is inconclusive on required signoffs (one yes/no signoff for two determinants, step 1.c.) and does not address a signoff for each requirement in the procedure text (No signoff to verify rope reeving).

The above listed examples are not meant to be an all inconclusive list of procedural inadequacies but only an indication of a generic problem with deficient procedures in this area. The Plant Superintendent was informed of the violation of 10 CFR 50, Appendix B, Criterion X for failure to execute MMI 125/130 testing program at the exit on April 10, 1984. (259, 260, 296/84-15-09).

On April 24, 1984, the inspector observed the removal of old fuel racks from the Unit 2 fuel pool. The racks were being removed in order to install the new high density fuel racks. The inspector requested to review the work plan or procedure used to carry out this major maintenance task. No work plan had been written to cover this specific task. The work was being performed under Work Plan 6941 which was written to install the new high density fuel racks. Typically in the past, one work plan has been written to remove the old racks and one work plan written to install the new racks. Browns Ferry Standard Practice 8.3 requires that work plans be written to address plant modifications. W.P. 6941 did not adequately address the removal of the old racks. The Plant Superintendent was informed that this item was a violation of 10 CFR 50, Appendix B, Criterion V. (260/84-15-10).

9. Reportable Occurrences

The below listed Licensee Event Reports (LERs) were reviewed to determine if the information provided met NRC requirements. The determination included: adequacy of event description, verification of compliance with Technical Specifications and regulatory requirements, corrective action taken, existence of potential generic problems, reporting requirements satisfied, and the relative safety significance of each event. Additional inplant reviews and discussion with plant personnel as appropriate were conducted for those reports indicated by an asterisk. The following licensee event reports are closed:

LER No.	Date	Event
260/82-28 R3	9-21-81	Breach of secondary containment.

260/82-33	1-15-82	Purge system charcoal efficiency below specification.
*260/83-74 R2	11-10-83	HPCI isolation due to turbine exhaust rupture disc failure.
*260/83-80 R1	12-14-83	SDIV level instrument slow response time.
*260/83-83	12-29-83	Hydrogen analyzer "B" inoperable.
260/83-84	12-29-83	EECW pump 'A3' pump suction restricted by debris.
*260/84-01	1-08-84	RCIC rated flowrate lower than specified.
*260/84-02	1-21-84	Scram due to operator error during testing.
*260/84-03	2-12-84	HPIC inoperable due to turbine governor control system failure.
*260/84-04	2-22-84	Scram due to high flux on IRM.
296/80-14	5-12-80	RCIC inoperable.
*296/82-43 R1	9-09-82	3DN LPCI MG set inoperable.
*296/82-56 R1	11-28-82	Drywell CAM inoperable.
*296/82-61	12-03-82	Drywell CAM inoperable.
*296/83-18 R1	2-28-83	SDIV level switch inoperable.
296/83-43	7-27-83	CAM 3-90-256 for drywell out-of-service.
*296/83-44	7-31-83	Control rod 54-47 inoperable.

*296/83-45	8-02-83	Hydrogen analyzer for drywell and torus inoperable.
296/83-46	8-01-83	Torus temperature indicator 64-55A inoperable.
*296/83-47	7-30-83	RHR pump seal cooler clogged.
*296/83-59	12-08-83	IGSCC in RHR head spray pipe weld.
*296/84-02	1-25-84	Secondary containment isolation of Unit 3 reactor and refuel zones.
*296/84-03	1-29-84	Loss of voltage of 4-K.V. shutdown boards 3EC/3ED.
*296/84-04	2-28-84	RHR outboard Loop II isolation valve 74-67 broken stem.

No violations or deviations were identified in this area.

10. Regulatory Performance Improvement Program (RPIP)

As part of the regional oversite of the RPIP, the responsible project section chief reviewed the status of the RPIP and the minutes of meetings February 6, 1984, February 21, 1984 and March 6,1984 of the RPIP Oversite Review Committee and attended the RPIP Meeting on April 23, 1984. The reported progress had met milestones as described in the RPIP. Attendance at the meetings appears to be at the proper level to ensure that the required resources are applied to ensure that goals continue to be met.

A review of the operating experience of key personnel was conducted at the plant and in the Division of Nuclear Power. The review disclosed that individuals met required qualifications; however, experience tended to be more in startup testing, maintenance, and outage work, with a minimum of actual operating experience (first and second level supervision of licensed reactor operator and senior reactor operator, and reactor engineers) in key positions.

Discussions were held with key plant personnel on the above subjects.

11. Unit One Dyrwell Cooling Problems

Unit one drywell atmosphere cooling system has had a history of problems identified in the performance of Refueling Test Instruction 72 (RTI-72), Drywell Atmosphere Cooling System. Most recently, at the beginning of cycle five, 55-100% plateau, there were the following exceptions for the test conducted December 12, 1981:

Exception #4	The maximum temperature over the bulkhead of 200°F. was exceeded by the point measured by TE-80-27 at 265°F.
Exception #5	The maximum temperature difference between adjacent points below the bulkhead of 25°F was exceeded by TE-80-24. A difference of 60°F was recorded.

For the beginning of cycle 6 55-100% plateau test conducted March 24, 1984, the following exceptions were listed:

Exception #13	The measured heatload of 5.748 MBTU/HR. exceeded the design capacity of 5.19 MBTU/HR.
Exception #14	The design temperature difference of 25°F. between adjacent points below the bulkhead assembly was exceeded by TE-80-22; resulting in 30°F. temperature difference.
Exception #15	The design maximum temperature of 200°F. above the bulkhead was exceeded by TE-80-27 at greater than 300°F.
Exception #16	The design maximum relative humidity of 50% was exceeded. Actual reading was 52%.

Unit 1 initial criticality date for cycle 6 was December 29, 1983, and has been operating near full load most of the time since initial criticality. However, the data for RTI-72 was not taken until March 24, 1984, almost three months later. Further steps in the procedure are marked as either criticality (C) or non-critical (NC). A critical step is one which confirms proper operation of a system necessary to plant safety or which confirms any assumptions made in the safety analysis report. For the test plateau from heatup to 55% power step 38 for recording drywell temperatures is marked critical, but for the 55-100% power test plateau, step 56 for recording drywell temperatures is marked non-critical. The FSAR uses an initial drywell temperature of 135°F. for its LOCA analysis. One set of data taken for RTI-72 on March 25, 1984 indicated a drywell temperature 133.2°F.

Due to the fact that RTI-72 for Unit 1 has never been completed without numerous exceptions and that serious questions about the adequacy of the drywell atmospheric cooling system have never been answered, this is being left as an unresolved item pending licensee resolution and further inspection. (259/84-15-11).

The licensee has been aware of the drywell cooling problems for a number of years, but no positive actions have been completed to date to rectify the concerns.

12. Plant Physical Protection

During the course of routine inspection activities, the inspectors made observations of certain plant physical protection activities. These included personnel badging, personnel search and escort, vehicle search and escort, communications and vital area access control.

No violations or deviations were identified within the areas inspected.