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

Report Nos.: 50-259/84-45, 50-260/84-45 and 50-296/84-45

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 Units 1, 2, and 3

Inspection Conducted: October 22 - November 1, 1984

Inspectors: L. W. Garner For — 2-8-85
G. L. Paulk, Senior Resident Inspector — Date Signed

L. W. Garner For — 2-8-85
C. A. Patterson, Resident Inspector — Date Signed

[signature] for — 2-8-85
W. K. Poertner, Regional Inspector — Date Signed

C. Julian for — 2/8/85
P. D. Wagner, Regional Inspector — Date Signed

Approved by: [signature] — 2/8/85
F. S. Cantrell, Section Chief — Date Signed
Division of Reactor Projects

C. Julian — 2/8/85
C. A. Julian, Section Chief, — Date Signed
Division of Reactor Safety

SUMMARY

Scope: This special inspection involved 170 inspector-hours in the area of Unit 3 startup activities on October 22, 1984.

Results: 1. 84-45-01, Violation of Technical Specification 6.3.A.1.

a. Failure to put floor drain sump level transmitter 3-LT-77-1A in service prior to reactor operation.

b. Failure to lock drywell equipment hatch trolley cranks per Procedure BF GOI 100-1.

c. Failure to complete test Plateau I of the MRTI Procedure prior to going to test Plateau II.

d. Failure to complete MRTI Procedure Steps 28 and 29 prior to performance of MRTI Procedure Step 30.

e. Failure to complete Section II.A of Procedure BF GOI 100-1 prior to taking the reactor critical.

f. Failure to attach a graph of Keff to S.I 4.3.B.1.a per Procedure BF GOI 100-1.

g. Procedure SI 4.6.E inadequate in that acceptance criteria would permit jet pump differential pressures in excess of Technical Specification requirements.

h. Failure to note and explain any exceptions for jet pump operability surveillance per Procedure SI 4.6.E.1.

2. 84-45-02, Violation of Technical Specification 3.3.B.3/4.3.B.3

   a. Violation of Technical Specification 3.3.B.3.c in that:

      (1) The RWM was not operable due to errors in control rod sequence input.

      (2) Failure to station a second operator when the RWM was inoperable.

   b. Violation of Technical Specification 4.3.B.3.c for failure to adequately verify the correctness of the RWM computer input.

   c. Violation of Technical Specification 3.3.B.3.d for failure to take the reactor to a shutdown condition when the RWM was inoperable.

3. 84-45-03, Violation of Technical Specification 3.6.E.1 for failure to have all jet pumps operable when in the startup mode.

REPORT DETAILS

1. Licensee Employees Contacted

   J. A. Coffey, Site Director
   G. T. Jones, Plant Manager
   J. E. Swindell, Superintendent - Operations/Engineering
   J. R. Pittman, Superintendent - Maintenance
   J. H. Rinne, Modifications Manager
   J. D. Carlson, Quality Engineering Supervisor
   D. C. Mims, Engineering Group Supervisor
   R. Hunkapillar, Operations Group Supervisor
   C. G. Wages, Mechanical Maintenance Supervisor
   T. D. Cosby, Electrical Maintenance Supervisor
   R. E. Burns, Instrument Maintenance Supervisor
   A. W. Sorrell, Health Physics Supervisor
   R. E. Jackson, Chief Public Safety
   T. L. Chinn, Technical Services Manager
   T. F. Ziegler, Site Services Manager
   J. R. Clark, Chemical Unit Supervisor
   B. C. Morris, Plant Compliance Supervisor
   A. L. Burnette, Assistant Operations Group Supervisor
   R. R. Smallwood, Assistant Operations Group Supervisor
   T. W. Jordan, Assistant Operations Group Supervisor
   S. R. Maehr, Planning/Scheduling Supervisor
   C. R. Hall, Design Services Manager
   W. C. Thomison, Engineering Section Supervisor
   A. L. Clement, Radwaste Group Controller

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

2. Exit Interview

   The inspection scope and findings were summarized on November 1, 1984, with the Plant Manager and/or Assistant Plant Managers and other members of his staff.

   The licensee acknowledged the findings and took no exceptions.

3. This special report covers the Nuclear Regulatory Commission's concerns during the startup of the Unit 3 reactor on October 22, 1984 after completion of a refueling outage which lasted over four hundred days.

   An Enforcement Conference was held at the Browns Ferry site on November 7, 1984 (See Inspection Report 50-259/84-46, 50-260/84-46 and 50-296/84-46).

a. Unit 3 Startup on October 22, 1984

The inspector toured the Unit 3 control room on the morning of October 22, 1984, after the unit had been taken critical to determine the shutdown margin. The unit had been taken critical using a "B" rod sequence since the most reactive rod in the core was a "B" rod. After completion of the shutdown margin determination the reactor was again to be brought critical using an "A" rod sequence since the unit was loaded with a controlled cell core configuration which requires the "A" sequence. The inspector noted that the unit was in shutdown cooling. A review of records indicated the reactor had been started in a single loop mode.

During a tour of the Unit 3 reactor building on October 22, 1984, the inspector noted that the LT-77-1A power On/Off toggle switch was in the "OFF" position. The instrument checklist for the drywell floor drain sump level transmitter LT-77-1A is contained in plant operating instruction 77. A review of the OI-77-1A instrument checklist indicated that a lineup was conducted on the drywell floor drain sump level transmitter LT-77-1A between February 6, 1984 and April 10, 1984. The inspector could not ascertain which unit had been lined up since the data sheet was for common (Units 1, 2 and 3) equipment; however, each unit has its own separate drywell floor drain sump system. The OI-77 procedure did not adequately specify which unit alignment had been verified. Also, the drywell shield plug trolley chain was not padlocked as required by procedure BF GOI 100-1. Technical Specification 6.3.A.1 requires that detailed written procedures be prepared, approved and adhered to. Failure to have the drywell floor drain sump level transmitter 3-FT-77-1A in service is a violation of Technical Specification 6.3.A.1 (50-296/84-45-01). Failure to have the drywell shield plug trolley chain padlocked as required by BF GOI 100-1 is a second example of violation 84-45-01. The inspector noted numerous people in the reactor building around the residual heat removal piping used for shutdown cooling. The inspector returned to the control room and discussed his concerns with the shift engineer. Of particular concern was the fact that activated reactor coolant had been circulated outside primary containment through unshielded piping and the possible exposure concern to personnel in the reactor building from nitrogen-16 decay. See paragraph 3.f for further discussion.

b. Sequence of Events (From operators log)

October 21, 1984

2025	Mode Switch to Startup
2100	"A" Recirculation Pump In Service
2215	"B" Recirculation Pump Out of Service
2230	Loop I RHR In Service - Shutdown Cooling

October 22, 1984

0600	Pulling Rods For Shutdown Margin Test
0635	Pulling Group II Rods
0640	RWM Problems
0730	Resolved Sequence Problems
0752	Resumed Pulling Rods
0844	Reactor Critical
0900	Back Subcritical
0910	Pushing Rods
1100	All Rods Fully Inserted
1140	Shutdown Cooling Secured
1415	Computer Inoperable

October 23, 1984

0900	Computer Operable

October 24, 1984

October 25, 1984

1540	Mode Switch To Refuel
2145	Mode Switch To Shutdown

c. Event Description

The root cause of the startup problem stems from not following the plant procedures. A Master Refueling Test Instruction (MRTI) coordinates unit operational and test activities following a refueling outage. The three test plateaus specified in the procedure are as follows:

Plateau I	Open Vessel
Plateau II	Initial heatup to 55% of rated power
Plateau III	55-100% of rated power

The following outline summarizes certain steps in the procedure:

Plateau I	Step 27	Authorized to go to Plateau II
Plateau II	Step 28	GOI-100-1 Sections I.A and II.A complete
Plateau II	Step 29	Plant Superintendent permission to go critical
Plateau II	Step 30	Perform RTI-4 per GOI-100-1 Section II.B and C.

MRTI Steps 27, 28, 29 and 30 were not signed as being completed. MRTI Step 28 requires that GOI-100 Section I.A., Pre-Startup Checklist, and Section II.A, Preparation for Approach to Criticality be complete.

GOI-100 Section II.A, step 9 secures shutdown cooling, step 10 starts the recirculation pumps, and step 14 secures head vents.

MRTI Steps 27, 28, 29 and 30 are identified in the procedure as critical steps. A critical step is defined as one which confirms proper operation of a system necessary to plant safety or which confirms any assumptions made in the safety analysis report, or one that must be completed prior to proceeding to the next test plateau.

Had these steps been completed prior to performing Refueling Test Instruction 4 (RTI-4), Full Core Shutdown Margin-Closed Vessel, the shutdown cooling system would have been secured and the residual heat removal (RHR) system would have been operable. Failure to follow plant procedures was identified as a violation of Technical Specification 6.3.A. Failure to complete step 27 of test plateau I prior to proceeding to test plateau II is a third example of violation 50-296/84-45-01. Also, failure to complete MRTI critical steps 28 and 29 prior to the performance of step 30 is a fourth example of violation 50-296/84-45-01. Further, BF GOI 100-1, Cold Startup Preparation for approach to Critical, requires that section II.A be completed prior to taking the reactor critical. The failure to complete steps II.A.9, II.A.10 and II.A. 14, which was another check prior to criticality, is the fifth example of violation 50-296/84-45-01. BF/GOI/100-1, Pre-startup Checklist step I.R.2 requires a graph of Keff, as a function of rods withdrawn, be attached to S.I.4.3.B.1.a data sheet. The graph of Keff was not attached to S.I.4.3.B.1.a data sheet dated October 22, 1984. This is the sixth example of violation 50-296/84-45-01. The Plant Manager was informed of these violations at the exit meeting.

d. Rod Worth Minimizer (RWM) Problems

Discussions with plant personnel revealed that during the reactor startup numerous changes were made to the rod pull sheets and the RWM was bypassed in order to correct the errors in the computer program. Thirty-one rods were withdrawn when selection of the next rod revealed the rod was not in the correct group. At this time it was discovered that several errors had occurred during the preparation of the rod pull sheets.

Technical Specification 3.3.B.3.c requires that the RWM be operable whenever the reactor is in the startup or run mode or a second licensed operator must be stationed at the reactor console to verify compliance with the control rod program. The RWM was not operable due to the computer program errors. This is a violation of Technical Specification 3.3.B.3.c (50-296/84-45-02). A second licensed operator was not stationed while the RWM was bypassed after identification of the errors.

Further it was subsequently discovered during a review of plant logs that the mode switch remained in "STARTUP" from 8:25 p.m., on October 21, 1984 to 3:40 p.m., on October 25, 1984 with the RWM inoperable from 2:15 p.m., on October 22, 1984 to 9:00 a.m., on October 23, 1984, due to a computer malfunction. Failure to post a second licensed operator when the RWM was inoperable is the second part of the first example of violation 50-296/84-45-02.

The errors in the RWM program resulted from loading an incorrect withdrawal sequence into the RWM computer. The correct withdrawal sequence was listed in Table 4.1.B of RTI-4. Technical Specification 4.3.B.3.c requires the correct sequence be verified before reactor startup. This is a violation for the reactor startup conducted October 22, 1984 with errors in the RWM computer due to an inadequate verification of the correct program. These errors were not detected until after 31 control rods had been withdrawn. This is the second example of violation 50-296/84-45-02.

Furthermore, once these errors were detected the reactor was not shutdown immediately. Technical Specification 3.3.B.3.d states that if Specifications 3.3.B.3.a through 3.3.B.3.c cannot be met the reactor shall not be started; or if the reactor is in the run or startup modes at less than 20% rated power, it shall be brought to a shutdown condition immediately. The reactor remained in the startup mode while the correct withdrawal sequence was input into the computer and then the approach to criticality was continued. This is the third example of violation 50-296/84-45-02.

These violations were discussed with the plant manager in the exit meeting.

e. Jet Pump Operability

On October 22, 1984, Surveillance Instruction 4.6.E.1 (Jet Pump Operability) was conducted on Unit 3. During the conduct of the surveillance the operator identified several suspect jet pump delta-pressure readings. Further investigation revealed that jet pump flow transmitter 3-FT-68-40 was incorrectly valved out and transmitter 3-FT-68-19 had the equalizer valve open. The equalizer valve being open would affect eight jet pump output instruments on the "B" recirculation loop. FT-68-40 is on the "A" recirculation loop and affected only one of the ten pumps in that loop. The surveillance had been conducted the previous day but the incorrect jet pump readings were incorrectly attributed to low flow conditions. During the surveillance process the shift technical advisor is required to evaluate the test results in accordance with Technical Instruction 52 (Jet Pump Operability). The evaluation conducted on October 21, 1984, was inadequate in that it did not note that several jet pumps were outside the 10% tolerance band with no electronic noise present on the delta-pressure instruments. Surveillance Instruction (S.I.) 4.6.E, Jet Pumps, is inadequate in that it does not fulfill the requirements of Technical Specification 4.6.E. Technical Specifications require that individual jet pump differential pressures be within 10% of the mean of all jet pump differential pressures when certain conditions exist. S.I. 4.6.E which states that completion of Section S.I. 4.6.E-1 fulfills the requirements of T.S. 4.6.E; however, provides a different acceptance criteria which is given in Technical Instruction (TI) 52. TI-52 requires that individual jet pump differential pressures be within 10% of its established baseline data (which is an historical percent deviation from the mean of all jet pump differential pressures). The TI-52 criterion would actually allow individual jet pump differential pressures in excess of 10% of the mean of all jet pump differential pressures. (Figure 12 of TI-52 shows that up to 15% deviation from the mean would be acceptable.) This is the seventh example of violation 50-296/84-45-01.

Step 20 of S.I. 4.6.E.1 for Jet Pump Operability requires that the results comply with TI-52 with any exceptions noted and explained in the Remarks Section. S.I. 4.6.E.1 was performed on Unit 3 on October 21, 1984 and the fact that step 6.7 (deviation from jet pump baseline by more than 10%) of TI-52 did not pass the acceptance criteria was not noted or explained in the remarks section of S.I. 4.6.E.1. This is the eighth example of violation 50-296/84-45-01.

Technical Specification 3.6.E.1 requires that all jet pumps be operable whenever the reactor is in the startup mode. The reactor was taken critical on October 22, 1984, without having all jet pumps fully operable. This is a violation (50-296/84-45-03).

The plant manager was informed of these violations at the exit meeting.

f. Radiological Hazards During Event:

The radiation levels around the unshielded RHR piping in the reactor building were estimated by the health physics staff based on a power level of 0.5%. The RHR pump corner room, containing the "A" and "C" RHR pumps, contains a radiation monitor mounted on the wall. This monitor was found to have increased by 5 mr/hr. Estimates of the piping on the 519' elevation were 20 mr/hr on contact and 541' elevation at 30 mr/hr on contact. The radiation levels would have been directly proportional to power level and would have increased by a factor of 200 for 100% power. The principle source of radiation in these estimates was from the short-lived decay gammas of nitrogen-16 from the activation of reactor coolant. The radiation levels posed no undue threat to the public health and safety.