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

Report Nos.: 50-259/93-18, 50-260/93-18, and 50-296/93-18

Licensee: Tennessee Valley Authority
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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 at Browns Ferry Site near Decatur, Alabama

Inspection Conducted: April 17 - May 14, 1993

Inspector: Linda J Watson for 6/11/93
C. A. Patterson, Senior Resident Inspector Date Signed

Accompanied by: J. Munday, Resident Inspector
R. Musser, Resident Inspector
G. Schnebli, Resident Inspector
T. Liu, Intern

Approved by: Linda J Watson for 6/11/93
Paul Kellogg, Chief, Date Signed
Reactor Projects, Section 4A
Division of Reactor Projects

SUMMARY

Scope: This routine resident inspection included surveillance observation, maintenance observation, operational safety verification, verification of Unit 2 cycle 6 outage commitments, and review of licensee self assessment, Unit 3 restart activities, reportable occurrences, and action on previous inspection findings.

One hour of backshift coverage was routinely worked during the work week. Deep backshift inspections were conducted on May 2 and May 9, 1993.

Results: During the end of this report period the licensee was nearing completion of a refueling outage, paragraph six. The licensee completed several major modifications to complete all TMI action items for Unit 2, upgraded the fire protection system to meet National Fire Protection Association standards, and completed other plant modifications to upgrade plant systems totaling over 200 design changes.

One unresolved item was identified concerning loss of primary pressure control during the combined performance of an instrument line excess flow check valve test and inservice vessel leak check, paragraph three. The primary pressure control gauge used during the leak check was taken out of service without knowledge of the plant operators. This event was reviewed by a special NRC human performance evaluation conducted May 13-14, 1993.

One noncited violation was identified for fuel movement errors during the core reload, paragraph four. This item was promptly identified by the licensee and corrective action taken. Core mapping verification revealed no other problems.

REPORT DETAILS

1. Persons Contacted

Licensee Employees:

*O. Zeringue, Vice President
*J. Scalice, Plant Manager
J. Rupert, Engineering and Modifications Manager
*R. Baron, Quality and Licensing Manager
D. Nye, Recovery Manager
*M. Herrell, Operations Manager
J. Maddox, Engineering Manager
*M. Bajestani, Technical Support Manager
A. Sorrell, Special Programs Manager
C. Crane, Maintenance Manager
*P. Salas, Licensing Manager
*R. Wells, Compliance Manager
J. Corey, Radiological Control Manager
J. Brazell, Acting Site Security Manager

Other licensee employees or contractors contacted included licensed reactor operators, auxiliary operators, craftsmen, technicians, and public safety officers; and quality assurance, design, and engineering personnel.

NRC Personnel:

P. Kellogg, Section Chief
*C. Patterson, Senior Resident Inspector
*J. Munday, Resident Inspector
R. Musser, Resident Inspector
*G. Schnebli, Resident Inspector
*T. Liu, Intern

*Attended exit interview

Acronyms and initialisms used throughout this report are listed in the last paragraph.

2. Surveillance Observation (61726)

The inspectors observed and/or reviewed the performance of required SIs. The inspections included reviews of the SIs for technical adequacy and conformance to TS, verification of test instrument calibration, observations of the conduct of testing, confirmation of proper removal from service and return to service of systems, and reviews of test data. The inspectors also verified that LCOs were met, testing was accomplished by qualified personnel, and the SIs were completed within the required frequency. The following SIs were reviewed during this reporting period:

a. 2-SI-4.4.A.2, Standby Liquid Control System Functional Test

On April 20, 1993, the inspectors observed the performance of portions of 2-SI-4.4.A.2, Standby Liquid Control System Functional Test. This system functional test is performed each operating cycle to determine pump capacity. The specific evolution observed by the inspectors was the 2B pump capacity test. During this particular test, the 2B pump was lined up to take suction from and discharge to the recirculation test tank. The capacity of the pump is monitored using an ultrasonic flowmeter mounted to a portion of the pump discharge line. The pump did not produce the required flow rate to satisfy the ASME Section XI criterion to test for vibration, and therefore the surveillance was stopped and a test deficiency was generated.

On April 22, 1993, the flowrate test was re-performed on the 2B pump. The inspector observed a portion of this SI. A flowrate of 48.9 GPM was observed and recorded with a pump vibration of .0994 inches/second. TS 4.4.A.b requires a minimum SLC pump flow rate of 39 gpm, however, 2-SI-4.4.A.2, Section 6.1.4, requires the 2B pump flow rate to be between 49.4 and 56.5 gpm. Following maintenance on the pump, 2-SI-3.1.14, SLC System Baseline Data Evaluation, was performed to obtain the new pump baseline reference values. The cognizant ASME Section XI engineer and the system engineer evaluated the test results of the flow rate and vibration values, and the results were found acceptable for use as the new reference baseline values. Other changes made to the SLC system during this inspection period included replacement of the 2-FCV-63-8A squib valve because it fired during initial performance of 2-SI-4.4.A.2, and replacement of the 2-FCV-63-8B squib valve assembly because its 5 year service life was approaching expiration.

On April 27, 1993, an LCO was initiated after the licensee discovered that the 2-FCV-63-8B squib valve continuity meter indicated over 10 milliamps. 2-SI-4.4.A.2, Step 7.13.34.3, requires the continuity meter to indicate between 3 and 7 milliamps. Following replacement of the 2-FCV-63-8B squib valve trigger assembly, the continuity meter indication returned to 6 milliamps, which was within the acceptable range, and the LCO was canceled.

The inspector had a concern over the orientation of the SLC squib valve spool piece after conducting a walkdown of the system on April 27, 1993. The spool pieces are flared on one side. The two spool pieces were installed with different orientation of the flared end. The inspector notified the Technical Support Manager of this concern. It was concluded that the flared ends do not affect the flow through the spool places. The limiting flow points are controlled by the squib valves which have inner diameters of 13/16 of an inch.

The inspector performed Unit 2 SLC system walkdown with the corresponding flow diagram, 2-47E854-1, Flow Diagram Standby Liquid Control System, on May 5, 1993. Each individual SLC system was found to be aligned in accordance with the drawing. The inspector found no deficiencies.

b. 2-SI-4.7.D.1.d.1, Marotta Excess Flow Check Valve Testing

During performance of the SI to test the instrument line excess flow check valve operability on May 11, 1993, a high reactor pressure condition resulted in an ATWS/ARI/RPT trip of both recirculation pumps, depressurization of the scram pilot air header and a subsequent scram condition. This SI was being performed in conjunction with the reactor pressure vessel system leak test, 2-SI-3.3.1.A, ASME Section XI System Leakage Test of the Reactor Pressure Vessel and Associated Piping (ASME Section III, Class 1). The primary reactor pressure instrument being used to control reactor pressure for the leak test was unknowingly valved out of service as required by the Marotta valve testing. Plant operators observed an indicated decrease in reactor pressure and decreased reactor vessel water letdown from the vessel to maintain the required indicated pressure band. This resulted in an actual high pressure signal to the ATWS/ARI/RPT system at 11:20 p.m., on May 11, 1993. This caused a depressurization of the scram pilot air header by the ATWS/ARI/RPT system and RPS actuation causing the control rods to travel from their full-in position to the full-in/overtravel position. No safety relief valves operated during the event.

The licensee made a 4-hour ENS notification to the NRC at 3:19 a.m., on May 12, 1993. The licensee initiated an incident investigation of this event. The NRC conducted a four man team human performance evaluation on May 13-14, 1993. This issue will remain unresolved pending further evaluation of operator performance, procedural adequacy, and equipment performance. This will be tracked as URI 260/93-18-01, Loss of Primary Pressure Control.

No violations or deviations were identified in the Surveillance Observation area.

3. Maintenance Observation (62703)

Plant maintenance activities were observed and/or reviewed for selected safety-related systems and components to ascertain that they were conducted in accordance with requirements. The following items were considered during these reviews: LCOs maintained, use of approved procedures, functional testing and/or calibrations were performed prior to returning components or systems to service, QC records maintained,

activities accomplished by qualified personnel, use of properly certified parts and materials, proper use of clearance procedures, and implementation of radiological controls as required.

Work documentation (MR, WR, and WO) were reviewed to determine the status of outstanding jobs and to assure that priority was assigned to safety-related equipment maintenance which might affect plant safety. The inspectors observed the following maintenance activities during this reporting period:

a. Core Spray Flow Transmitter

During the outage, Core Spray System II flow transmitter, 2-FT-75-49, was replaced under DCN W17439 and WP W19558. On April 15, 1993, the inspector observed maintenance personnel during the performance of WO 93-00252-00. This WO was written to perform a loop calibration on Core Spray System II flow transmitter, 2-FT-75-49 using LCI-2-F-75-49, Core Spray System II Flow. This work would also satisfy the PMT requirements of WP W19558. When the technicians applied input pressure the indicated flow decreased rather than increased as expected. The technicians stopped the surveillance and determined that the transmitter had been plumbed backwards. The high side sensing line was connected to the low side of the transmitter and the low side sensing line was connected to the high side of the transmitter. The technicians backed out of the procedure and turned their findings over to their supervision. This transmitter replaced an older model that had the high and low sensing lines opposite of the new transmitter. The DCN that installed the new transmitter, W17439, did not identify this difference and resulted in the new transmitter being plumbed backwards. A field change was made to the DCN and a new WP, 2508-93, was written that reversed the sensing lines. On April 20, 1993, maintenance re-performed the calibration with satisfactory results.

b. Main Steam Relief Valves

The inspector monitored maintenance activities associated with the MSRVs during the Unit 2 Cycle 6 refueling outage. These activities included testing of the MSRV pilot assemblies, refurbishment of the MSRV solenoid valves, and tear down and inspection of one of the MSRVs. While the as-found lift tests indicated that nine pilot assemblies did not lift within required tolerance and two did not lift at all, the performance is not inconsistent with that of other utilities, as described in IR 93-12. The resident inspectors will review the manual lift tests which will be performed during startup with the reactor pressure at 250 pounds to confirm operability of the valves.

No violations or deviations were identified in the Maintenance Observation area.

4. Operational Safety Verification (71707)

The NRC inspectors followed the overall plant status and any significant safety matters related to plant operations. Daily discussions were held with plant management and various members of the plant operating staff. The inspectors made routine visits to the control rooms. Inspection observations included instrument readings, setpoints and recordings, status of operating systems, status and alignments of emergency standby systems, verification of onsite and offsite power supplies, emergency power sources available for automatic operation, the purpose of temporary tags on equipment controls and switches, annunciator alarm status, adherence to procedures, adherence to LCOs, nuclear instruments operability, temporary alterations in effect, daily journals and logs, stack monitor recorder traces, and control room manning. This inspection activity also included numerous informal discussions with operators and supervisors.

General plant tours were conducted. Portions of the turbine buildings, each reactor building, and general plant areas were visited. Observations included valve position and system alignment, snubber and hanger conditions, containment isolation alignments, instrument readings, housekeeping, power supply and breaker alignments, radiation and contaminated area controls, tag controls on equipment, work activities in progress, and radiological protection controls. Informal discussions were held with selected plant personnel in their functional areas during these tours.

a. Secondary Containment Door Interlocks

On April 27, 1993, at approximately 7:00 a.m., with core reload in progress, the inspector observed the two personnel airlock doors between the Unit 2 reactor building and the turbine building open simultaneously. The doors were observed to be open only momentarily and then were closed by personnel in the airlock. Having the two doors open at the same time constitutes a loss of secondary containment integrity as defined by TS 1.0/P.1.a and a failure of the secondary containment door interlocks. The inspector brought this matter to the attention of the Unit 2 ASOS and the Plant Manager. The licensee stopped passage through the personnel airlock so that operations and maintenance personnel could perform troubleshooting of the interlocks. No problems were detected. The licensee stated that if personnel were attempting to enter the airlock at precisely the same time, interlock failure could occur and both doors would open.

On April 29, 1993, at approximately 2:00 a.m., an operations AUO observed another failure of the interlocks to prevent a simultaneous opening of the turbine building and reactor building airlock doors. The licensee again secured the personnel airlock. The inspectors reviewed the maintenance history for the airlock and found that problems were occurring on roughly a monthly basis. Plant technical support performed an evaluation to determine if

further improvements could be made to the personnel door access interlock system. The licensee intends to take steps to improve reliability of the interlock system by replacing the proximity switches currently installed on the doors with a switch which is activated when a magnet is energized and a metal plate is pressing against the magnet. In addition, push button switches would be added for the turbine building door which would ensure that the magnet on this door is energized (thus keeping the door closed) unless the button is depressed and the two reactor building doors (Units 1 and 2) are closed. The licensee also intends to include instruction on secondary containment and the importance of proper operation of the reactor building to turbine building airlock in General Employee Training. The inspector has completed a review of the circumstances of these failures and the licensee's corrective actions and considers them acceptable. Further due to the relative infrequency and short duration of these failures these events are considered to be of relatively minor safety significance. The inspectors are continuing to monitor the licensee's corrective actions to prevent future momentary losses of secondary containment integrity.

b. Electromagnetic Interference Mapping On Refuel Floor

Special Test, 0-ST-93-01, Electromagnetic Interference Mapping, was conducted during this inspection period. This test was to perform phase one of the EMI mapping of Units 1, 2, and 3 refuel floor in proximity to the reactor and refuel zone radiation monitors. This test was performed to fulfill the requirements of proposed TS amendment 316.

The refuel floor environment was setup to simulate the normal plant operating conditions including the operation of the refueling floor overhead crane and the use of hand held radios. Current probes were clamped over power and signal cables to monitor the levels of EMI with their interfacing plant equipment in operation. Oscilloscope probes were attached to selected terminal points to record data. Three types of antennae, active rod, biconical, log periodic, with radiated emission frequency range from 14 KHZ to 1 GHZ were positioned at various locations and rotated to map the EMI profile present in each area surveyed while the plant was in operation. The antenna orientation was changed through multiple axes until a worst case level was detected.

The GE Nuclear Measurement Analysis and Control system was tested to a field strength level of 65 V/M over the frequency range of 20-990 MHz. The maximum radiated electric field susceptibility from the EMI mapping survey was around 3V/M when the plant hand-held radios were keyed on and off. Another significant emission level was 2V/M when the refuel floor overhead crane was included as part of the normal plant operating conditions. The test

results showed that the plant's EMI environmental conditions are within the tested envelope by the site survey.

The inspector observed portions of this test performed in the Unit 2 refuel floor on April 30, 1993. The inspector discussed the results of the special test with licensee representatives and reviewed the safety assessment for this special test. No test deficiencies were identified with the completed special test.

EMI mapping survey for Unit 2 control room locations will be performed during Unit 2 startup. The survey for Unit 1 and 3 will be performed upon the completion of Unit 2 EMI mapping.

c. Steam Dryer Placement

While placing the steam dryer in the reactor vessel, the top of the guide rod contacted a guide bracket resulting in a weld tearing between the bracket and the bracket gusset plate. The dryer has two guide brackets 180 degrees apart that guide the dryer while it is being lowered into the reactor vessel so that it does not contact the vessel wall or the shroud head and separator assembly. The bracket construction consists of two vertical plates supported by horizontal gusset plates. The guide rod, that is attached to the shroud head at the bottom, extends upward to serve as a guide for the dryer, with the guide brackets sliding over the guide rod. It appears that the dryer turned slightly while being lowered resulting in the gusset plate contacting the top of the guide rod as the dryer was lowered. When enough weight was put on the gusset plate the weld broke. The operators lowering the dryer heard a loud "pop" and pulled the dryer back up. A video camera was used to document the damage. GE provided the analysis which indicated that the dryer was acceptable "as is" for use during this operating cycle. GE determined that sufficient weld still existed to prevent the gusset plate from detaching during this cycle. A recommendation was made to re-inspect the area following this cycle to determine if any degradation due to IGSCC has occurred. SIL 558, Steam Dryer Damage Prevention, was issued by GE April 22, 1993 but not received by the licensee before this event. The SIL describes other events associated with movement of the dryer and provides guidance on how to prevent damage.

The inspectors have reviewed the video tape of the dryer weld and consider the licensee's actions acceptable.

d. Unit 2 Core Reload

1) Core Reload

On April 25, 1993, at approximately 10:40 a.m., the licensee commenced reloading the Unit 2 core following the establishment of secondary containment on April 22. The

entire core had been offloaded during the earlier stages of the outage. Of the 764 bundles contained in the core, roughly 39 percent or 296 bundles were replaced with new bundles of GE9B and GE7B barrier fuel, while the remaining 61 percent of the core consists of either once or twice burned non-barrier fuel. While the core was offloaded, the licensee sipped 552 bundles and determined that 2 of the bundles intended for reload contained minor flaws and would not be reused. Core reload was completed on May 3, 1993, at approximately 2:15 p.m., with core verification being completed on May 4, 1993.

2) Fuel Movement Errors

During the reloading of the core, two errors were made by the refueling bridge operators. On April 27, between 2:00 a.m. and 4:00 a.m., the operators installed fuel bundle LY6517 into the core 90 degrees from its specified orientation. This matter was discovered approximately 12 hours later by the day shift refueling bridge operators. Fuel movement was stopped and a field change to the FATF was written and approved to allow reorientation of the bundle. Following the reorientation of the bundle, core reloading was resumed. The next morning, April 28, at approximately 2:30 a.m., the refueling bridge operator removed an incorrect bundle from the spent fuel pool and installed it into the reactor core. The error was noticed immediately after the operator moved the bridge back to the spent fuel pool as the bundle had been removed from the incorrect row in the spent fuel rack. This type of error is easily recognizable as the fuel bundles in the spent fuel pool were previously installed such that they were to be removed in a consecutive row by row manner. Fuel movement was stopped and a field change was initiated to the FATF to remove the incorrect fuel bundle and install the correct bundle into the core. As a part of the licensee's investigation of this occurrence, it was determined that the refueling floor SRO and one of the bridge operators were preoccupied with a fuel pool skimmer surge tank water level problem. Procedure 0-GOI-100-3, Refueling Operations, and TS 6.2.2.f state in part that a licensed SRO who has no concurrent responsibilities be present during fuel handling and shall directly supervise all core alterations. Procedure 0-GOI-100-3 further states that two fuel handlers be stationed on the refuel bridge when moving fuel. Although the second fuel handler remained on the bridge, discussions between himself and the SRO diverted their attention from the on-going fuel movement.

Prior to resuming fuel handling operations, the refueling floor SRO stressed the importance of taking extreme caution when handling fuel to ensure that each evolution is

conducted properly. After issuing the field change to the FATF, the improper fuel bundle was removed from the core and returned to the spent fuel pool. The licensee briefed all fuel handlers on the fuel handling errors and stressed the importance of attention to detail when handling fuel. Following the two fuel handling errors, the licensee's Fuels Engineering organization reviewed the consequences of starting up the plant without finding the errors. Based on their reviews, shutdown margin would have decreased an insignificant amount and any changes to the Critical Power Ratio Limit were bounded by the GESTAR analysis.

TS 6.8.1.1.a requires that written procedures be implemented as recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978. Regulatory Guide 1.33, paragraph 2.1, recommends procedures for Refueling and Core Alterations. The Fuel Assembly Transfer Form (Appendix B to SSP-5.4, Special Nuclear Material Control) and Procedure 0-GOI-100-3, Refueling Operations, implement these requirements. The two fuel handling errors are a violation of the FATF and procedure 0-GOI-100-3 and will be tracked as NCV 50-260/93-18-02, Fuel Handling Errors. This violation will not be subject to enforcement action because the licensee's efforts in correcting the violation meet the criteria specified in Section VII.B of the Enforcement Policy.

3) Fuel Assembly Orientation Inspection

On May 4, 1993, the inspector watched a video of the fuel assemblies in the Unit 2 reactor core. The purpose of this video was for nuclear engineers to verify proper core loading. The inspector watched portions of the two VCR tapes provided by the licensee, and verified that the fuel assemblies were oriented properly. Proper fuel assembly orientation was verified by the location of the channel fasteners, identification lug pointing toward the center of the fuel cell, channel spacer button's location, readable bundle serial number from the center of fuel cell, and cell to cell symmetry throughout the core. No deficiencies were noted.

e. Core Spray System Returned to Service Without Hydrostatic Test of Weld Repair

On April 27, 1993, the licensee requested relief from ASME Section XI Inservice System Pressure Test Program. During the Unit 2 Cycle 6 refueling outage a 12 inch weld repair on the core spray system and two two-inch socket welds on the vessel bottom head drain to the RWCU system were made. Due to the absence of isolation valves between these areas and the RPV compliance with the hydrostatic testing would require pressurizing the entire RPV.

The licensee requested relief from the NRC and proposed as one alternative delaying the test until the performance of a leak check at normal operating pressure.

On April 29, 1993, in the POD the plant manager inquired from his staff if the relief had been granted. It was stated that the relief was granted verbally and a letter would follow in a day or two. This discussion was the basis for declaring both loops of CS operable. Per TS 3.5.A.4 one loop of CS is required with fuel in the vessel and the reactor cavity drained.

On May 10, 1993, the licensee was notified that the relief would not be granted. At the end of this report period no written acceptability of these weld repairs and hydrostatic test requirements had been received. This issue is under review by NRC management and final resolution will be discussed in IR 93-23.

One non-cited violation was identified in the Operational Safety Verification area.

5. Unit 2 Cycle 6 Outage Commitments

The inspectors reviewed the licensee's commitments to be completed prior to startup from the current refueling outage. These consisted of field inspections, test results review, and training.

a. Training

The inspector reviewed the training materials provided for the Unit 2 Cycle 6 modifications as part of the operator requalification training program. The materials covered 31 DCNs and covered the more significant modifications that will change the operation of the unit. Included in this were the hardened wetwell vent, containment isolation status system as part of the CRDR modifications, NFPA upgrades, CREV system, and others. The training material provided a detailed description of the modifications made during the outage affecting plant operations.

b. Examination of EECW Welds for MIC

By letter dated September 29, 1988, TVA committed to the NRC that a population of the Unit 2 EECW butt welds which were previously inspected by RT will be re-radiographed before Unit 2 restart and at each Unit 2 outage to ensure structural integrity of the system. Any increase in indications will be re-analyzed to determine any affect on structural integrity of the system.

This commitment was placed in procedure SSP-13.5, Raw Water Fouling and Corrosion Control Program. More specifically, paragraph 3.5.2.G of SSP-13.5 requires the Site Quality Organization to radiographically examine a population of the Unit 2 butt welds which were previously inspected by RT prior to the

Unit 2 restart (in 1991) and at each Unit 2 outage. The inspector examined the results of the inspection performed for the Unit 2 Cycle 6 outage. These inspections were performed in November and December 1992, prior to the completion of cycle 6. The licensee examined 37 welds in the EECW system, several of which had previous MIC indications. The results of this inspection indicate that there was some growth of the MIC, however based on engineering calculations and TVA's engineering judgement, the integrity of the system is not affected and is acceptable for continued use. These welds and a sampling of other welds will be radiographed next refueling outage. Based on this review of TVA's actions on this matter, the licensee has completed their commitment. The radiographs were reviewed by a regional inspector and the results documented in IR 93-21.

c. Hardened Wetwell Vent

Throughout the outage the inspectors monitored the construction of the Unit 2 HWWV. The HWWV was installed to satisfy the recommendations of NRC Generic Letter 89-16. During this inspection period, with construction of the modification being complete, the inspection effort focused primarily on post modification testing. On April 15, 1993, the inspector observed the performance of portions of 2-PMT-BF-064.034, Hardened Vent Valve Indication and Alarm Functional Test. This test was performed to ensure the proper function of the various valves, indications and alarms associated with the system. Specifically, the PMT tested the position indications for valves 2-SHV-64-737, 2-FCV-64-221, and 2-FCV-64-222, and the Valves Mispositioned annunciator on panel 2-XA-55-4C. In addition, the PMT verified the interlock function of the valve permissive switches for valves 2-FCV-64-221 and 222.

All aspects of the test associated with valves 2-FCV-64-221 and 222 were performed with satisfactory results. However, the testing of the indication and alarm functions associated with valve 2-SHV-64-737 demonstrated that the limit switches associated with the valve were out of adjustment. The test director properly documented the test deficiencies and initiated a work order to troubleshoot and ultimately adjust the limit switches. Following the adjustment of the limit switches, valve 2-SHV-64-737 was tested satisfactorily.

The final aspect of the functional test dealt with the verification of a clear flow path in the hardened wetwell vent from the reactor building to the vent's exit point in the stack. This portion of the PMT was prompted by the inspectors questioning the licensee on how they planned on verifying a clear flow path. The inspectors major concern was the possibility of foreign objects being left in the vent piping during its construction. Following the inspectors inquiry, the licensee determined that this testing was prudent and a procedure was developed. The

testing involved the injection of helium into the piping within the reactor building and detecting the gas as it exits the HWWV in the stack. The test was performed and properly demonstrated that a flow path exists between the reactor building and the HWWV exit point.

d. Control Room Design Review

By letter to the NRC dated December 28, 1993, the licensee committed to completing NUREG-0737 (TMI Action Plan), item I.D.1, Detailed Control Room Design Review, category 1 and 2 HEDs prior to restart from cycle 6 refueling outage. The inspector reviewed the completed packages for the HEDs and found no discrepancies. In addition several HEDs were selected and verified to have been implemented as designed. Based on this review, this licensee has completed their commitment.

e. Process Computer

By letter dated July 11, 1989, the licensee committed to implement NUREG-0737, (TMI Action Plan), item I.D.2, installation of the Safety Parameter Display System. The ICS which includes SPDS was installed during this outage but will not be plant accepted until a reliability run is completed. The reliability run and testing is scheduled to take place during restart and power ascension. Testing of field connections were accomplished during installation by verifying continuity of the new wiring and by verifying the new connection did not impact a previously existing connection, such as a recorder or indicator. Verification of the heat balance and thermal limit calculations will be accomplished at various stages during startup by performing hand calculations and comparing them to the calculations performed by the ICS. Although the majority of the testing will be completed, acceptability and turnover from the vendor of the ICS is expected to take approximately six weeks. This process of testing the ICS is in accordance with the guidelines set forth in Regulatory Guide 1.68, Initial Test Programs For Water-Cooled Nuclear Power Plants. The inspector will continue to monitor testing of the ICS during this period.

f. RG 1.97 Instruments

In NRC IR 90-32 it was concluded that adequate controls and planning to ensure that the RG 1.97 program as committed to by TVA to the NRC, will be fully implemented prior to Unit 2 restart. Before cycle 7 operation the licensee committed to install the following qualified instruments:

CS flow
LPCI flow
RHR flow
RHR heat exchanger outlet temperature
Cooling water temperature to ESF components

Cooling water flow to ESF components
Emergency ventilation damper positions

The inspector reviewed the licensee closure package for this item. First, the CS flow instrument was replaced under DCN W17439A. The inspector reviewed this DCN that was closed April 28, 1993. For RHR flow and LPCI flow the same flow instrument is used. This was replaced under DCN W17440.

For the RHR heat exchanger outlet temperature the licensee classified this item as a category three item based on no operator action with RHR heat exchanger outlet temperature. Therefore, this was not included in the EQ program. Other parameters such as suppression pool temperature, suppression pool level, drywell temperature and pressure, and reactor coolant level are in the EQ program and the EOIs key on these parameters.

For the cooling water temperature to ESF components, the range of instruments specified by RG 1.97 was 40 degrees F to 200 degrees F. This was for closed cycle cooling systems. At BFNP the cooling water temperature can only be affected by the ambient river water temperature. The temperature monitors in the forebay and associated components are in a mild environment except for a cable that passes through Unit 3 reactor building. This cable will be qualified prior to Unit 3 restart. For cooling water flow to ESF components four instruments were replaced under DCN W17421. The inspector reviewed the applicable DCN.

For the emergency ventilation damper positions, DCNs S17250A, W17317A, and W17316A were performed to address this item.

Additionally, the inspector reviewed correspondence in 1988, 1989, 1990, and 1991, and concluded that except for neutron flux monitoring the licensee has completed the commitment for RG 1.97. An NRC safety evaluation was issued on May 10, 1991. The NRC staff concluded in the SER that the licensee conforms with or has adequately justified deviations from the guidance of RG 1.97, with the one exception. Implementation of qualified neutron flux monitoring capability has been deferred pending review of an appeal of this issue by the BWR Owners Group to the Director of NRR.

g. Public Address and Evacuation System

The licensee committed to upgrade the public address and evacuation alarm system during the Unit 2 Cycle 6 refueling outage in order to meet the intent of IE Bulletin 79-18, Audibility Problems Encountered On Evacuation Of Personnel From High-Noise Areas. This item was tracked by the BFNP NPP, Volume III, Section II-77, as item 40. In 1986, the licensee employed an engineering contractor to evaluate the operation of BFNs alarm and evacuation systems. The results indicated many inadequacies in the existing

system. As a result of these findings, DCNs W15757, W15723, W15724, and 15756, were written to upgrade the system for Units 1,2,3, and plant common areas, respectively. The upgrades included among other things, the addition of strobe lights, electronic sirens, speakers, and uninterruptible power supplies. The modifications were completed and testing was in progress at the time of this inspection. The inspector witnessed portions of PMT-233, that tested the evacuation system for Unit 2. One discrepancy was noted due to a typographical error which wrongly identified the location of a junction box. This was corrected by implementing a nonintent change. Test deficiencies were being tracked and corrected as found. All testing is expected to be completed by unit startup.

6. Self-Assessment (40500)

a. Generic Letter 92-08

The inspector reviewed the licensee's response to NRC Generic Letter 92-08, Thermo-Lag 330-1 Fire Barriers, dated April 14, 1993. Modifications were made to eliminate the use of thermo-lag except in the intake pump station. The licensee requested an exemption from 10 CFR 50 Appendix R requirements for RHRSW power cables installed at the intake pump station in Enclosure 2 of the response. The division I and II are less than 20 feet apart and do not meet the separation criteria. Stated in the response was that the division II cable located in the cable trays are coated with "Flamastic". However, the cables are not in flamastic but are laying in the cable trays on top of existing flamastic. New cables were installed prior to Unit 2 restart.

The inspector toured the intake cable tray tunnel on April 21, 1993, to confirm the cable location and coating. Six groups of three cables for each of the three electrical phases for six of the 12 or one division of the RHRSW pumps were located in the cable trays uncoated. This error in the response was discussed with the licensee's licensing manager and NRC project manager on April 21, 1993. The licensee stated a correct submittal would be made.

b. Concerns Resolution Programs

On April 19, 1993, the inspectors met with TVA's Concern Resolution Corporate Manager and Browns Ferry's Concerns Resolution Site Representative to discuss Concerns Resolution Programs at Browns Ferry. The discussion focused on the Concerns Resolution programs of TVA's primary contractors at the site. Recently, TVA performed an audit of the contractors concerns programs and found that overall the programs were adequate. However, some discrepancies were identified during the review, the most significant of which was that although personnel were familiar with TVA's concerns program and the ability to speak with the NRC, they were not that familiar with their own company's employee concerns program. This finding has been presented to the contractors and corrective actions should be in place by the end of May. TVA plans to audit their contractors concerns programs on an annual basis.

The inspectors and the concerns resolution staff also discussed TVA's Concerns Resolution Program (as delineated in SSP 1.2) as is done on an approximately bimonthly basis. This program presents an avenue for TVA employees to resolve quality or safety issues. Current matters being handled by the concerns resolution staff were discussed in general with the inspectors. Of the items discussed, the inspectors determined that the concerns resolution staff was taking the appropriate corrective actions. The inspectors will continue to meet with the licensee's Concern Resolution Staff to ensure that quality and safety concerns are being properly dispositioned.

7. Unit 3 Restart Activities (30702, 37828, 61726, 62703, 71707)

Minimal construction activities continued on Unit 3, with the majority of the licensee's efforts being directed towards the completion of the Unit 2 cycle 6 refueling outage. The licensee is currently finalizing the new Unit 3 restart schedule which should be issued later this month. The inspectors will continue to follow the progress of the schedule and will provide copies of it to NRC management when issued. The inspectors continued to review the progress for Unit 3 restart by attending the daily POD meetings on site and periodically attending the Bechtel engineering POD meetings at their complex in Athens, Alabama.

a. System SPOC Status

The inspectors reviewed completed SPOC packages for the following systems during this period:

- System 27 C (Cooling Tower System). This SPOC package included portions of System 25 (Raw Service Water), System 27 (Condenser Circulating Water), System 77 (Radwaste), System 205 (4-KV Cooling Tower Switchgear), System 232 (480-V Cooling Tower Boards), and System 246 (Cooling Tower Transformer) which were required to support Cooling Tower System operation.
- System 39 (Carbon Dioxide Fire Protection System). This package was for the Unit 3 generator purge portion only, as the remainder of this system was placed in service prior to the Unit 2 restart.

The packages reviewed met the requirements of SSP-12.55, Unit 3 System Pre-Operability Checklist, Revision 3, with no major deficiencies being identified.

No violations or deviations were identified.

8. Reportable Occurrences (92700)

The LERs listed below were reviewed to determine if the information provided met NRC requirements. The determinations included the verification of compliance with TS and regulatory requirements, and addressed the adequacy of the event description, the corrective actions taken, the existence of potential generic problems, compliance with reporting requirements, and the relative safety significance of each event. Additional in-plant reviews and discussions with plant personnel, as appropriate, were conducted.

a. (CLOSED) LER 260/91-06, Unplanned Reactor Protection System Actuation Resulting From Local Power Monitor Leakage Current

On March 29, 1991 an unplanned RPS actuation occurred on Unit 2 when APRM B drifted high exceeding the Hi-Hi setpoint. The cause was determined to be an LPRM which inputs to this APRM, drifting high due to high leakage current. The reactor was in cold shutdown at the time of the event. As corrective action the licensee replaced the LPRM. In addition, the licensee cleaned or replaced various connectors which also exhibited high leakage and performed other maintenance recommended by the vendor to reduce EMI and noise problems. The inspector reviewed LPRM maintenance procedures SII-00-XX-92-051, SII-2-XX-92-052, and SII-0-XX-92-0054 which were revised to incorporate vendor recommendations. Based on the above the inspector determined the licensee has completed corrective actions which would preclude this event from recurring.

b. (CLOSED) LER 260/92-05, Engineered Safety Feature Actuation Resulting From the Wrong Test Switch Being Turned During a Surveillance Instruction.

On May 5, 1992, isolation of the refueling zone ventilation and initiation of ESF equipment occurred when an instrument mechanic performed a functional test on the wrong radiation monitor. A shorting plug had been installed on the correct monitor in the auxiliary instrument room to prevent actuation of the ESF during testing, as required. However, when the mechanic returned to the control room he went to the wrong monitor and when he placed the test switch to "trip", the actuation occurred. As corrective action, the individuals involved were counselled on the importance of paying close attention to detail, following work procedures, and self-checking each step. Maintenance supervisors held discussions with craft personnel on this event and the need to perform self-checking and maintain a questioning attitude. Additionally, craft personnel received training on SSP-12.6, Verification Program. Based on a review of records pertaining to the above corrective action the inspector determined that adequate licensee actions have occurred to preclude a recurrence of this event.

c. (CLOSED) LER 260/92-08, Unit 2 Hydrogen Analyzers for the Offgas System Were Inoperable Due to the Loss of Control Air to the Sample Panel.

On October 12, 1993, at 9:08 a.m., the licensee declared the Unit 2 hydrogen analyzers for the off-gas system inoperable due to the loss of control air to the sample panel. The actual time of occurrence of the loss of control air to the Unit 2 hydrogen analyzers was at 5:30 a.m. when Unit 1 operations personnel implemented a hold order that closed valves on a Unit 1 control air supply header and inadvertently isolated the control air to the Unit 2 hydrogen analyzers. Consequently, the required compensatory off-gas sample was not taken within the TS timeframe.

The cause of the isolation of the control air to the Unit 2 hydrogen analyzer was determined to be a drawing discrepancy in the control air system. The drawing did not show that the Unit 1 off-gas hydrogen analyzer control air also supplies the Unit 2 off-gas hydrogen analyzer. Subsequently, the hold order was released and the system was returned to normal. A caution order was initiated and caution tags were placed on the valves. A DCN was issued to revise the affected drawings to show the correct air flow configuration for the Unit 1 and Unit 2 hydrogen and oxygen analyzers. Revision 5 of drawing 1-47E1847-5 incorporated the changes to reflect the correct plant configuration.

The inspector reviewed the closure package provided by the licensee, interviewed licensee representatives, and concluded that the corrective actions which should preclude recurrence of this issue have been completed.

9. Action on Previous Inspection Findings (92701, 92702)

a. (CLOSED) IFI 259, 260, 296/92-17-02, Documentation Drawing Change Authorization and Electrical Cable Installation to Work Documents.

This item was opened due to a difficulty in tying some DCAs from a DCN directly to an implementing WP during closure by SWEC. The second item concerned an observation that several electrical cable reels did not have the required IN USE tags attached with the work document specified. To correct the first issue the licensee canceled the contractors procedure (EDPI-4-90-10) and replaced it with TVA procedure PI-88-07, System Plant Acceptance Evaluation. This will ensure that contractor activities are controlled to the licensee's specifications. The second concern appears to have been an isolated case as subsequent QA inspectors and NRC observations indicated that IN USE tags were attached to equipment remaining in the field for longer than one shift as required by SSP-10.4, Material Issue, Control, and Return. In addition, MAI-3.2, Cable Pulling for Insulated Cables Rated Up to 15,000 Volts, requires that craft personnel assigned to pull cables log the cable mark and cable reel numbers on the cable

installation/pullback data sheet. This data sheet is also reviewed by QC which would ensure the proper cable is being used in the field. Based on the above review the inspector determined that appropriate corrective actions had occurred to preclude recurrence.

b. (CLOSED) URI 260/92-30-02, Unclear Calibration Surveillance.

While observing the performance of Surveillance Instruction 2-SI-4.2.B.24(II), Core Spray Sparger To Reactor Pressure Vessel Differential Pressure Calibration, the inspector noted that the instrument mechanic performing the procedure did not follow the

procedure verbatim. The procedure verifies proper calibration of the installed gauge by connecting a calibrated gauge in parallel, pressurizing both, and verifying they compare within a certain tolerance. The procedure was originally written to pressurize the gauges until the installed gauge indicated a specified value and then compare this to the calibrated gauge. The technician pressurized the gauges until the calibrated gauge indicated a specified value and then compared this to the installed gauge. The licensee stated that the method used by the technician is typically the method used for calibrations of this type. The step which was performed incorrectly was not clear as originally written. The test, even though not performed as intended, produced the same results. The licensee revised the procedure to clarify this step so that the calibration will be performed using the method typically used.

c. (CLOSED) VIO 296/92-08-01, Fire Watch Inattentive to His Duties.

On February 28, 1992, during an inspection of weld repairs to Unit 3 valve 74-49, the inspector observed the designated fire watch for the repair activity to be asleep. The individual was sitting with his back to the drywell wall, his head tilted back, and his eyes closed. The inspector observed the individual in this condition for approximately two minutes before arousing him. This matter was identified as violation 296/92-08-01.

The licensee's procedures, specifically Attachment-F of FPP-2, "Fire Protection System/Equipment Removal From Service", requires that a fire watch be on continuous alert for signs of fire or anything which could result in a fire. Specific training received by the involved fire watch further emphasized this aspect of the duties of a fire watch. Since the individual involved had successfully completed fire watch training and then failed to properly perform his assigned duties, appropriate personnel action was taken by the licensee towards the involved individual. Based on this review of the licensee's corrective actions, this violation is closed.

10. Exit Interview (30703)

The inspection scope and findings were summarized on May 18, 1993, with those persons indicated in paragraph 1 above. The inspectors described the areas inspected and discussed in detail the inspection findings listed below. The licensee did not identify as proprietary any of the material provided to or reviewed by the inspectors during this inspection. Dissenting comments were not received from the licensee.

Item Number	Description and Reference
50-260/93-18-01	URI, Loss of Primary Pressure Control, paragraph two.
50-260/93-18-02	NCV, Fuel Handling Errors, paragraph four.

Licensee management was informed that 3 LERs, 1 IFI, 1 URI, and 1 VIO was closed.

11. Acronyms and Initialisms

APRM	Average Power Range Monitor
ARI	Alternate Rod Injection
ASME	American Society of Mechanical Engineers
ASOS	Assistant Shift Operations Supervisor
ATWS	Anticipated Transient Without Scram
AUO	Auxiliary Unit Operator
BFNP	Browns Ferry Nuclear Plant
BWR	Boiling Water Reactor
CFR	Code of Federal Regulations
CRDR	Control Room Design Review
CREV	Control Room Emergency Ventilation
CS	Core Spray
DCA	Drawing Change Authorization
DCN	Design Change Notice
EECW	Emergency Equipment Cooling Water
EMI	Electromagnetic Interference
ENS	Emergency Notification System
EOI	Emergency Operating Instruction
EQ	Environmental Qualification
ESF	Engineered Safety Feature
FATF	Fuel Assembly Transfer Form
FCV	Flow Control Valve
FPP	Fire Protection Plan
GE	General Electric
GOI	General Operating Instruction
GPM	Gallons Per Minute
HED	Human Engineering Discrepancy
HWWV	Hardened Wetwell Vent
HZ	Hertz
ICS	Integrated Computer System
IE	Inspection and Enforcement

IFI	Inspector Followup Item
IGSCC	Intergranular Stress Corrosion Cracking
IR	Inspection Report
KV	Kilovolt
LCI	Loop Calibration Instruction
LCO	Limiting Condition for Operation
LER	Licensee Event Report
LPCI	Low Pressure Coolant Injection
LPRM	Local Power Range Monitor
MAI	Modification Alteration Instruction
MIC	Microbiological Induced Corrosion
MSRV	Main Steam Relief Valve
NCV	Non-cited Violation
NFPA	National Fire Protection Association
NPP	Nuclear Performance Plan
NRC	Nuclear Regulatory Commission
PMT	Post Modification Testing
POD	Plan of Day
QA	Quality Assurance
QC	Quality Control
RG	Regulatory Guide
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
RPS	Reactor Protection System
RPT	Recirculation Pump Trip
RPV	Reactor Pressure Vessel
RWCU	Reactor Water Cleanup
SER	Safety Evaluation Report
SI	Surveillance Instruction
SII	Special Instrument Instruction
SIL	Service Information Letter
SLC	Standby Liquid Control
SPDS	Safety Parameter Display System
SPOC	System Pre-Operation Checklist
SRO	Senior Reactor Operator
SSP	Site Standard Practice
SWEC	Stone & Webster Engineering Corp.
TMI	Three Mile Island
TS	Technical Specification
TVA	Tennessee Valley Authority
URI	Unresolved Item
VIO	Violation
V/M	Volt/Meter
WO	Work Order
WP	Work Plan
WR	Work Request