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

Report Nos. 50-259/86-32, 50-260/86-32, and 50-296/86-32

Licensee: Tennessee Valley Authority
6N 38A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Docket Nos. 50-259, 50-260, and 50-296

License Nos. DPR-33, DPR-52, and DPR-68

Facility Name: Browns Ferry Nuclear Plant

Inspection Conducted: September 1 - 30, 1986

Inspectors: A D Johnson for — 10/21/86
G. L. Paulk, Senior Resident Inspector — Date Signed

A D Johnson for — 10/21/86
C. A. Patterson, Resident Inspector — Date Signed

A D Johnson for — 10/21/86
C. R. Brooks, Resident Inspector — Date Signed

Approved by: F. S. Cantrell — 10/22/86
F. S. Cantrell, Section Chief — Date Signed
Division of Reactor Projects

SUMMARY

Scope: This routine inspection was in the areas of operational safety, maintenance observation, surveillance testing observation, reportable occurrences, maintenance program, review of special reports, design baseline and verification program, and annual emergency drill.

Results: Two violations were identified:

1) Technical Specification 6.3.A for an inadequate test procedure for the control room emergency ventilation flow rate test.

2) 10 CFR 50 Appendix B Criterion V for failure to document equipment problems during an emergency equipment flow-rate test.

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REPORT DETAILS

1. Licensee Employees Contacted:

   H. P. Pomrehn, Site Director
   J. G. Walker, Deputy Site Director
   J. P. Stapleton, Project Engineer
   *R. L. Lewis, Plant Manager
   E. A. Grimm, Assistant to the Plant Manager
   *J. E. Swindell, Superintendent - Unit Three
   R. M. McKeon, Superintendent - Unit Two
   T. D. Cosby, Superintendent - Unit One
   T. F. Ziegler, Superintendent - Maintenance
   *D. C. Mims, Technical Services Supervisor
   J. G. Turner, Manager - Site Quality Assurance
   *M. J. May, Manager - Site Licensing
   *R. D. Schulz, Compliance Supervisor
   A. W. Sorrell, Health Physics Supervisor
   R. E. Jackson, Chief Public Safety
   *J. Savage, Compliance Licensing
   *J. Webster, Electrical Section
   *D. Pullen, ONP Site Representative

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

   *Attended exit interview.

2. Exit Interview (30703)

   The inspection scope and findings were summarized on October 3, 1986, with those persons indicated in Paragraph 1 above.

   The licensee acknowledged the findings and took no exceptions. The licensee did not identify as proprietary any of the materials provided to or reviewed by the inspectors during this inspection.

3. Licensee Action on Previous Enforcement Matters (92702)

   (Closed) Unresolved Item (259/260/296/86-25-11) This item pertains to the adequacy of Surveillance Instruction (SI) 4.7.E.5, Control Room Emergency Ventilation System Flow Rate Test. Inspection Report 86-25 contains the details for several areas of the SI which do not satisfy the requirements of Technical Specifications 3.7.E.2.c and 4.7.E.2.a. The licensee was unable to justify deviation from the requirements and therefore this item has been upgraded to a violation (259, 260, 296/86-32-01).

(Closed) Open Item (259/84-53-02) As a result of three incorrect computer data points for the Reactor Protection System Channels, the licensee conducted a sample of 16 other alarm points. The previously incorrect points were verified correct and no additional problems were found. The cognizant electrical maintenance, instrument maintenance, and modifications personnel concluded that the problem was an isolated one caused by wiring errors, drawing errors, and an inadequate post modification test associated with one Engineering change notice. The inspector concluded that this was another example of the programmatic breakdowns at the facility. Volume III of the Performance Plan addresses the programmatic problems. This item is closed.

(Closed) Inspector Followup Item (259/84-26-11) The clearance in question is no longer required and has been released. All tags have been verified as removed. This item is closed.

(Closed) Violation (259, 260, 296/82-34-06) To establish control of thermometers all existing power stores stock have been removed. Future thermometer purchases will only be through power stores in Chattanooga. All thermometers will be certified prior to issue at Browns Ferry. Standard Practice 17.19, "Program to Establish and Maintain Certifiably Accurate Thermometers". This item is closed.

(Open) Inspector Followup Item (259, 260, 296/86-05-08) The inspector reviewed the procedure changes and new program for control of static charges on control room meters. The issue of static charges on meter surfaces causing misleading information to the operator has been adequately resolved. S.I. 4.2.A.10 has been adequately revised to address all of the previously identified problems however a new issue has surfaced and therefore the Inspector Followup Item will remain open. The new issue involves system interaction concerns related to the procedure revision. The revised procedure now requires that prior to calibration or functional test of a radiation monitor, the normal ventilation in the reactor zone be isolated and the SGTS started. No cautions are contained in the procedure nor is it clear that the potential for a reactor trip as a result of this action has been evaluated. The plant has a record of steam tunnel temperature control problems which are magnified during a reactor zone isolation on an operating unit, sometimes resulting in a reactor trip. Licensee management indicated that they would reevaluate the procedure change. The reason for the change was to avoid spurious ESF actuations during the surveillance by manually initiating the ESFs before the surveillance begins. This was considered necessary due to the excessive number of spurious actuations which have occurred in the recent past.

(Open) Unresolved Items (259, 260, 296/85-20, 85-24) HPCI System Operability Determination due to failed transient suppression networks on HPCI system valve circuits. The inspector attempted to close out open and unresolved items associated with I.E. Reports 85-20 and 85-24 related to the HPCI system operability. The licensee evaluation for the system design and functional capability remains incomplete, therefore, this item remains open.

(Open) Violation (259, 260, 296/85-36-01) HPCI system safety evaluation. A review of the response to this violation indicated the response to be inadequate in that the corrective action and reasons for violation section description were based on an erroneous safety evaluation. The safety evaluation conclusions were based on a system circuit analysis whose location did not exist. The inspector asked for a followup to this response based on documented conclusions.

4. Unresolved Items* (92701)

There were four new unresolved items identified in paragraphs 5 and 9 related to the following topics:

a. Preplanned alternate monitoring procedures to be used in the event of containment high range radiation monitor inoperability.

b. Technical Specification surveillance interval for analog trip unit instrumentation.

c. Potential licensee identified violations found during a corporate maintenance study.

d. Training deficiencies in the "Rigging Fundamental" course.

5. Operational Safety (71707, 71710)

The inspectors were kept informed of 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 when an inspector was on site. Observations included instrument readings, setpoints and recordings; status of operating systems; status and alignments of emergency standby systems; onsite and offsite emergency power sources available for automatic operation; purpose of temporary tags on equipment controls and switches; annunciator alarm status; adherence to procedures; adherence to limiting conditions for operations; nuclear instruments operable; 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 their supervisors.

General plant tours were conducted on at least a biweekly 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; containment isolation alignments; instrument

*An Unresolved Item is a matter about which more information is required to determine whether it is acceptable or may involve a violation or deviation.

readings; housekeeping; proper power supply and breaker; alignments; radiation area controls; tag controls on equipment; work activities in progress; and radiation protection controls. Informal discussions were held with selected plant personnel in their functional areas during these tours.

Biweekly verifications of system status which included major flow path valve alignment, instrument alignment, and switch position alignments were performed on the Electrical Distribution systems.

In the course of the monthly activities, the inspectors included a review of the licensee's physical security program. The performance of various shifts of the security force was observed in the conduct of daily activities to include; protected and vital areas access controls, searching of personnel, packages and vehicles, badge issuance and retrieval, escorting of visitors, patrols and compensatory posts. In addition, the inspectors observed protected area lighting, protected and vital areas barrier integrity.

a. Technical Specification Changes

Amendment No. 125 to Unit 2 Technical Specifications was issued by NRR on August 19, 1986. This amendment contains the core reload analysis and several significant changes due to hardware modifications and TMI equipment installations. Note 8 to Table 3.2.F, Surveillance Instrumentation, contains an action statement related to the inoperability of NUREG-0737 required Containment High Range Radiation Monitors and the Wide Range Gaseous Effluent Radiation Monitors. This note requires that the equipment be restored to operable within 72 hours or initiate the preplanned alternate method of monitoring the appropriate parameter. The inspector requested to review the preplanned alternate monitoring instructions but was informed by the licensee that no instructions had yet been developed. This will be tracked as an Unresolved Item (260/86-32-02) pending review of the approved procedures.

The inspector detected an apparent inconsistency between the Safety Evaluation supporting the amendment and the amendment itself. Section 2.C.3 of the Safety Evaluation contains the basis for the calibration interval of the new Analog Transmitter Trip System (ATTS). It states that total loop accuracy and total loop drift are added to obtain the trip setpoint. With total loop drift being dependent upon the calibration interval, any calibration interval in excess of that specified (18 months) could result in trip setpoints outside the required band. Since the manufacturer's specified drift values are extrapolated to determine the 18 month total drift, the calibration interval is therefore limited to once per operating cycle, not to exceed 18 months (according to the safety evaluation). The actual calibration interval specified in Table 4.1.B of the amended technical specifications (T.S.) is once per 18 months. This terminology together with the provision in the T.S. definition of surveillance interval, which allows an extension

of up to 25% of the surveillance interval, could result in a calibration interval in excess of 22 months. It appears as though the T.S. Table should have listed "no greater than 18 months" as the surveillance interval. This is supported by NRR's letter to TVA dated December 6, 1985, in which NRR stated that it was their position that the calibration intervals be limited to 18 months (550 days). This will be tracked as an Unresolved Item to investigate the potential for actual instrument trip setpoints to exceed the design basis. (260/86-32-03)

b. Setpoint Scaling Documents

In order to determine if sufficient margin was included in the Reactor Protection System instrument setpoints being modified by the ATTS discussed above, the inspector reviewed the scaling and setpoint documents (SSD) for several instruments. Contrary to the NRR safety evaluation of the ATTS which stated that manufacturer's specified drift values were extrapolated to 18 months, the inspector found that for the high reactor pressure transmitter (PIS-3-22 AA, BB, C, and D in T.S. Table 4.1.B), only the 6 month drift value was used in the setpoint calculation. This would require a 6 month calibration interval as opposed to the 18 months allowed by technical specifications. A licensee representative stated that the Surveillance Instruction being prepared for this instrument did indeed specify a 6 month interval. The inspector additionally found that the drift error for the trip unit associated with this instrument was determined to be ±0.1% of calibrated span based upon engineering judgement. It could not be determined if this value was for 1 month, 6 month or 18 month calibration intervals nor could any further justification for the engineering judgement be located. This information is totally contrary to the safety evaluation which assumed that total loop error (transmitter plus trip unit) was based upon manufacturer's specified data extrapolated to 18 months. Several errors were detected in the SSD's which were in the process of being updated and corrected. Some calculations referenced calibration intervals from the old technical specifications (T.S.) which were much shorter than the amended T.S. Some calculations indicated that the drift error for the trip units was based upon engineering judgement and would be verified by the calibration procedure while other calculations had deleted this verification requirement. The major problem however with the new RPS setpoints associated with the ATTS modification is that the reactor water level scram setpoint is calculated to be 22.5 inches (increased from the 11 inch setpoint of the previous electromechanical switches) and the main steam isolation valve (MSIV) closure setpoint is calculated to be -35 inches (increased from the previous -51" setpoint). It is expected that the effect of operating with setpoints this close to the normal water level of 33 inches would be a large increase in the number of reactor trips with almost every trip being accompanied by MSIV isolation. This is not in the spirit of reducing the number of unnecessary challenges to safety systems. The licensee has contracted with General Electric to aid in reducing the RPS water level setpoints and some initial results

are favorable. It is not known whether additional T.S. changes will be required to implement the GE setpoints. This issue will be tracked as an Inspector Followup Item. (259, 260, 296/86-32-04)

c. Incorrect Switch Labeling

On September 3, 1986, the inspector observed a test to verify the correct labeling of transfer switches for circuit breaker control power to 480 volt shutdown boards 3A and 3B. The test confirmed the permanent labels were incorrect. This concern was brought to the attention of the licensee last month (Reference IE Report 86-28). This item is similar to a discrepancy found on a 4160 shutdown board discussed in Licensee Event Report (LER) 259/85056. In the LER it was stated that the wiring discrepancy in shutdown boards were inspected and were found to the proper wiring configuration. Also stated was that inadequate labeling of the transfer switch contributed to the event.

The transfer switch was labeled "EMERGENCY-NORMAL". Above the switch are two lights which indicate the availability of control power from one of two 250 VDC battery boards. Both lights are normally illuminated if power is available. The transfer switch is a spring return to the neutral position type. During transfer of control power sources there is no indication of a successful transfer of sources unless the back cabinet door is opened to observe relay contacts. Above EMERGENCY was penciled NORM and above NORMAL was ALT. (Alternate is used to mean emergency.) Similar problems were found in both board rooms. The test concluded that the labels were incorrect. It was also found on the 3A board that the battery board potential lights wiring were reversed from all other similar boards. The location of the emergency supply and normal supply were reversed. The licensee planned to reverse the leads to these lights and install new labels.

The labeling problem is another example of a continuing lack of inquisitiveness on the part of the licensee. When this problem was found, licensee attention was not directed toward similar areas. This followup action on correction of a problem was inadequate and narrow in scope.

d. Low Air Pressure Scram

The inspector noted that in Amendment No. 125 the reason for the low scram pilot air header pressure trip was unclear. In Section 3.1, Bases, page 44, it states that the trip performs the same function as the high water level in the scram discharge instrument volume (SDIV) for fast fill events in which the high level instrument response time may be inadequate. This trip is unique to Browns Ferry and raises questions about the adequacy of the response time of SDIV high level instruments. This item will remain an Inspector Followup Item (259, 260, 296/86-32-05).

6. Maintenance Observation (62703)

Plant maintenance activities of selected safety-related systems and components were observed/reviewed to ascertain that they were conducted in accordance with requirements. The following items were considered during this review: the limiting conditions for operations were met; activities were accomplished using approved procedures; functional testing and/or calibrations were performed prior to returning components or system to service; quality control records were maintained; activities were accomplished by qualified personnel; parts and materials used were properly certified; proper tagout clearance procedures were adhered to; Technical Specification adherence; and radiological controls were implemented as required.

Maintenance requests were reviewed to determine 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 below listed maintenance activities during this report period:

a. Electrical Maintenance Instruction (EMI) 13, Reactor Protection System (RPS) Motor Generator (MG) set Periodic Maintenance.

b. Electrical Maintenance Instruction 58.2, Terminating of CSSC Low-Level Signal and Control, Low-Level Voltage Power Cables, and Internal Panel Wiring.

c. An inspection of the plant diesel fire pump building to observe maintenance on valve 26-1418, on September 30, 1986, indicated several material and housekeeping deficiencies as noted below:

(1) flash arrestors on the two direct current diesel start batteries were not attached properly to the cells.

(2) the battery mounting rack was not secured to the foundation by any means.

(3) general battery cleanliness was inadequate with excess acid spills, paint overspray on all cells, and loosely mounted cells with no side support spacers.

(4) the building housekeeping was poor with trash, boxes of old fluorescent lights, mechanical parts, oil bottles, etc. scattered throughout the room.

(5) the pump discharge strainer indicated a 30# delta pressure with no flow in the system. Expected delta pressure would be zero pounds.

(6) the pump discharge strainer backwash controller was found in the "hand" position as opposed to "Automatic". No guidelines for the operation of this controller which would automatically backwash

the strainer upon reaching a predetermined differential pressure could be located in Operating Instruction 26, High Pressure Fire Protection System. The required position of the controller was not listed in either the valve, instrument or switch alignment checklist in OI-26.

These items were made known to the licensee for corrective action and are listed as an Inspector Followup Item (259, 260, 296/86-32-06).

7. Surveillance Testing Observation (61726)

The inspectors observed and/or reviewed the below listed surveillance procedures. The inspection consisted of a review of the procedures 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, a review of test data, limiting condition for operation met, testing accomplished by qualified personnel, and that the surveillance was completed at the required frequency.

a. Surveillance Instruction 4.5.C.1 (4), EECW System Annual Flow Rate Test

On September 2, 1986, the inspector observed a repeat of Surveillance Instruction (SI) 4.5.C.1 (4). This test was repeated to resolve a number of inspector concerns observed on August 22, 1986. Although the inspector had expressed a number of concerns with the procedure and had requested to observe the test, no engineering personnel or supervision were present to observe the test. The test on August 22, 1986, was terminated due to a number of problems discussed in Inspection Report 86-28. The inspector requested to observe the test when performed again. The test was performed on August 23, 1986, apparently without difficulty, and the inspector was not informed the test was being performed. The test was repeated on September 2, 1986 and again a number of problems with the procedure and equipment were observed.

A licensee representative stated that some problems encountered during the test on September 2 were not observed on August 23 due to a change in several plant variables. The river water temperature had changed 10 degrees changing the setting of temperature regulating valves. The raw cooling water pressure was different due to different cooling loads and equipment out-of-service for maintenance. The licensee initiated procedure changes to clarify the procedure as to plant conditions during the test.

(1) Equipment Problems Noted on September 2, 1986 Test

First, problems were observed when the emergency equipment cooling water (EECW) cross-connect valves to the reactor building closed cooling water (RBCCW) system were positioned to the open position. The Unit 1 valve (67-50) would not open. The Unit 2 and Unit 3

valves came to an intermediate position as indicated by illumination of both the green and red valve position indicating lights. Observation of the valve stem movement in the reactor building indicated the Unit 2 valve was one-eighth open and Unit 2 three-fourths open. The following flow rates were observed as the valves were opened in sequence:

Pump with no valves open	3800 GPM
Unit 1 Valve	3800 GPM
Unit 2 Valve	3800 GPM
Unit 3 Valve	4200 GPM

When the test was performed on August 23, 1986, a flow-rate of 4500 GPM was achieved by only opening Units 1 and 3 valves. After the test on September 2, the following Maintenance Requests (MR) were initiated:

Unit 1 (MR 755513)	Valve found out of adjustment, needle valve in control circuit adjusted.
Unit 2 (MR 753912)	Replaced solenoid coil, core assembly was stuck causing the solenoid to leak through.
Unit 3	No licensee action as a result of the test.

Additionally, problems with these valves were discussed in Inspection Report 86-11. The valves use EECW system water directed to the valve diaphragm as the hydraulic medium for valve operation. The control circuit consists of an isolation valve, strainer, solenoid valve, two pressure gauges, two regulators, one needle valve and connecting lines. The settings of the needle valve and regulators were not known. The cognizant engineer initiated steps for a design change to the control system.

(2) Temporary Procedure Change

After only 4200 GPM could be achieved by opening all of the cross-connect valves, the diesel generator cooler throttle valves were to be opened. But, the procedure incorrectly listed the cooler outlet valves which are normally fully open. An immediate temporary change to the procedure was made to correct the valves listed. Although this SI had been through the SI upgrade program, this error was not detected. The procedure writer did not have a drawing which had been verified by a system walkdown to be correct.

As part of the configuration control problem all systems are being walked down to verify accurate drawings. No verified drawings have been completed as yet. However, the Plant Operations Review Committee continues to approve new procedures which may be inaccurate.

The installation of the diesel generator cooler inlet or throttle valves was a modification made in the past two years to prevent exceeding the 75 psig design pressure rating of the coolers. This was the subject of a civil penalty and resulted in installation of the inlet throttle valves to prevent exceeding the design pressure. The normal operating pressure with more than one EECW pump running is 100-125 psig. The procedure makes no mention of a precaution for operation of these valves. If these valves had been open on August 22, 1986, when two EECW were started unexpectedly (discussed in Report 86-28) the design pressure would have been exceeded.

(3) Inadequate Flow

After all of the diesel generator cooler valves were opened the flow-rate was only 4400 GPM. A decision was made to remove the hold-order clearance on the 3A diesel generator cooler valves as the maintenance had not started. In past tests each diesel cooler full flow condition was 500 GPM. However, opening of the 3A cooler valves provided only 50 GPM extra flow-rate. There was some speculation the pump (C3) was near a runout condition. This was one of five pumps which had the wrong type impeller installed on it. This is still an Unresolved Item (86-14-04). The licensee has sent one of the pumps back to the vendor for testing.

Next, although not mentioned in the procedure, the raw cooling water (RCW) pressure was lowered by stopping several RCW pumps. RCW is a backup to some of the EECW supplied components. The system with the highest pressure supplies cooling to the components. By lowering RCW pressure more components are supplied by EECW. The flow-rate was then observed to be 4500-4550 GPM.

(4) Review of Completed Data Sheets

Due to the difficulties encountered on September 2, 1986, the inspector reviewed the completed successful test of August 23, 1986. The surveillance instruction review cover sheet indicated there were no problems with the procedure. The surveillance instruction review form indicated in the remarks section that the SI was stopped at 1345 on 8-22-86 due to a conflict with another SI. It was resumed at 1305 on 8-23-86 and completed at 1340 on 8-23-86. No mention was made of any equipment deficiencies or procedure problems. A reviewer of the completed SI would have no way of knowing of the problems with the procedure or equipment.

Also, the completed data sheets were a combination of sheets from 8/22 and 8/23. A complete record of the 8/22 test was not retained.

Browns Ferry Standard Practice BF-10.9, Handling of Test Deficiencies, requires that deficiencies in surveillance instructions be documented along with the final disposition as part of the documentation package for the test being performed. A test deficiency is defined as any condition during which the equipment or system being tested: (1) fails to operate, (2) operates in a suspected adverse manner, or (3) operates outside the limits of a documented acceptance criteria. Failure to document the deficiencies with the cross-connect valves (67-50) during the test is a violation of 10 CFR 50, Appendix B, Criterion V, Instructions, Procedures, and Drawings for failure to follow BF-10.9 (259, 260, 296/86-32-07). This procedure implements the Nuclear Quality Assurance Manual, Part II, Section 4.9.

(5) Red-line on Gauge

The inspector noted that SI-4.5.C.1 (4) precaution and limitation step 3.6 states to prevent damage to the EECW pump motor, normal operating current should be less than 53 amps. The main control board ampere meter contained a red-line at 60 amps. The inspector questioned the reason for the difference. A control room operator might allow operation of the motor at 53-60 amps. This will remain an Inspector Followup Item for resolution of the reason for the differences (259, 260, 296/86-32-08).

8. Reportable Occurrences (90712, 92700)

The below listed licensee events 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 in-plant 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
*260/86-11 R1	August 22, 1986	RPS inadvertent trips due to MG set problems.

LER 260/86-11 R1 will remain open pending a more thorough evaluation of the cause of the fire in RPS MG set 1A and an evaluation of the potential failure of the thermostatically controlled high motor temperature trip circuit. This circuit tripped the 2A RPS MG set and prevented a fire but a very similar condition on the 1A RPS MG set led to a fire. A revision to

this LER will be required due to a failure of the licensee to include manufacturer and model number for the failed component as well as the elapsed time from failure to full operability.

9. Maintenance Program (62700)

The objectives of this annual inspection are to determine: 1) whether the maintenance program is being implemented in accordance with regulatory requirements; 2) the effectiveness of the maintenance program on important plant equipment; and 3) the ability of the maintenance staff to conduct an effective maintenance program.

It was noted that some improvements have been realized in maintenance practices at Browns Ferry as a result of the RPIP. Of these, the predictive diagnostic testing of motor operated valves with the MOVATS system appears to have the most direct impact on safety by allowing the detection of faulty conditions prior to actual failure. Upgraded training, expanded use of independent verification, and planning and scheduling improvements are also providing benefits. To further place in perspective the maintenance program at Browns Ferry, the inspector would judge the program to be no better nor worse than the typical nuclear plant maintenance program as described in NUREG-1212, "Status of Maintenance in the U.S. Nuclear Power Industry 1985". This judgement is applicable to only the program itself; implementation of the maintenance program would be judged below average.

The inspector became aware of maintenance in progress on the Unit 3 Refueling Bridge and investigated the administrative controls over the maintenance activity. Maintenance Request number A-730132 was initiated on June 20, 1986, to perform Mechanical Maintenance Instruction (MMI)-34, Refueling Platform Inspection and Repair, on the Unit 3 platform. Work commenced on August 21, 1986, with Section 8.9 which involves lubrication and adjustment of the bridge chain drive and sprockets. A maintenance representative stated that individual steps in the procedure may be performed in any order and since a problem was known to exist with the drive sprocket alignment, this step was entered first to allow inspection of the sprocket bearing. Step 8.9.1 which states, "Repair or replace parts as required" was considered ample instruction guidance by the workers. In order to perform this activity, however, the refueling bridge had to be lifted about one foot off the rails and set down on blocks. Since this lift involved over 1,000 lbs. and it occurred within 15 feet of the spent fuel pool (a critical lifting zone), the prerequisites, precautions and administrative controls contained in MMI-119, Lifting Instructions for the Control of Heavy Loads, were applicable. MMI-119 requires that a dedicated person be in charge of the lift, that the lift be verified to be an approved lift per MMI-119, that the reactor building crane be verified to be within its inspection interval requirements, that slings or other special lifting devices satisfy their inspection requirements, that operators meet all the training and qualification requirements, and that all of these prerequisites be documented on Data Sheet A of MMI-119.

No reference was made to the MMI-119 controls in MMI-34 nor were any lifting guidelines contained in the MR work instructions. Although the workers did control the lift properly, it could not be anticipated from the work instructions that a critical lift was required or that the Plant Operations Review Committee (PORC) understood that a critical lift would be necessary during their review of the procedures. After the bridge had been lifted and set down on cribbing blocks, a licensee review of piping penetrations through the secondary containment resulted in the secondary containment being declared inoperable. All work ceased on the bridge for over a month. No clearance tags were initiated to prevent inadvertent movement of the refuel bridge while it was elevated off of the rails for this duration. No safety evaluation was performed to assess the unqualified condition of the refuel bridge for an extended duration. Following a discussion with licensee management another MR was written and PORC approved which contained detailed instructions and prerequisites in order to remove the cribbing blocks from underneath the refuel bridge. An Unreviewed Safety Question Determination (USQD) was performed to evaluate the potential consequences of an accident or malfunction during the process. The bridge was restored to its proper configuration on September 26, 1986.

With regard to the first lift of the bridge on August 21, 1986, although the inspector was satisfied that applicable prerequisites were met prior to the lift, a question of proper rigging equipment selection was raised. The question arose after a visual inspection of the wire rope slings, which were used on the lift, identified a kink in the slings apparently caused by the sharp corner of the I-beam around which it was wrapped. A search of refuel bridge vendor documents could find no specified attachment points for rigging. The licensee used two wire rope slings rigged in a basket hitch around the top I-beam of the refuel bridge end A-frame. MMI-102, Rigging Equipment Program, identifies the rated capacity for this rigging configuration in Table 4. An asterisked note to Table 4 cautions that the rated capacities only apply with D/d ratios 20 or greater where:

D is diameter of curvature of the load and,
d is the diameter of the rope.

For the 5/8 inch wire rope, the required curvature of the load would be 12.5 inches. This criteria was not met since the I-beam around which the rope was wrapped was less than 12 inches. There are no other values, correction factors or tables in the MMI which provided the rated capacity of the rigging configuration. The inspector learned through a discussion with the instructors of the "Rigging Fundamentals" course that required D/d ratios are not taught to riggers nor is there any rigging selection techniques taught for lifts which involve I-beams. This training program weakness will be tracked as an Unresolved Item (259, 260, 296/86-32-09) pending further evaluation. It should be noted that the lift performed on September 26, 1986, to lower the bridge back onto its rails was properly performed with a nylon sling which has no D/d requirements.

With regard to the adequacy of MMI-334, Refueling Platform Inspection and Repair, the licensee is reviewing this procedure to determine needed changes. At issue is a general preventive maintenance instruction which includes such actions as visual inspections, lubrication, fastener tightness checks, retaining device checks, clearance checks and adjustments and which also allows corrective maintenance actions to be performed under no more specific guidance than "repair or replace as necessary", therefore relying on the skill of the craft. The licensee additionally considers that training programs need to stress the mechanisms to be used by craftsmen when the need for corrective maintenance is identified during other routine maintenance activities. The proper procedure is to stop the work, write another MR which describes the problem found, how discovered and action required. This way, maintenance history records can be used to accurately trend recurring defects. These issues will be tracked as an Inspector Followup Item (259, 260, 296/86-32-10) to assure a proper revision to MMI-34.

The inspector conducted a general tour of the facility on September 30, 1986, to identify any potential problems with rigging equipment. Some hooks were identified that did not have safety latches although MMI-102 lists this as a criteria for rejection. One defective equipment tag was attached to the rigging cage at Elevation 639 on Unit 2 which was applicable to all rigging within the cage. It is questionable whether this practice complies with plant requirements. Additionally, the rated capacity table for wire ropes which is posted at Elevation 639 on Unit 2 was in error in that the line of ratings corresponding to 1/2 inch ropes is labeled 1/4 inch. This error could possibly lead to use of an underrated rope. These deficiencies will be tracked as as Inspector Followup Item pending correction by the licensee (259, 260, 296/86-32-11).

10. Review of Special Reports (90713)

On June 17, 1986, the NRC requested that TVA make available for NRC review copies of all Institute of Nuclear Power Operations (INPO) generated documentation pertaining to TVA which are in TVA's possession. The purpose of the NRC review is to provide additional assurance that all major safety issues have been adequately addressed as part of the NRC's ongoing review of TVA. In a letter dated September 11, 1986, from R. L. Gridley to H. R. Denton, TVA informed the NRC that the requested INPO documents had been accumulated and were available for review at each nuclear plant site. This letter also stated that the special arrangements regarding NRC review of INPO documents would be terminated after December 31, 1986. The resident inspectors reviewed a sampling of the INPO documents related to the Browns Ferry Nuclear Plant and TVA Corporate including the most recent Browns Ferry annual evaluation performed on January 20-27, 1986, (in first draft form, dated April 3, 1986) and the Corporate evaluation, first draft dated December 30, 1985. The inspectors confirmed that no previously unidentified safety issues were contained in the documents reviewed.

11. Design Baseline and Verification Program

The inspector reviewed the procedure which implements the design baseline and verification program. This procedure was approved on July 7, 1986.

The Browns Ferry Design Baseline and Verification Program will be used to provide the required level of confidence that the plant is ready for restart.

This program has been developed by the Division of Nuclear Engineering to reconcile design control issues described in several evaluations and audits as follows:

- Comprehensive review of the existing program.
- Studies/evaluations performed by TVA and others.
- Interview of key personnel within the TVA organization.
- Two reviews performed by Gilbert/Commonwealth (G/C) at Sequoyah.
- The Corporate evaluation conducted by INPO.

These program objectives will be met in two phases. The pre-restart phase of this program will be devoted to compiling data from tests and walkdowns, performing system safety evaluations, implementing and testing required restart modifications with corresponding changes to control room drawings, defining actions to resolve document discrepancies, and preparing a report documenting the results for DNE management approval. The post-restart phase will include implementation of the remaining modifications to the systems not required for restart, completion and revision of the design criteria documentation including the criteria to support the post-restart modifications, completion of system safety evaluations not required for restart, implement corrective actions to other safety systems as required, formal revision of control room drawings (i.e. configuration control drawings), and issue post-restart design change supplements (DCSs).

In conjunction with this review, the status of the system walkdowns was reviewed. Although this program has been ongoing under different groups and staffing levels, no drawings have been completed. During a TVA audit of the program, numerous deficiencies were discovered and the program stopped. The procedure for conducting a system walkdown was redone and completed 8/28/86. The procedure is Site Director's Standard Practice SDSP 9.6-01, Mechanical System Walkdowns. This procedure provides requirements to the Field Engineering Section for the mechanical system walkdown (baseline) program necessary to provide the Baseline Evaluation Section verified as-constructed drawings for evaluation with respect to design criteria and licensing requirements, and to support upgrade of plant Operating Instructions for plant restart.

Since this program feeds numerous other programs, completion of the walk-downs is essential for completion of other programs.

12. Annual Emergency Drill

On September 24, 1986, Browns Ferry conducted its annual radiological emergency preparedness drill. The resident inspectors observed plant actions during the drill from several plant areas including the Technical Support Center, the unit control rooms, and the turbine and reactor buildings. Several items of concern were noted and are listed below:

a. All of the phones assigned to the NRC in the Technical Support Center (TSC) were not working. There is no priority on the TSC phones as they are shared with other site areas and are normally busy. The red phone in the TSC does not ring when a call is placed from another location. Addition of this feature would allow monitoring of calls without continuous manning of the phone. This is a repeat comment from last year.

b. The communication of reactor water level was confusing due to the usage of three different instruments for tracking. The training in this area continues to be deficient. One instrument (-100 to +200) indicated a reading in the scenario but should have indicated off scale high with the recirculation pumps operating. This is a repeat comment from last year.

c. The initial actions by the TSC appeared slow to assess key plant parameters such as water level, etc. The Health Physics status board was updated at 0810 with survey information but the plant parameters were not listed until 0844. Also, there is no real time tracking of key plant parameters in the TSC which may delay the decision making process.

d. Participation in the drill by some key managers was not observed. The maintenance, technical services, and unit three superintendents were missing or positions filled by subordinates.

e. There were no steam suits on site. This drill involved an entry into the reactor building to identify a steam leak. In a real situation, a readily available steam suit would have been most beneficial in isolating a leak.

f. Operating Instruction (OI) 84, Containment Atmosphere Dilution, was missing in the TSC.

g. None of the Radiological Control Instructions were available in the TSC or Control Room.

h. The abnormal section of OI-65 for the Standby Gas Treatment System (SBGT) for monitoring operation was not observed until questioned by the inspector.

i. The SBGT system initiated at 0800 but the status of this system was unknown for over an hour to TSC Managers. Possibly a scenario problem although it was indicated running on the data sheets.

j. Health Physics did not consider use of the plant robot for entry into the reactor building to take radiation readings.

k. The TSC was misdirected from their initial assessment of fuel failure by Chattanooga personnel which believed the situation normal for leaking fuel.

l. Various stack release rate information was placed on TSC status boards without an understanding by TSC Managers if the information was normal or abnormal. Writing normal expected rates in parentheses may be beneficial and avoid confusion.

m. The decision to do a controlled shutdown of Unit 1, which was operating at full power, was not made until 0920. Due to numerous common systems, a more timely shutdown might be considered. In fact, the Unit 1 would probably in reality scram out on Main Steam Tunnel High Temperature due to loss of ventilation.

n. The ratio of Operations to Health Physics Personnel in the TSC was too low. Health Physics people were overflowing into space designated for the NRC with air sampling and counting stations. Due to significant number of H.P. personnel in the drill, the H.P. section appeared to control the events.

o. Shift Engineer transfer of Emergency Director to Plant Manager was not announced in Control Room. Unit Operator asked about 15 minutes after the transfer was made, "Who was Emergency Director?"

p. The Shift Engineer was not informed of plant status initially since at 0845 he ordered one loop of RHR to be put in torus cooling. In fact all loops were put in torus cooling at 0808 by the Operator. This may have been an interpretation error on the part of the Operator on EOI requirements. RC/Q-3 should be clarified.

q. Communication in Control Room was not formal. Shift Engineer made several announcements to Control Room Personnel that went unacknowledged (and probably unheard by Operator on phone or communicating with other people). Important annunciators were not announced.

r. The Operators did not check off the EOI steps which made it difficult to follow Accident response progress or determine current status of plant.

s. TSC Communicator was involved in discussion with Operator about whether they were or were not in RC/Q EOI. This is not his job; the Operator should resolve problem with Senior Reactor Operator.

t. Health Physics, as directed by TSC, held up Operators' entry into Reactor Building by two hours until air sample taken to determine if potassium iodide tablets needed to be taken. This is despite order from Shift Engineer to go in and line up instrument air to drywell air header due to a loss of drywell air. Overall, Operations had lost control to Health Physics even though dose rates were relatively small (3R/hr in spots). This type of delay in a real accident situation is unacceptable.

u. It took too long for Operators to realize that SDIV vent/drain valves were leaking by. All briefing scenario sheets indicated that containment was intact. This was not the case with scram valves leaking by and steam coming from a leak in the Reactor Building.

v. Health Physics was late in setting up control point at Unit 3 Reactor Building entrance. Public Safety Officer would not move from Unit 3 turnstile area and remained inside C-Zone when control point was finally established. They simulated dressing out the Public Safety Officer but should have required dress out to see how security requirements might adversely impact on plant operations per the Davis Besse Syndrome.

w. Data from Local and Perimeter Environmental Monitors was not used by Radiological Assessment Personnel. This can be used as confirmatory measurement for calculated doses and is especially important to detect unmonitored releases.

x. TSC status board did not indicate that ADS was inoperable. This was because an illegible note on the Check-List stated "no manual control". This section of checklist was never properly filled in per instructions. These instructions stated that a check mark be used to indicate operability and if a system is inoperable, include reasons in the remarks section. No comments were made in remarks section.

y. Control Room logs were messy, illegible in parts and missing information. Reconstruction of events using the logs would be impossible. Operators generally had spare time during some points of the drill and could have updated logs. The Shift Technical Advisor as a minimum should have had very detailed loops since he performed no useful function otherwise (as could be discerned by the inspector).

z. Operations at one point degraded to discussions about deviating from EOIs, bypassing various interlocks, defeating other protective equipment. In general the inspector worried that Operations was being too creative in correcting the situation. Symptom-based EOIs were designed for such situations and if EOI alternatives have run out, maybe they need to be looked at again to include an additional layer of alternatives.

aa. Shift Engineer should have the wireless phone for his use to keep from being tied down. The Control Room TSC Communicator tied up the wireless phone for the duration. He probably has the least need for it.

bb. Use of EOIs, GOIs, OIs, IPs, ARPs were very difficult; Operators lost track of which OI, EOI, GOI etc. they were in and when they exited. Reactor Operator's desk was extremely cluttered with procedures.

The above listed items were made known to the licensee and will be tracked as an Inspector Followup Item (259/86-32-12).

13. Followup of IE Bulletin 84-03 (92703)

The licensee's August 31, 1984, response to IE Bulletin 84-03 provided a summary of the licensee's evaluation of the potential and consequences of a refueling cavity water seal failure for the Browns Ferry Plant.

The water seal between the reactor cavity and the drywell is provided by the refueling bulkhead, the drywell to reactor well bellows seal, and the refueling bellows seal. The seal boundaries at Browns Ferry are all steel and are either welded bolted or held in place by hydrostatic pressure, such that they cannot be displaced. Gross failure of the refueling bellows or the drywell to reactor well bellows would require structural failure of welded steel passive components and is considered highly unlikely.

In conclusion, the licensee stated that the event described in the subject bulletin has little applicability to Browns Ferry. IEB 84-03 is closed.

14. Accelerated Operator Requalification Training Program Assessment

From August 11-13, 1986, and September 30 - October 1, 1986, a regional inspector was on site to assess the classroom portion of the licensed operator accelerated requalification training program. Additionally, he inspected the adequacy of the corrective actions for violation 259, 260, 296/84-24-01 and conducted an initial assessment on Emergency Procedure training, per the Resident Inspector's request.

The Training Department is continuing to provide intensive instruction to the Browns Ferry operators. The inspector observed classroom instruction in reactor theory, thermodynamics, and emergency operating procedures. The instruction was pertinent, comprehensive, and technically accurate. Contract instructors from General Electric and General Physics are supplementing the TVA staff; they are qualified and competent.

As was the case with the first group of operators to complete the program (February - June 1986), there were numerous concerns expressed to the inspector. Foremost of the operators concerns centered on the depth of the coursework in the thermal sciences (thermodynamics, heat transfer, and fluid flow). The training in this topic requires a familiarity with mathematics (at the algebra level) and investigates engineering-type topics. Since both this depth and breadth of knowledge has been instituted since TMI, many of

the Browns Ferry operators have minimal previous training. The concern about the rapidity of the training and the depth required is understandable; it also is one of the root causes of the poor performance of Browns Ferry operators on the November, 1985 examination. Training Department management has realized the significance the training in this area and has instituted an additional evaluation criterion for it; to be considered proficient, an operator must have an 80% average in the reactor theory and thermal science topics. This criterion is both appropriate and realistic, and should not be considered excessive by the trainees. Nevertheless, at the end of Week 5, eight of the 22 operators who started the program had been removed for academic deficiency.

Operator interviews revealed two additional concerns: first, the learning objectives developed to define those areas of the training program where they should concentrate their efforts are not being used by the trainees to complete their studies; second, the operators morale is suffering from their concern over the new utility evaluation criteria. While there is no requirement to use learning objectives, failure to do so only hampers an operators change for success. The site management was made aware of both of these operator concerns and encouraged to take corrective action to alleviate these concerns.

As a result of the first examination, a generic weakness concerning the instruction on Technical Specifications was identified. The Training Department has developed a curriculum with specific Learning Objectives to correct this problem. It was reviewed by the inspector and should provide for much improved performance in this area.

The performance of the Control Room personnel during the Emergency Plan exercise (September 1986) raised questions concerning the adequacy of the training the operators had received on the new symptom-based Emergency Operating Procedures. Of the nine operators on shift during the exercise, only two had received training in this area; thus the apparent lack of familiarity with the new procedures. Utility plans for completing operator training prior to power operations appear adequate. Commission evaluation via simulator examination will continue.

The corrective actions instituted by Browns Ferry management for violation 259, 260, 296/84-24-01 were reviewed for adequacy. All three corrective actions detailed in the TVA letter, dated October 22, 1984, have been fully implemented. They are appropriate for the violation noted, and should preclude a recurrence of the problem. Violation 259, 260, 296/84-24-01 is considered closed with this report.

15. Regulatory Performance Improvement Program (RPIP)

In a letter, R. L. Gridley to Dr. J. N. Grace, dated March 18, 1986, TVA requested that Confirmatory Order EA 84-54 issued July 13, 1984, imposing the RPIP, be closed and that any remaining open item be included in the Nuclear Performance Plan (NPP). TVA's Nuclear Performance Plan - Volume 3 for Browns Ferry Appendix A discusses the status of certain RPIP items.

The responsible section chief and the senior resident inspector reviewed the remaining RPIP open items with a licensee representative. The current status is shown below:

Short Term

Item I-3.4 (84-SC-21)	Continue with procedure and system training for craft personnel. Specialized training procedure BF-PMI-4.3, specifies "retraining as needed" for foremen and general foremen. Procedure BF-PMI-4.3 should specify frequency of retraining. (Open)
Item I-3.5 (84-SC-22)	Assign system responsibility to section engineers and provide specific training to each. Specialized Training Procedure BF-PMI-4.3 specifies retraining as needed for section engineers. Procedure BF-PMI-4.3 should specify frequency of retraining. (Open)
Item I-3.7 (84-SC-24)	Provide training in preparation of safety evaluation techniques to those individuals assigned such duties. Specialized Training Procedure BF-PMI-4.3 does not require this training for licensing staff or all other engineering processing Unreviewed Safety Question Determination Evaluation (USQD). (Open)
Item I-3.8 (84-SC-25)	Ensure each employee understands his/her responsibility regarding quality and compliance. Procedure BF-PMI-4.3 requires retraining for foremen every year and all others every two years. Training reported 1075 persons (54 classes) had completed retraining in 1986. (Open)
Item I-4.5 (84-SC-31)	Review and revise appropriate documents to define and describe the new organization. Discussed in NPP Section II.2.4. (Open)
Item I-4.11 (84-SC-37)	Establishment of Site Independent Safety Engineering Group Discussed in NPP Section II.1.2.7.5. (Open)

Long Term

Item II-1.1 (84-SC-58)	Vendor Manual Upgrade Program

Discussed in NPP Section II.2.4. (Open)

Item II-1.2 (84-SC-59) Evaluate for closeout those items that Nuclear Power has indicated corrective action is complete.

Inadequate corporate procedures - to be included in NPP.

Technical training on preparation of "work plans". (Open)

Item II-2.1 (84-SC-60) Configuration task force activities, workplan backlog, as-constructed drawings and temporary alterations.

Discussed in NPP Section III-2. (Open)

Item II-3.1 (84-SC-61) Rework and upgrade upper-tiered procedures.

Discussed in NPP Section II.2.4 (Open)

Item II-3.2 (84-SC-62) Management observation of workability of procedures in the field.

Insufficient evidence of management field observations and objective results of improvement use and useability of procedures. (Open)

Item II-3.3 (84-SC-63) Assign an assistant operations supervisor solely to procedure revision and development.

This item is incorporated in II.3.7. (Closed)

Item II-3.4 (84-SC-64) Define the method and limitation imposed to revise procedures on the spot.

Procedure BF-PMI-4.3 does not require or specify retraining frequency. (Open)

Item II-3.5 (84-SC-65) Revise modification control procedure to assure completion of of work, post modification testing, configuration control, instruction and training.

New procedure being developed. (Open)

Item II-3.6 (84-SC-66) Programmatic improvement plans in areas of Health Physics, Security, QA, Assessments, Chemistry, and Fire Protection.

Discussed in NPP Section II.6.1, II.7, II.2.5, II.2.6, II.6.2, and III.5. (Open)

Item II-3.7 (84-SC-67) Plant procedures upgrade activities.

Discussed in NPP Section II.2.4. (Open)

Item II-6.2 (84-SC-74)	Procurement Procurement problems are still being evaluated to determine required actions. Plans and procedures being developed. (Open)
Item II-9.1 (84-SC-78)	Assign trained individual to conduct daily review of control room and other operational activities. Log books are not being reviewed daily in all case and log books lack adequate entries to document and/or reconstruct events as they occurred. (Open)
Item II-9.2 (84-SC-79)	Develop procedure to verify status of abnormal valves. Procedure being revised. Question use of "fourth period student" to perform the verification. (Open)
Item II-9.6 (84-SC-83)	Implementation of recommendation from GE contractors of NSSS operation. Discussed in NPP Appendix B.
Item II-9.7 (84-SC-84)	Implementation of recommendations from SAI Corporation on Technical Specification. Discussed in NPP Appendix B.
Item II-9.9 (84-SC-86)	Utilize outside contractor to reevaluate administrative burden on the plant. Reevaluation of administrative burden on operation staff is still required to ensure objectives have been met. (Open)