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

Report Nos.: 50-259/88-32, 50-260/88-32, and 50-296/88-32

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 1, 2, and 3

Inspection at Browns Ferry Site near Decatur, Alabama

Inspection Conducted: October 1 - 31, 1988

Inspector: for D. R. Carpenter, NRC Site Manager — 2/1/89 Date Signed

Accompanied by: C. Brooks, Resident Inspector
E. Christnot, Resident Inspector
W. Bearden, Resident Inspector
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Approved by: W. S. Little, Section Chief, Inspection Programs, TVA Projects Division — 2/1/89 Date Signed


SUMMARY

Scope: This routine resident inspection included the areas of operational safety verification, surveillance observation, modifications, system return to service, reportable occurrences, restart test program, followup of NRC Bulletins, followup of open inspection items, and licensee action on previous enforcement matters.

Results: One violation with two examples, were identified:

259, 260/88-32-01, Failure to follow procedures while tagging out components for maintenance - Example 1 (Paragraph 2.b.)

259, 260/88-32-01, Failure to follow procedures while performing a surveillance test - Example 2 (Paragraph 3.b.)

One inspector followup item (IFI) was identified:

259, 260, 296/88-32-02: Review of system 82, diesel Generator, RTP results. (Paragraph 7.b.)

The violation, and the IFI described above are required to be resolved prior to the restart of Unit 2.

The program for system return to service has not totally met the expectation of the NRC with regard to meticulous attention to detail and thoroughness of open item resolution. Followup inspection activity will be performed during future NRC resident inspections. (Refer to paragraph 5 for details.)

In paragraph 5.c., a concern is documented regarding operator access to local control panels. The NRC considers the licensee's response to this concern to be thorough, prompt, and well directed.

Paragraph 2.a. documents examples of minor administrative errors discovered in the Temporary Alteration Control Program. These omissions, although not constituting actual uncontrolled temporary alterations, are examples of the type of errors that could occur if a large future backlog were allowed to redevelop. The NRC inspector noted that the licensee had continued to make progress in the ongoing program to continue to reduce the current backlog.

REPORT DETAILS

1. Licensee Employees Contacted:

   S. White, Senior Vice President, Nuclear Power
   C. Fox, Vice President and Nuclear Technical Director
   *J. Bynum, Vice President, Nuclear Power Production
   *C. Mason, Acting Site Director
   *G. Campbell, Plant Manager
   H. Bounds, Project Engineer
   *R. McKeon, Assistant to the Plant Manager
   *J. Hutton, Operations Superintendent
   *R. Laverne, Maintenance Superintendent
   *D. Mims, Technical Services Supervisor
   G. Turner, Site Quality Assurance Manager
   *P. Carier, Site Licensing Manager
   *J. Savage, Compliance Supervisor
   A. Sorrell, Site Radiological Control Superintendent
   R. Tuttle, Site Security Manager
   L. Retter, Fire Protection Supervisor
   H. Kuhnert, Office of Nuclear Power, Site Representative
   T. Valenzano, Restart Director

   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 Exit Interview Attendees

   *D. Carpenter
   *E. Christnot
   *C. Brooks
   *W. Bearden
   *A. Johnson

   *Attended exit interview

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

2. Operational Safety Verification (71707)

   The NRC 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. Inspection 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 instrument operability; temporary alterations in effect; daily journals and logs; stack monitor recorder traces; and control room staffing. This inspection activity also included numerous discussions with operators and supervisors.

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

a. Temporary Alteration Control

An NRC inspector reviewed the temporary alteration change form (TACF) file located in the main control room area, and noted that the number of open and outstanding Unit 2 and common TACFs was continuing to decrease in accordance with the schedule published by the licensee as part of an ongoing management program. However, the NRC inspector noted that the three TACFs listed below had an indeterminate status. Although the TACFs were no longer present in the Unit 2 TACF file and the inspector believed them to be inactive, no closure dates were entered in the closure column of the TACF index.

| TACF | Subject |
|---|---|
| 2-83-029 | Steam packing exhauster |
| 2-83-030 | HPCI steam packing exhauster bypass |
| 2-83-031 | Test gauge on FE-32-75 to monitor leak in drywell control air system |

The inspector brought this issue to the attention of licensee management. In response, the licensee verified from other documentation and by actual hardware walkdowns that the TACFs in question were no longer installed in the plant. The Unit 2 TACF index was subsequently updated to reflect that the TACFs were closed. After further evaluation, the licensee determined that the missing dates were an oversight and that each of the TACFs had been closed out during 1983. The omission had not been detected earlier because all outstanding open TACFs (pre-1984) were closed in 1984 and reissued under a new numbering system which included a new index. A large number of open TACFs existed at that time and the pre-1984 portion of the index contained many entries identifying TACFs which had been closed and reissued under new numbers. The NRC inspector considered that the omission was an isolated occurrence with no safety significance. However, it was recommended that the licensee review all TACF files to verify that no other similar omissions existed that would constitute an uncontrolled temporary alteration to the facility. The licensee agreed to perform an audit to verify that no other problems existed. This issue will be followed up during future resident inspector coverage.

b. Inadvertent Removal From Service Of Wrong Component

On October 16, 1988, diesel generators "A", "B" and "D" for Units 1 and 2 were out of service for maintenance. One of the air compressors ("B" compressor) supplying starting air for the operable "C" DG was out of service. Two EECW pumps were operable as required by Technical Specifications (TS) supplying cooling water to the eight Units 1, 2 and 3 DGs.

On October 16, 1988, the reactor operators were directed to tag out the "A" Diesel Air Start System air compressor for the Units 1 and 2 diesel generator "B". However, the "A" air compressor for the Unit 1 and 2 (DG) "C" was tagged out instead. Independent verification of the tagging was not performed and the fact that the compressor had been tagged out on the wrong DG was not detected until an alarm was received which indicated that the Units 1 and 2 DG "C" had low starting air pressure. This resulted in DG "C" being technically inoperable according to Technical Specifications. Loss of the four DGs resulted in EECW pump B-3 being TS inoperable. With the B-3 pump inoperable, only one EECW pump, D3, remained TS operable. In this configuration, with only one EECW pump operable rather than two required by TS, all eight diesel generators became TS inoperable. With no fuel in the vessel or fuel handling in progress, TSs did not require any operable DGs. SDSP 14.9, Equipment Clearance Procedure, requires component positioning and tagging to be as provided on the clearance sheet. SDSP 14.9 and SDSP 3.15, Independent Verification, require an independent verification of the position of the component. The independent verification must be completely separate and independent of the initial alignment, installation, or verification. SDSP 3.15, Attachment A, Systems and Components Requiring Independent Verification, lists System 86, Diesel Air Start System, as requiring independent verification. The failure to tag the correct component and perform an independent verification when tagging out the Diesel Air Start System air compressor was a violation of TS 6.8.1 for failure to follow SDSP 3.15 and SDSP 14.9, and was identified as the first example of Violation (VIO) 259, 260/88-32-01.

One violation was identified in the Operational Safety Verification program area.

3. Surveillance Observation (61726)

The NRC inspector observed and/or reviewed the surveillance instructions (SI) discussed below. The inspection consisted of a review of the procedures for technical adequacy and conformance to TS, verification of test instrument calibration, observation of the conduct of the test, confirmation of proper removal from service and return to service of the system, and a review of the test data. The inspector also verified that limiting conditions for operation were met, testing was accomplished by qualified personnel, and the surveillances were completed at the required frequency.

a. Fire Protection Surveillance Discrepancies

An NRC inspector accompanied licensee fire protection personnel during the performance of 0-SI-4.11.A.5, High Pressure Fire Protection Valve Position Verification. During the SI performance, the inspector identified the following discrepancies:

- Battery Room 2 sprinkler isolation valve 2-26-1358 was not included in the valve lineup, although the same valve for battery Room 1 was included.

- Hose Station 2-26-807A had a connection wrench attached with a chain that was too short to allow use.

These items were discussed with licensee management and it was agreed that the issues would be investigated and corrective action taken. Resolution of this issue will be reviewed during future resident inspection coverage.

b. Performance of Incorrect Step in Surveillance Instruction

On October 17, 1988, an unplanned Engineered Safety Features (ESF) actuation occurred while performing 2-SI-4.2.A-10, Reactor Building and Refueling Floor Ventilation Radiation Monitor Calibration and Functional Test. After performing step 7.6.110 of the SI, the technician turned to page 38A instead of page 38 and performed step 7.6.111.6 instead of step 7.6.111.1. Step 7.6.111.6 was the step to reset the radiation monitor for the reactor zone exhaust. When step 7.6.111.6 was performed without first performing steps 7.6.111.1 thru 7.6.111.5, an ESF actuation occurred. Failure to perform the SI in the proper sequence was identified as a second example of VIO 259,260/88-32-01.

One violation was identified in the Surveillance Observation Area.

4. Modifications (37700)

An NRC inspector followed the licensee's ongoing work associated with Engineering Change Notice (ECN) E-2-P7131, which was related to NUREG-0737, Item II.F.2. This modification was to reroute the Unit 2 reactor water level reference legs, in order to reduce the routing of the reference legs inside the drywell. This would minimize the potential of erroneous reactor water level indications in the event of post-accident boiling in the reference legs. When completed, the modification will satisfy the actions identified in Generic Letter (GL) 84-23, as presented in the TVA Nuclear Performance Plan (NPP).

Four reactor vessel water level lines were being rerouted outside of the drywell. Two were to be routed through existing penetration X-17, an abandoned Residual Heat Removal (RHR) system penetration containing capped piping. The other lines were to be rerouted inside the drywell through the debris screen into existing 18 inch diameter containment atmospheric dilution (CAD) ducting. The completed ducting with the two lines were to exit the drywell through existing penetration X-26. The existing reactor water level sensing line penetrations (X-28A, X-28D, X-29A, and X-29D) were to be capped.

The inspector reviewed the documentation associated with the ECN, including the licensee safety evaluation, and accompanied the system engineer on a walkdown of the ongoing work. No problems were noted with the ECN documentation or the observed work.

The activity inspected in this area appeared to be effective with respect to meeting the objectives of the NUREG-0737 modification. However, at the time of the inspection, the work was not yet complete. Further review and evaluation will be performed during future reporting periods as part of the normal NRC resident inspector activities.

No violations or deviations were identified in the modification area.

5. System Return to Service (71711)

In preparation for fuel loading, the licensee was completing a systematic evaluation of known restart issues and deficiencies, establishing prerequisites, and completing specific work required to ensure fuel loading would be conducted in a safe and reliable manner. For each system required by TS to support fuel loading, the licensee was to complete modifications, correct known deficiencies, and complete work requests that would impact the safety function or operability of the system. For those NPP Volume III Special Programs where the discovery and corrective action implementation were incomplete, the licensee was to prepare written justification that system operability was not likely to be impaired by undiscovered deficiencies or unfinished corrective actions.

The NRC review of a sample of the licensee's return to service activities included the following aspects of the program:

The licensee's position papers developed to justify the acceptability of fuel load, given the status of the major NPP programs such as Electrical Issues, Seismic Issues, Instrument Line Slope, and Procedure Upgrades

The scope of systems required for Fuel Load and system boundaries required to be reviewed under the system pre-operation checklist (SPOC) process

System design completion verification as controlled by the Department of Nuclear Engineering (DNE) system acceptance evaluation process

System alignment, status assessment, and operability determination, and

System configuration control and control over special operating conditions following declaration of system operability.

a. System Safety Evaluations

On October 4, 1988, the NRC inspector observed a meeting between a plant system engineer and a DNE system engineer to review the configuration of system 78, Fuel Pool Cooling System, as part of the DNE system acceptance for fuel load per Browns Ferry Engineering Project (BFEP) PI 88-07, Systems Plant Acceptance Evaluation. The engineers reviewed the results of the safety evaluation and an unreviewed safety question determination as part of the return to service process. The plant system engineer identified an apparent contradiction between the system safety functions identified in the DNE safety evaluation and the safety functions described in the Final Safety Analysis Report (FSAR). The DNE safety evaluation identified the spent fuel pool heat removal function as non-safety related, whereas the FSAR described this function as part of the safety design basis. The DNE safety evaluation also described the spent fuel pool water level monitoring, maintenance, and prevention of inadvertent drainage function as safety related, whereas the FSAR described this function as a power generation design basis. The contradiction resulted from the Design Baseline Verification Program (DBVP), which included a reconstitution of design basis criteria documents. These documents were used as the basis of the DNE system safety evaluation. The apparent contradiction led to revision 1 to the safety evaluation, which was submitted by the DNE project engineer to the Plant Manager for use in the SPOC for system 78 on October 18, 1988. This revision stated that the fuel pool cooling function of the system was a safety function but that this safety function was not required for fuel loading. No further justification was included to document why this function could be excluded for fuel load. Subsequently, following completion of the SPOC and establishment of system status and configuration control, the NRC inspector learned that the DNE position on the safety function had not changed and that the restart design criteria still listed the heat removal function as non-safety related. During a meeting on October 21, 1988, the NRC inspector informed the Plant Manager of this problem and expressed concern that something as fundamental as system safety function could be in question at this point in the restart process. This specific issue had not been resolved by the end of this inspection period.

The NRC inspector met with members of licensee management in order to determine management controls over similar contradictions. There was apparently no planned transition for promulgating an effective date for when the DBVP design criteria documents would supersede the FSAR for conflicts such as this. This step was considered necessary by the NRC inspector, since the FSAR update is one of the last activities in the DBVP and a fairly lengthy period may transpire before the FSAR is brought into conformance with the reconstituted facility design basis. Subsequent to this meeting, the DNE safety evaluation was revised to clarify the system safety functions. That revision states that fuel pool heat removal is the system's primary function, but still does not list the function as a safety function (refer to paragraph 10.q of this report for a discussion of the FSAR long term update program, Unresolved Item (URI) 88-02-03). The licensee stated that contradictions of this type would be corrected by conducting a review of the DBVP punchlist to identify all the FSAR changes currently known, and providing this list to all personnel qualified to perform safety evaluations per 10 CFR 50.59. Licensee management further stated that these discrepancies are punchlisted for revision of the FSAR to bring the two documents into agreement. This issue will be tracked along with URI 88-02-03 and must be resolved prior to restart.

Resolution of these issues will be reviewed in conjunction with future resident inspector coverage of system return to service.

b. 10 CFR 21 Reports

The licensee's program for addressing outstanding 10 CFR Part 21 reports for fuel load systems was reviewed. Specifically, activities related to IE Bulletin 88-03, Inadequate Latch Engagement in HFA-Type Latching Relays Manufactured by General Electric, were reviewed. The licensee indicated in their response to this bulletin, dated July 6, 1988, that inspections and any necessary repair or replacement of the relays would be accomplished prior to restart. The NRC inspector observed that this item was not on the licensee's Site Master Punch List (SMPL) for tracking, and confirmed through discussions with licensee management that the activities were not considered to be required for the fuel load systems. The NRC inspector considered this position to be unacceptable, given the age of this issue. General Electric (GE) first made purchasers of the subject relays aware of the potential deficiencies via a November 16, 1987, Service Advice Letter (SAL). The prompt notification requirements of 10 CFR 21 are meaningless if prompt action is not taken by the licensee. Licensee management representatives were informed of the NRC inspector's position, and at the end of the inspection period had not decided how the issue would be resolved. The licensee was expected to evaluate the results of the limited inspection activities that had been accomplished and perform an engineering evaluation of the remaining inspection attributes in order to determine whether a failed latching relay could adversely impact a system required for fuel load.

Due to the apparently excessive time period that this Part 21 report had remained open at Browns Ferry, the inspector requested a listing from the TVA licensing organization of any other 10 CFR Part 21 reports that remained open pending final corrective action. This listing was not available at the end of the inspection period. When this listing becomes available, the inspector will review and assess the effectiveness of the licensee's program for Part 21 report resolution.

Followup on these issues will be part of the continuing resident inspector coverage of system return to service.

c. System Preoperability Checklist (SPOC)

The SPOC package for system 69, Reactor Water Cleanup (RWCU) System, was reviewed. Operability Item Deferral Number 69-1 documented deferral of the approval by the Joint Test Group (JTG) of the restart test results until system operability declaration. The NRC inspector held discussions with the system engineer, the Restart Test Manager, and the Return to Service Manager regarding this deferral and learned that not only had the JTG review of the results package been deferred, but also the performance of the entire restart test for system 69. The test was not just deferred until the declaration of system operability (required before fuel load) but was in fact deferred until after fuel load. The licensee position was supported by an engineering justification attached to the deferral form which concluded that the restart test was not required for fuel load.

The NRC inspector held discussions with licensee management on this issue, and stated that the Restart Test Program (RTP) tests were considered by the NRC to be the foundation for system operability declaration and system return to service. Furthermore, the licensee had stated that all discovery programs of the NPP would be complete at fuel load or a justification would be provided for considering that possible unidentified discrepancies would not impact system operability. The NRC considered the RTP to be an effective means for identifying system operability and functional concerns.

Further discussions were held with licensee management representatives, who indicated that a more cohesive decision making process had generally been used to justify exceptions to RTP testing. Basically, all functions of the systems which must be operable for fuel loading, as identified by the fuel load system boundaries, were to be verified by the RTP. Other system functions which were not required to be operable until just prior to restart might not be confirmed by performance of the RTP. This logic was considered by the NRC inspector to be technically acceptable and applicable to all RTP deferrals except in the case of the RWCU system. The licensee indicated that a review of the need for deferral of RWCU system RTP testing would be accomplished and the RTP completed, if possible, prior to fuel load. Followup on this specific item will be accomplished during future resident inspection coverage.

The NRC inspector reviewed the SMPL entries associated with the return to service of system 78, Fuel Pool Cooling (FPC) System. The contractor recommendations report for system 78 stated that the local control panel (panel 25-16) was inside a contaminated area. This impeded operator access to local control of FPC system pumps and valves. The NRC inspector determined that this condition had not been corrected. When informed, the Radiological Controls Manager reviewed the feasibility of decontaminating the immediate area around the panel. Operations and radiological controls personnel performed several joint plant walkdowns in order to identify other areas which should be decontaminated to facilitate operator access to key plant equipment. The NRC inspector observed examples where this had been adequately accomplished. The inspector judged the licensee's response to this concern to be through and well directed.

In summary, the System Return to Service program had not totally met the expectations of the NRC with regard to meticulous attention to detail and thoroughness of open item resolution. The NRC inspectors concluded that further review and evaluation were required, and that the following weaknesses described above will be reviewed during upcoming inspections.

1) System safety function definition
2) Outstanding Part 21 reports
3) Completion of RTP testing for fuel load functions
4) Review of open contractor recommendations for fuel load systems

No violations or deviations were identified in the area of system return to service.

6. Reportable Occurrences (90712, 92700)

The Licensee Event Reports (LERs) listed below were reviewed to determine if the information provided met NRC requirements. The determination included the verification of compliance with TS and regulatory requirements, and addressed the adequacy of the event description, the corrective action 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 No. 296/82-35: Failure of Drywell Floor Drain Sump Outboard Isolation Valve To Close.

While the licensee was confirming the operability of the water flow integrator from the drywell floor drain sump, the outboard isolation valve failed to close because of a stuck piston in the three-way solenoid valve operator. To correct the problem, the licensee replaced the three way solenoid valves on all three units.

The NRC inspector reviewed and evaluated the completed work plans and concluded that corrective actions were adequate. This LER is closed.

b. (CLOSED) LER No. 260/85-15: Insufficient Voltage To High Pressure Coolant Injection Controls.

During a special test performed on the Unit 2 High Pressure Coolant Injection (HPCI) control circuitry, the licensee determined that the electric governor motor (EGM) control box was not receiving the voltage required to meet minimum voltage specifications for the HPCI turbine controller when the input voltage was at design minimum. The licensee replaced the voltage dropping network feeding the HPCI governor with a 48 volt DC power supply which would meet the voltage requirements for all analyzed conditions.

The NRC inspector reviewed and evaluated the ECN, engineering analysis, and work plan completion and verification form, and considered the corrective action adequate. This LER is closed.

c. (CLOSED) LER No. 259/85-17: Lack of Environmental Qualification for $H^2O^2$ Analyzer Valves.

A licensee design evaluation of the teflon valve seats and valve packing in the $H^2O^2$ analyzers had determined that accident radiation levels would exceed the radiation failure threshold of teflon. The licensee changed the valve stem packing and replaced the valves as applicable.

The NRC inspector reviewed and evaluated the work plan specification and the completed work plans for the valve and valve stem packing replacements. Corrective actions taken were considered adequate. This LER is closed.

d. (CLOSED) LER No. 259/85-50: Failure to Perform Surveillance Instructions.

During an October 1985, licensee management review of surveillance scheduling, the licensee identified eight SIs that were not being performed as required by TS for a unit in shutdown for refueling.

Violation 259, 260, 296/85-57-09 was issued on February 11, 1986, for failure to perform sixteen required SIs, including the eight identified by the licensee (See paragraph 10.c). Closure of the violation closes this LER.

e. (CLOSED) LER No. 259/85-55 and Rev. 1: Open Fire Barrier Penetrations.

During licensee maintenance activities, a spare sleeve penetration in a fire barrier was found to be unsealed.

Violation 259, 260, 296/86-09-03 was issued on May 21, 1986, for failure to include piping fire barrier penetrations in a surveillance program. The violation was closed in NRC Inspection Report 259, 260, 296/87-21, based on appropriate licensee corrective action. This LER is therefore closed.

f. (CLOSED) LER No. 259/86-06, Rev. 1 and Rev. 2: Tornado Missile Protection for Vent Towers.

During a 1986 design evaluation of control bay ventilation modifications, design engineers identified an unanalyzed condition involving tornado/missile protection for equipment located in the control bay vent towers. The design basis evaluation for protecting existing equipment had been previously established, and the results were used to perform a site specific risk assessment. The assessment results indicated that the risk to the equipment was extremely low and that no modifications to the vent towers were required.

The NRC Materials Engineering Branch of NRR reviewed the LER and "Calculations of Probability of Occurrence and Consequences of Tornado-Generated Missile Strike of Safety-Related Equipment in Vent Towers", and all NRC questions were resolved through a series of discussions with the licensee. Corrective actions were considered adequate and this LER is closed.

g. (CLOSED) LER No. 260/86-10, Rev. 1 and Rev. 2: Recirculation Inlet Nozzle Safe End Cracks.

In July 1986, the licensee determined by ultrasonic inspection that all ten of the Unit 2 recirculation system reactor vessel inlet nozzle safe ends were cracked. The licensee replaced the Unit 2 inlet nozzle safe ends and a portion of the associated recirculation system piping.

The NRC inspector reviewed and evaluated the ECN, the specifications for the replacement of the recirculation inlet nozzle safe ends, the work plans, and the completion notifications for the work plans. The corrective actions taken were considered adequate. This LER is closed.

h. (CLOSED) LER No. 259/86-22: Nonsafety Grade Air Actuators on Containment Isolation Testable Check Valves.

During a design review, the licensee determined that nonsafety grade air actuators on containment isolation testable check valves could fail, causing the check valve to open and relieve reactor coolant into piping not designed for reactor system temperature and pressure. The licensee removed the air supplies to the valves to prevent inadvertent actuation during plant operations, and installed quick disconnect couplings to allow easy reconnection for testing during shutdown.

The NRC inspector reviewed and evaluated the work plans and work plan closures. The corrective actions taken were considered adequate. This LER is closed.

i. (CLOSED) LER No. 259/88-01: Unplanned Reactor Water Cleanup Isolation Due to Loose Connection.

The licensee had determined during troubleshooting that the cause of an unplanned RWCU isolation was a loose solder connection in the RWCU temperature indication circuitry. The licensee repaired the solder connection and recalibrated the switch. A brush recorder was temporarily connected for thirty hours to monitor switch behavior. No abnormalities were observed.

The NRC inspector reviewed and evaluated the operator logs and the completed maintenance requests. Corrective action taken by the licensee was considered adequate. This LER is closed.

j. (CLOSED) LER No. 260/88-02: Trip of Reactor Protection System Bus 2B Feeder Breaker Initiates Engineered Safety Features Actuations.

On May 26, and May 27, 1988, the breaker (952) feeding reactor protection system (RPS) bus 2B tripped and caused an ESF actuation. The licensee performed a failure investigation and no root cause could be determined. On May 27, 1988, breaker 952 was replaced with a molded case switch that was previously approved by an ECN and work plan. No trip of RPS bus 2B feeder had occurred since breaker 952 was replaced.

The NRC inspector reviewed the ECN, failure investigation, and work plan. Actions taken were considered adequate. This LER is closed.

k. (CLOSED) LER No. 296/88-02: Unplanned Reactor Water Cleanup System Isolation Due to Personnel Error.

An isolation of the RWCU system resulted from a personnel error, when the temperature trip setpoint knob was accidentally turned during the decontamination of instrument panels.

The NRC inspector reviewed and evaluated the maintenance request and the recalibration of the setpoint from the "as found" value of 52 degrees F to the correct RWCU trip setpoint of 140 degrees F. The NRC inspector also reviewed the training attendance records for the training given to all decontamination crews reminding them to use caution when decontaminating any plant panels. This corrective action was considered appropriate. This LER is closed.

l. (CLOSED) LER No. 260/88-10: Inadequate Procedures Cause Two Cases of Missed Samples That Were Required to Compensate for Inoperable Radiation Monitors.

In January 1988, two similar events occurred involving missed compensatory sampling for inoperable effluent radiation monitors. The licensee determined that inadequate procedures were the cause of both events. The licensee revised the applicable operating instruction and SI to ensure that the chemistry lab would be notified when any required sampling should be initiated or stopped to meet TS requirements.

The NRC inspector reviewed the revised procedures for all three units, and concluded that the corrective actions taken were adequate. This LER is closed.

No violations or deviations were identified in the area of Reportable Occurrences.

7. Restart Test Program (99030B)

The inspector attended RTP status meetings, reviewed RTP test procedures, observed RTP tests and associated test performances, reviewed RTP test results (including test exceptions), and attended selected Restart Operations Center (War Room) and Joint Test Group (JTG) meetings. The RTP activities and associated activities monitored, and status of testing during the period of the inspection, are discussed below.

a. RTP Status and Restart Test Performances

The inspector maintained cognizance of ongoing restart test activities, and monitored particular activities in detail as appropriate. Specific inspection observations are discussed in paragraphs 7.b and 7.c below.

The following information summarizes the status of procedures, tests performed, and the hardware related test exceptions identified by the RTP group, at the time of the inspection:

| | Required for Fuel Load | Required for Criticality | Total |
|---|---|---|---|
| Procedures Issued and Approved | 28 | 15 | 43 |
| Tests completed as of 10/31/88 | 22 | 4 | 26 |
| Completed test approved by the Plant Manager | 20 | 1 | 21 |
| Unresolved Hardware TEs | 18 | 35 | 53 |

The total number of procedures required for fuel load was originally identified as 26, but had been increased to 28. However, discussions with licensee management indicated that this number might later be reduced as a result of the return to service of systems for fuel load.

The following restart tests were in progress during this reporting period:

- RTP-023, Residual Heat Removal Service Water
- RTP-030, Diesel Generator and Reactor Building Ventilation
- RTP-031A, Control Building HVAC (Water Side)
- RTP-031B, Control Building HVAC (Air Side)
- RTP-047, Turbine Generator/Electro-Hydraulic Control
- RTP-57-3, 250 Volt DC Unit Battery
- RTP-06A, Primary Containment Isolation
- RTP-069, Reactor Water Cleanup
- RTP-070, Reactor Building Closed Cooling Water
- RTP-082, Diesel Generators
- RTP-085, Control Rod Drive
- RTP-099, Reactor Protection System
- RTP-ICF, Integrated Cold Functional

The above tests were either in the prerequisite stages, system performance stages, initial RTP group reviews, DNE reviews or final JTG reviews.

b. Diesel Generator Testing

Although it was previously reported in NRC Inspection Report 259, 260, 296/88-28 that field activities involved in test 2-BFN-RTP-082, Diesel Generators, had been completed, further review by the RTP procedures review group indicated that the load reject test of DG 3A had either not been adequately documented or had not been performed. The test was subsequently reperformed. The failure of the DG to trip from 2600 KW, when the output breaker was opened was identified as a test exception. NRC review of the DG RTP results is identified as IFI 259, 260, 296/88-32-02. This issue must be resolved prior to Unit 2 restart.

c. Specific Test Witnessing and Results Evaluation

The NRC inspectors observed and reviewed portions of the performance of 2-BFN-RTP-099, Reactor Protection System. Section 5.6 involved testing the RPS high water level trips in the scram discharge tanks, and Section 5.26 involved cable voltage drop testing to verify that the voltage available at the system components will be greater than the component minimum operating voltage. No deficiencies were identified.

No violations or deviations were identified in the Restart Test Program area.

8. Followup of NRC Bulletins (92703)

a. (CLOSED, Unit 2 only) IE Bulletin No. 84-02: Failure of General Electric (GE) Type HFA Relays in Use in Class 1E Safety Systems.

This bulletin addressed similar failures of GE HFA relays, which had been reported in several GE service reports. The licensees were requested to inform the NRC of their plans, including schedules, for implementing the manufacturer's recommendation in the subject GE reports. The licensee had completed all but the following two items documented in IE Report 88-28:

(1) Completion of the replacement of all normally-deengerized relays in Unit 2

(2) Completion of the replacement of normally-energized relays in Unit 1 Systems required for the startup of Unit 2

The NRC inspector reviewed the completed work requests and computer printout documenting the completion of the relay replacement for the two items. This action was considered adequate to support Unit 2 restart.

This bulletin is closed for Unit 2 only.

b. (OPEN) IE Bulletin No. 85-03 and Supplement I: Motor-Operated Valve Common Mode Failures During Plant Transients Due to Improper Switch Settings.

As requested by action item e. of Bulletin 85-03 and Supplement I, the licensee identified the selected safety-related valves, the maximum differential pressures of the valves, and the program to assure valve operability in letters dated May 13, 1988, September 30, 1986, and May 1, 1987. Review of these responses indicated the need for additional information, which was requested in an NRC Region II letter dated April 1, 1988.

Review of the licensee's response dated July 15, 1988, to the request for additional information indicated that the licensee's selection of the applicable safety-related valves, and the maximum differential pressures of the valves, met the requirements of the bulletin. The program to assure valve operability per action item e. of the bulletin was considered acceptable.

The results of the inspections to verify proper implementation of this program, and the review of the final response required by action item f. of the bulletin, will be addressed in future inspection reports. Resolution is required for restart.

c. (OPEN) IE Bulletin No. 88-03: Inadequate Latch Engagement in HFA-Type Latching Relays manufactured by General Electric Company.

This bulletin was issued as a result of a report from GE which stated that some HFA type latching relays were malfunctioning. The NRC stated that operability of all HFA-151B, -154B, and -154E relays with a manufacturing date code prior to November, 1987 should be inspected. The licensee response to the bulletin committed to inspect, repair, or replace the relays failing the inspection criteria before the restart of each unit. During the period of the inspection, the licensee was completing portions of the inspections on the systems required to support fuel load. An engineering evaluation of the remaining inspection attributes will be accomplished prior to fuel load. (See paragraph 5.b. of this report), therefore, this item remains open.

d. (OPEN) IE Bulletin No. 88-04: Potential Safety-Related Pump Loss.

This bulletin addressed the issue that when two centrifugal pumps are operated in parallel and one of the pumps is stronger than the other, the weaker pump may be dead-headed when the pumps are operating in the minimum flow mode. This could cause excessive pump impeller wear. The phenomenon is manifested at low flow rates because of the flatness of the pump characteristic curve in this range.

The licensee's response stated that verification of the adequacy of the miniflow line sizing for the Residual Heat Removal Service Water/Emergency Equipment Cooling Water (RHRSW/EECW), RHR, and Core Spray (CS) pumps is considered to be a portion of the essential design calculations for Browns Ferry. These calculations are under TVA's Design Calculation Review Program for essential calculations, which are commitment items 78, 78A, and 78B of the Browns Ferry NPP, Volume 3, revision 1. This program is required to be completed prior to restart of Unit 2. The licensee has committed to a supplemental response of 30 days upon review completion of items 78, 78A, and 78B of the NPP program. Based on the above, the NRC inspector concluded that this item is acceptable for fuel load but will remain open pending completion of the licensee commitments. This item must be completed prior to unit restart.

No violations or deviations were identified in the area of NRC Bulletins.

9. Followup of Open Inspection Items (92701)

a. (CLOSED) Inspector Followup Item (259, 260, 296/86-05-08), Questionable Instrument Calibration Techniques for Radiation Monitors.

The original issues associated with this item were inspected and reported in NRC Inspection Report No. 259, 260, 296/86-32; however, a new issue related to the revised SI was detected which prevented closure at that time. The new issue related to the possibility that the SI could cause a reactor trip if performed during power operations. Such a trip could result from a high main steam tunnel temperature condition created because the SI called for isolating the normal ventilation in the reactor zone under test. The SI had been written in this manner in order to avoid inadvertent ESF actuations during the SI by manually initiating the ESF as a prerequisite to the test. The NRC inspector had asked the licensee in September 1986, to reevaluate this approach. The licensee implemented Design Change Request (DCR) D3311, which installed permanent test boxes to allow testing of the radiation monitors with the ESF actuation relays defeated in order to prevent the spurious actions which had occurred too frequently in the past.

The NRC inspector reviewed documentation associated with this modification and inspected the installation of the test boxes in the field. The inspector confirmed that 2-SI-4.2.A-10, Reactor Building and Refueling Floor Ventilation Radiation Monitor Calibration and Functional Test, was revised on September 9, 1988, to incorporate the hardware changes and that the reactor zone normal ventilation system would remain in service throughout performance of the SI. These changes should eliminate the potential for a reactor trip to occur from high main steam tunnel temperatures. This item is therefore closed.

b. (CLOSED) Inspector Followup Item (259, 260, 296/86-06-07), Design Requirements for Instrument Sensing Line Slope.

This item was written to ensure that engineering requirements were established for the slope of instrument sensing lines at the Browns Ferry Nuclear Plant (BFNP). In February 1986, the NRC inspector learned that the only source of requirements for instrument sensing lines was TVA General Construction Specification G-60. G-60 required a target slope of 1-inch per foot with a one-eighth inch per foot absolute minimum. The preface to G-60 stated that the specification was only applicable to Bellefonte Nuclear Plant, therefore leaving BFNP with no requirements. The licensee was using G-60 for work at BFNP since there was no other applicable document. The NRC inspector reviewed Specification Revision Notice (SRN) G-60-1, dated May 22, 1986, which made G-60 applicable to future modifications at Browns Ferry. The inspector concurred that the specifications in G-60 were appropriate for Browns Ferry, and the item is closed.

c. (CLOSED) Inspector Followup Item (259, 260, 296/86-32-06), Deficiencies in Diesel Fire Pump Building

An NRC inspector identified several material and housekeeping deficiencies that existed in the Diesel Fire Pump Building.

The NRC inspector reviewed documentation provided by the licensee to support actions taken to correct the identified deficiencies. Additionally, the NRC inspector conducted a tour of the Diesel Fire Pump Building to observe actual conditions. During the tour, the inspector observed that all previously identified housekeeping and material deficiencies, with one exception, had been corrected. The one exception was the battery mounting rack not being secured to the building foundation. This condition still existed and had been evaluated by the licensee as acceptable. FSAR section 10.11.5.1 stated that the High Pressure Raw Water Fire Protection System is not designed Class I seismic and does not necessarily remain functional in an earthquake. However, a fire in any component of an essential system will not prevent safe shutdown of the reactor because essential components are redundant and meet separation criteria and the plant construction does not easily propagate fires. Portable fire protection equipment is provided for use following an earthquake.

The inspector noted a requirement in Plant Manager Instruction (PMI) 12.12, Conduct of Operations, for a daily tour of the building by an operator during routine rounds. Additionally, several new minor material deficiencies were noted by the inspector, which were pointed out to licensee fire protection personnel accompanying the tour. The deficiencies included damaged piping insulation, deteriorated rubber boots, painted rubber expansion joints, and apparently damaged or unused heat tracing. The deficiencies were documented by the licensee on MRs 902120, 902121, 902122, and 902123. The overall condition of the building was much improved and the NRC inspector agreed with the licensee's evaluation that mounting was not required (the construction of the battery racks was otherwise adequate). Corrective actions taken by the licensee were considered to be adequate. This item is closed.

d. (CLOSED) Inspector Followup Item (260/86-40-05), Material Discrepancies and Housekeeping Problems in the Main Steam Valve Vault.

An attempt was made by the NRC inspector to close this item in April 1988, following notification by the licensee that a cleanup had been performed. The inspector toured the area in April and found conditions still unacceptable (refer to Inspection Report 259, 260, 296/88-10). On October 18, 1988, the inspector made a followup tour of the area and noted a significant improvement. All previously identified items were corrected and no new concerns were detected. This item is closed.

e. (OPEN) Inspector Followup Item (259, 260, 296/86-40-12), Potential For Overpressurization of Residual Heat Removal System Piping.

A modification was installed in order to reduce an excessive pressure drop across a throttling valve in the RHR system. An orifice plate was installed in a section of piping rated at 150 psig and increased the pressure in this section of pipe to an undetermined valve. Although the section of pipe in question (the test return line) was not instrumented during the post modification test, the nearest portion with pressure indication exceeded 300 psig during the test. The NRC identified that the potential for exceeding the pipe design pressure had not been analyzed as part of the modification. This finding was made as part of the Unit 3 modification package review.

The licensee completed the modification on Unit 2 and performed a more extensive post modification test on October 4, 1988. This PMT duplicated the worst condition of both RHR pumps operating with full flow through the orifice, and measured the pressure in the suspect piping. The test results indicated that the worst case pressure increase with the drywell at atmospheric pressure was 137 psig, which was within the 150 psig rating. Additionally, a design calculation was performed by the licensee which resulted in an expected pressure drop of 133 psid across the orifice under maximum flow conditions.

The NRC inspector reviewed the test data and the design calculation and confirmed that these values were appropriately derived. The inspector determined that under normal conditions the piping would remain within its design rating; however, following the design basis LOCA, FSAR Section 14.6.3.3.2 indicates the Torus pressure can be as high as 27 psig. Under accident conditions this piping section pressure could be as high as 164 psig exceeding the design pressure of 150 psig. The licensee's review of the test data did not result in their identifying this problem.

This IFI remains open pending TVA's evaluation of the acceptability of the piping design for the potential accident conditions.

f. (CLOSED) Inspector Followup Item (259, 260, 296/88-05-05), Close out of Restart Test Program Maintenance Work Request.

This item documented a concern identified during the RTP testing of air dampers in the DG rooms. It was identified by the Site Quality Assurance Monitoring Group that maintenance requests (MRs) not being addressed was a site wide problem and not just an RTP problem. The QA inspector initiated Condition Adverse to Quality (CAQR) BFQ 88 0143 to document missing MR's. Although the initial CAQR was aimed at MRs generated as a result of environmental qualification (EQ) walkdowns, a further revision of the CAQR was aimed at the site in general. The NRC inspector will monitor the followup of the CAQR. This item is closed.

g. (OPEN) Inspector Followup Item (260/88-10-01), Main Steam Tunnel Blowout Panel Function Possibly Defeated.

The licensee identified and documented on CAQR 880293 that the Unit 2 Main Steam Tunnel blowout panels were not installed in accordance with the drawings. Specifically, RTV sealant had been used to fill large gaps between the panels and framing. The RTV could act like an adhesive to prevent blowout of the panels at their design differential pressure.

Corrective action was completed by the licensee with the exception that many of the explosive bolts were inadequate and had been replaced on a short term basis with non-explosive bolts. Only the secondary containment integrity function of the panels is required for fuel load. The blowout function needed only to be operable to mitigate a steam pipe break, which would not be possible until after restart. This item was therefore acceptable for fuel load but remained open pending completion of corrective action by the licensee. Final corrective action is required prior to restart.

h. (CLOSED) Inspector Followup Item (259, 260, 296/88-10-03), Lack of Understanding of the Restart Test Program by On-Shift Senior Personnel.

This item documented a concern that on-shift senior reactor operators (SROs) upon returning from extended periods of training were not fully aware of the RTP. The NRC inspector reviewed a memo dated July 13, 1988, from the Operations Superintendent to all operation personnel, in which the purpose of the RTP was clarified. The inspector continued to observe operations personnel, especially senior or shift personnel, and the RTP Test Director's activities, and determined that the RTP program was adequately understood by operations personnel. This item is closed.

One IFI was upgraded to a violation in the area of Followup on Open Inspection Items.

10. Licensee Action on Previous Enforcement Matters (92702)

a. (CLOSED) Violation (259/85-13-03), Failures to Follow Procedures and an Inadequate Procedure During Retest of Control Rod 34-03.

This violation was identified in Inspection Report 259/85-13-03, but was not cited at that time. Subsequently, Violation 259, 260, 296/85-36-01 was issued to address this finding. The violation comprised examples of failure to follow procedures and an inadequate procedure regarding Unit 1 control rod 34-03 maintenance activities. Resolution of this item is addressed through the followup on Violation 85-36-01, which is discussed in paragraph 10.b of this report. This violation is therefore closed.

b. (OPEN) Violation (259, 260, 296/85-36-01), Failure to Follow Procedures and an Inadequate MR for CRD 34-03 Post-Maintenance Testing.

Violation 259, 260, 296/85-36-01 consisted of four examples in which procedures were not adhered to or were inadequate. Three of the four examples pertained to maintenance work done on Unit 1 CRD Module 34-03. The fourth example involved the licensee's failure to perform a safety evaluation in order to determine HPCI system operability with failed-open resistors on HPCI steam line drain isolation valves. The licensee's corrective actions for the fourth example of the violation were being reviewed separately and will be addressed in a future inspection report.

The licensee's response to the violation was provided in a letter to the NRC dated September 27, 1985. The NRC inspector reviewed the licensee's reasons for the violation and their corrective actions.

1) Example 1: Inadequate CRDH Maintenance Request

The first example of the violation dealt with an inadequate maintenance request, MR A126652, which failed to contain functional and post maintenance testing (PMT) requirements as required by the Mechanical Maintenance Instruction (MMI) 28, Control Rod Drive Hydraulic Unit (Repair, Removal, and Replacement). The licensee's reason for the violation was that the foreman failed to follow MMI-28, and that MMI-28 lacked clarity and did not adequately cross-reference applicable sections within the procedure (e.g., testing). MMI-28 was revised accordingly.

The NRC inspector reviewed the most recent version of MMI-28, revision 6, dated August 23, 1986, and verified that appropriate revisions had been made for clarification of PMT requirements. Section 10.3 of MMI-28 provided a listing of PMTs for different areas of maintenance performed on HCU units and delineated the individual responsible for performing the PMT. Also, the NRC inspector reviewed the training attendance record dated November 24, 1987, verifying receipt of training for draft personnel on MMI-28 requirements. The NRC inspector concluded that adequate corrective action had been taken for this example of the violation.

2) Example 2: Failure to Exercise Control Rod within Time Limit

The second example of the violation involved the licensee's failure to exercise control rod 34-03 within the required time limits specified in MMI-28, Section 10.3, and Operating Instruction (OI) 85, Control Rod Drive (CRD) System, Section 3.H.1.e.2.e. Reasons given for the violation in the licensee's response were that MMI-28 and OI-85 required control rod insertion and withdrawal times (i.e. 48 plus or minus 3 seconds) which were too restrictive and were inconsistent with the RTI-5 and vendor recommendation criterion of 40-60 seconds.

The inspector reviewed the revised procedures and verified that they incorporated the acceptance criterion recommended by the vendor. Section 8.8 of the upgraded Unit 2 OI-85, revision 3, provided timing adjustment of control rods within the tolerance of 40-60 seconds. Technical Instruction (TI) 20, Control Rod Drive System Testing, Revision 0, provided the same requirement reflected in Sections 7.3.7.4 and 7.3.8. Also, the NRC inspector verified that the same criterion was provided by the vendor (GE) in the GEK-9585/9586 document. The NRC inspector concluded that this example of the violation had been adequately resolved.

3) Example 3: Failure to Follow Procedure Limits on CRD Pressure and Control Rod Position

The third example of the violation involved the failure to follow procedure OI-85, Control Rod Drive System, Section 3.D.9, in that during withdrawal of control rod 34-03 from the fully inserted position (00 notch position) the drive water pressure was not returned to normal limits before the rod passed the 02 notch position. The drive water pressure was determined to be approximately 50 psi above the normal limits when the rod passed notch position 02. The licensee's reason for this violation was that procedure OI-85 was too restrictive in its limitations of CRD pressure and control rod position, which resulted in an inadvertent failure to follow procedures.

For corrective action, OI-85 was revised to permit drive pressure to remain above normal levels until the 06 notch position is reached. Based on the CRD design, which is of a finger/collet configuration with a traveling distance of 3 inches per step and a normal withdrawal/insertion speed of 3 inches per seconds $\pm$ 20 percent, the inspector agreed with the licensee's position that OI-85 had previously been too restrictive in the limitations on CRD pressure control with respect to control rod notch positions. Section 8.16 of the upgraded Unit 2 procedure (OI-85, revision 3), provided clearer and more detailed instructions on what to do when a control rod becomes difficult to withdraw. Caution statements in OI-85 were changed to return the CRD drive water header differential pressure to between 250 and 270 psid (normal limits) as soon as possible in order to prevent a drive from double notching in a high rod worth region, and to reduce exposure of drive seals and directional control valves to excessive pressures. Furthermore, the licensee incorporated GE's recommendation from the GE contractor recommendations work to use the double clutch method of withdrawing a control rod from notch 00 to notch 02 just before applying elevated drive water pressure. OI-85 for Units 1 and 3 had also been revised, but not through the upgraded procedure process. The NRC inspector expected that the procedure would be revised in the future so as to be consistent with the Unit 2 procedure. This example of the violation was considered resolved.

The violation remains open pending inspection of corrective actions for the fourth example, which concerned HPCI operability.

c. (CLOSED) Violation (259, 260, 296/85-57-09), Failure to Conduct Sixteen Surveillance Instructions During Shutdown and Refueling.

The licensee determined that the root cause included the failure to fully implement TS requirements in plant procedures and personnel error in TS interpretation. The licensee subsequently performed all the SI's not performed on Units 1 and 3, with the exception of SI - , Recirculation Pump Trip Reactor High Pressure, on Unit 1. This SI was not performed because the recirculation pumps were not operating and the reactor vessel head had been removed. Unit 2 SI's were not performed because the fuel had been unloaded and the applicable systems were no longer required by TS.

The licensee corrective action to avoid further violations was to update SI-1, Surveillance Program, Appendix C to accurately reflect the TS requirements for performance of these SI's during shutdowns and refueling. On August 23, 1988, the licensee also updated SI-1 because of issuance of TS Amendments 136 through 144 for the restart of Unit 2.

The NRC inspector reviewed and evaluated the corrective action documented above, and considered it appropriate to support Unit 2 restart. This item is closed.

d. (OPEN) Violation (259, 260, 296/86-25-01), Failure to Follow Procedures (Three Examples).

1) Example A: Fire Protection Sprinklers not Configured in Accordance with Approved Plant Drawings

Example A of Violation 86-25-01 was attributed to inadequate coordination in work plan preparation in December 1976, and lack of a post-modification test in 1977, which resulted in a failure to remove a welded blank in the fire protection line. The condition went undetected until approximately July 14, 1986, when the licensee was prompted to determine why a section of piping in the Unit 3 reactor building could not be flushed during a non-routine flush of the fire protection pre-action sprinkler system for removal of any accumulation of mud or clams. The affected section contained 19 sprinkler heads which were rendered inoperable.

For corrective action, the licensee improved the control and documentation of work plans and control of temporary alterations, and implemented a program requiring the Fire Protection Unit's overview of all modifications and post-modification testing pertaining to fire protection systems. Further, an approximate 30 percent random sample of pre-action system branch lines were selected for a special air test. More than 100 branch lines were tested and no blockages were identified. The testing was completed by November 30, 1986. The licensee's basis for selection of a 30 percent random testing scheme was provided in their supplemental response to the violation, dated March 2, 1987. The supplemental response was reviewed by the inspector and found to be acceptable.

This item is also addressed in LER 296/86-06, which was reviewed and closed in NRC Inspection Report 259, 260, 296/87-20. Subsequently, another inspection was performed in which the inspector reviewed all licensee generated Licensee Reportable Event Determinations (LREDs), LERs and CAQRs/CARs issued after December 1, 1986, in the area of fire protection. No recurrence of a similar event was identified.

The corrective actions taken by the licensee in response to Example A of the violation were considered adequate.

2) Example B: Control Rod Drive Hydraulic Control Units not Installed in the Design Support Mounting Configuration Required by Design Drawings

This example of the violation was the outcome of URI 259, 260, 296/85-25-01, which was closed and upgraded to a violation for failure to have hydraulic control units (HCUs) mounted as required by design drawing 919D615. The inspector found loose bolting, several free-standing HCUs, misaligned channel nuts, and missing washers.

In the response to the violation, dated October 16, 1986, the licensee stated that the violation resulted from poor work practices and inadequate inspections conducted during the plant's construction phase. For corrective action, the licensee stated that they replaced all CRD HCU mounting bolts and that the floor mounting hardware had been installed in accordance with design drawing 919D615.

The inspector followed up on the licensee's corrective actions by reviewing all MRs associated with the inspection, replacement, and torque work done on base mounting bolts. Also, the inspector performed a walkdown of CRD HCUs, in particular those for Unit 2, and verified that hardware installation was in accordance with design drawing 919D615. Vertical back-to-back mounting and horizontal mounting of the HCUs were checked. A number of 3/8 inch and 1/2 inch base mounting bolts, flat washers, and lockwashers were found to be relatively new and properly installed. Work had been completed in late 1985 and 1986. Per the review of MRs A-570904 and A-581910 (for Unit 2) the 1/2 inch bolts were torqued 50 ft-lbs as recommended by Unistrut, and the 3/8 inch HCV back mount bolts were torqued to 19 ft-lbs. For Units 1 and 3, the same work was performed per MRs A-570905, A-706829 and A-592255. The torque work received quality control verifications. No discrepancies were found and the inspector considered the corrective actions to be adequate for Example B of the violation, and the issue is resolved.

This violation remains open pending NRC inspection of the third example, which pertained to CREV mounting details. Resolution of the final violation example is required prior to restart.

e. (OPEN) Violation (259, 260, 296/86-25-06), Failure to Maintain Records of Facility Changes, Including the 10 CFR 50.59 Safety Evaluation.

This violation resulted from a change to plant flood protection features. Originally, flood doors to the Reactor Building and Radwaste Building were normally maintained closed except for personnel and equipment access as stated in the FSAR. In 1981, the licensee changed the normal practice such that the doors were maintained normally open. When questioned by the NRC inspector in 1986, no safety evaluation could be retrieved which would document acceptability of the change per 10 CFR 50.59.

The licensee's corrective action consisted of reevaluating the condition and performing a new safety evaluation. The NRC inspector reviewed revision 2 of the safety evaluation, dated June 18, 1987. The evaluation adequately justified changing the FSAR to reflect the current practice of leaving the doors open. This change was made in Amendment 5 to the FSAR in August 1987. The evaluation further recommended that the Bases for Section 3.2 of the TS be changed to delete the statement, "Plant flood protection is always in place and does not depend in any way on advanced warning." This statement was not accurate under the current circumstances, which required operator action to close the flood doors when required. As of October 18, 1988, this change had not been made. The evaluation additionally recommended that an administrative instruction be developed to ensure that operators close the flood doors whenever the Wheeler Reservoir elevation reaches 558 feet. The plant responded by adding the necessary operator action to Annunciator Response Procedure (ARP) 9-20. The NRC considered this to be inappropriate since the entry condition into the procedure was the actuation of the "Lake Elevation High" alarm which occurs at 564 feet, 6 feet above that at which operator action is required. This item will remain open pending resolution of the above two outstanding deficiencies by the licensee. This item is acceptable for fuel load, because the plant will be in the action statement (shutdown) required by TS upon high water level conditions. Final closure of the item is required prior to restart.

f. (CLOSED) Deviation (259, 260, 296/87-30-04), Failure to Maintain Written Justification for Changes to the FSAR.

In response to this Notice of Deviation, the licensee committed to the following:

- o Review the 1987 annual FSAR update to ensure proper justification existed for each change and correct any changes which could not be justified by the 1988 annual update.
- o Revise the administrative procedure governing FSAR updates to require formal justification for all changes.
- o Reinstate the commitment to perform a periodic examination of the site surroundings to provide a reasonable representation of area population and land use in the next FSAR.
- o Submit a letter to the NRC describing the program for periodically updating the FSAR chapter which deals with area population and land use.

The NRC inspector reviewed documentation associated with the licensee's commitments and determined that they had been adequately implemented. The licensee's review of the 1987 annual FSAR update detected three changes which could not be justified. The NRC inspector confirmed that these changes had been reinserted into the FSAR in Amendment 6. The NRC inspector also confirmed that generic implications for other TVA facilities had been addressed through issuance of a TVA corporate-wide Office of Nuclear Power Standard. This standard (ONP-STD-6.1.6 Rev. 0, Maintaining and Controlling Safety Analysis Reports) contained specific guidance on periodically evaluating and updating the FSAR chapter dealing with site description, land use, and representation of area population. As a final check on the adequacy of the administrative procedure governing FSAR changes, the NRC inspector selected a sample of 16 changes made to the FSAR in Amendment 6 issued in July, 1988. The licensee was able to provide adequate justification and safety evaluation for all of these changes. This deviation is closed.

g. (CLOSED) Violation (260/87-37-04), Control of Measuring and Test Equipment.

This violation identified the unauthorized adjustment of the zero adjust screw on TVA pressure gauge # E82214, which was being used in a post-modification test performed on instrumentation associated with safety-related systems. A licensee craftsman performed the field adjustment because the gauge was reading off-zero with no pressure applied. The only authorized adjustment of this gauge was during a multi-point calibration procedure traceable to the National Bureau of Standards.

When informed of the event, the Measuring and Test Equipment (M&TE) Coordinator and the Unit 2 Instrumentation and Controls Section Supervisor took immediate action to have the gauge retrieved and tested. Additionally, a memorandum was initiated to inform all instrument mechanics on proper use of gauges.

The NRC inspector reviewed the licensee's response to the violation, dated January 15, 1988, and determined that the stated corrective actions should be adequate to prevent recurrence. The licensee evaluated the circumstances associated with the violation and determined that field adjustment of the gauge zero, although acceptable on some types of M&TE pressure gauge, was not acceptable on those manufactured by Wallace & Tiernan, Inc. The improper adjustment of the gauge was attributed to inadequate training. Browns Ferry instrument mechanics had since received additional training on proper actions using Wallace & Tiernan gauges. Special caution tags have been prepared for use with any M&TE pressure gauge that can not be zero adjusted in the field. Additionally, SDSP-29.1, Control of Measuring and Test Equipment, was revised to include the requirement to attach the special caution tags to all associated analog and digital M&TE pressure gauges. This item is closed.

h. (CLOSED) Violation (259, 260, 296/88-05-01), Failure to Control the Issuance of Documents and Changes.

This violation identified the failure by the licensee to properly control revisions to TACFs. Revision 1 to TACF number 3-88-001-111 was not properly reviewed for adequacy, approved for release, or properly distributed. Similar problems were also noted on other licensee TACFs.

The NRC inspector reviewed the licensee's response to the violation, dated June 24, 1988, and determined that the stated corrective actions should be adequate to prevent recurrence. The licensee has corrected the deficiencies as noted in the original NRC inspection report. Additionally, a licensee review of all open TACFs to verify proper handling and correct documentation was conducted and all identified deficiencies were corrected. There was also an ongoing licensee program to reduce the number of outstanding TACFs, with a goal of zero open for Unit 2 and common systems TACFs prior to restart.

PMI - 8.1, Temporary Alterations, has been revised to clarify the TACF revision process. Requirements have been included to distribute copies of revised TACFs to appropriate organizations. This item is closed.

i. (CLOSED) Violation (260/88-05-02), EECW Check Valve Installed Reversed.

This violation identified the licensee's failure to properly verify correct valve operation by inspection following the maintenance activity performed under MR# 792717. EECW check valve 2-67-659 was removed during MR 792717 and reinstalled backwards, resulting in the inability of the north EECW header to supply cooling water to safety-related components. Existing licensee procedure, BF-3.2, Quality Control Inspection Program, Section 5.2.1, contained examples of activities that should be verified by using QC hold points. Satisfactory operation of a valve following maintenance was one of the examples.

As corrective action, the licensee conducted additional training for maintenance personnel to review the event and emphasize the need to understand and fully implement the appropriate procedures when performing the work. MMI-51, Maintenance of CSSC/Non-CSSC Valves and Flanges, was revised to include the requirement, as step 8.2.2.5, to check for proper valve orientation whenever a valve is reinstalled in a system.

The NRC inspector reviewed the licensee's response to the violation, dated May 25, 1988, and additional supporting documentation which verified the performance of the corrective actions. The licensee evaluated the failure and attributed it to the following causes:

- o Failure of maintenance personnel involved in reinstalling the check valve to follow existing procedural guidelines to ensure the valve was properly installed
- o Failure of maintenance planners for the work activity to include a verification step to check and document valve orientation in the work instructions
- o Difficulty in determining actual direction of EECW flow in piping adjacent to the valve location

The check valve was removed and reinstalled in the proper orientation under MR 886541. Proper orientation was verified by the licensee as part of the corrective action of CAQR BFP 880193. Additionally, an NRC inspector observed the correct valve orientation. The licensee has labeled the EECW piping adjacent to the valve to show actual flow direction. The NRC inspector concluded that adequate corrective action had been taken to prevent recurrence. This item is closed.

j. (CLOSED) Violation (259, 260, 296/88-05-04), Failure to Comply With the PORC Composition Requirements of Technical Specifications and Failure to Maintain PORC Meeting Minutes.

The NRC inspector reviewed the licensee's response to this violation and concurred with actions taken to prevent recurrence. The NRC inspector attended a Plant Operations Review Committee (PORC) meeting on October 18, 1988, and confirmed that the membership complied with the TS quorum requirements. This violation is closed.

k. (CLOSED) Violation (259, 260, 296/88-05-08), Failure to Provide Adequate Training for Craft Personnel.

This violation resulted from an inspector followup of TVA's implementation of the Browns Ferry Regulatory Performance Improvement Program (RPIP) action items. The violation pertained to non-compliance with procedure BF PMI-4.3, Specialized Training, in which a certain number of craft personnel were found to be delinquent in receiving periodic general employee training (GET) retraining, including regulatory compliance.

The inspector reviewed the licensee's response to the violation, dated June 24, 1988, stating the reason for the violation and the corrective actions taken to preclude further violations. The cause of the violation was attributed to inadequate supervisory enforcement of training attendance requirements. Corrective action included the following: (1) Issuance of a memorandum by the Site Director requiring action to correct delinquent GET training; (2) Development by the Training Department of a training schedule to ensure personnel attendance; and (3) Consolidation of retraining in regulatory compliance (RPI 1.383), Introduction to QA/QC (GET 4), and Plant Procedures and Instructions (GET 6) into one course.

The inspector followed up on the licensee's corrective actions by reviewing their memoranda issued to correct GET absences and other delinquencies in the training plan for the new consolidated course, and the training schedule. The licensee implemented the revised training schedule on July 3, 1988, which required retraining on an annual basis. Per procedure, delinquent individuals will be informed in writing, and be required to attend rescheduled GET by the end of the calendar quarter.

The corrective actions should be adequate to preclude recurrence of the violation. This item is closed.

l. (OPEN) Violation (259, 260, 296/88-22-01), Inadequate Corrective Action.

In January 1987, licensee management became aware that four temporarily promoted shift engineers did not satisfy the minimum qualifications delineated in Nuclear Plant Operator Training Program procedure PMP 0202.05. Adequate corrective action was not taken to disposition the issue. PMP 0202.05 required that the candidate for the position of Shift Engineer must pass the shift engineer accrediting examination unless waived by the Chief, Operator Training

Branch, and the Plant Training Review Board or Accrediting Subcommittee. In addition, there were four other permanently assigned shift engineers for whom records could not be found to show that their certification examinations were successfully passed.

In August 1988, the four temporarily promoted shift engineers received and passed the accrediting examination for the Shift Operations Supervisor position and were interviewed by the Site Director. The licensee subsequently searched for plant records to show that the other four permanently assigned shift engineers had passed their accrediting examinations. The examination cover sheets were found on microfiche, and this provided documentation to show that their certification examinations had been successfully passed.

The NRC inspector reviewed the documented corrective action and considered it acceptable to support Unit 2 restart. However, the item remains open pending NRC acceptance and inspection, if applicable, of the formal licensee response to the violation.

m. (CLOSED) Unresolved Item (259, 260, 296/86-06-08), Inadequate Slope on Instrument Sensing Lines.

The NRC inspector closed all aspects of this URI in July 1988 (refer to Inspection Report 259, 260, 296/88-21), with the exception of one instrument line which did not comply with the established minimum slope requirements. The licensee performed an in-depth engineering evaluation of the suspect line, and concluded that the as-built configuration was acceptable. The evaluation assessed the entire length of sensing line from the pressure transmitter to its dead-ended termination in the drywell (the parameter being monitored was drywell pressure). Although, a low point did exist in the line, the geometric configuration would prevent buildup of condensate to more than one-half of the sensing line's internal diameter. Although, this could create an orifice effect in the sensing line, there would be essentially no flow in this situation and therefore no detrimental pressure drop. The inspector interviewed the engineers responsible for the engineering evaluation to ascertain whether corrosion from the standing water in the low point had been considered and to discuss the evaluation in general. The inspector concluded that the evaluation adequately assessed the as-built condition.

This concern was originally identified during the field work phase of the modification, and the licensee demonstrated compliance with the slope requirements or adequately evaluated any nonconformances prior to completion of the modification. Therefore, no violations existed and this item is closed.

n. (OPEN) Unresolved Item (259, 260, 296,/86-28-02), Discrepant Scram Valve Opening Times.

The licensee discovered during the performance of Special Test 86-10 in July 1986, that several scram inlet and outlet valves delayed opening for up to 20 seconds. This item was reviewed in NRC Inspection Report 259, 260, 296/88-16, and the following issues remained open at that time:

1) Acceptance criteria for scram pilot valve timing upon scram air header blowdown should be established. The data already accumulated should be shown to support compliance with this time or followup tests should be performed to demonstrate compliance.

2) Either single rod scram testing prior to plant startup or scram valve time tests prior to plant startup should be accomplished for each scram solenoid pilot valve that has been refurbished in accordance with the GE recommendations in Service Information Letter (SIL) No. 441. This is to ensure HCU operability and to detect any further anomalies.

3) The licensee should check the adjustment of all scram valve opening air pressures which have indicated a potential for noncompliance with the recommended spring tension settings in GE SIL No. 373.

The inspector determined that item 1) would be addressed prior to restart by the performance of the post modification test for the alternate rod injection (ARI) modification, which includes the acceptance criterion of 15 seconds on scram outlet valve opening. Item 2) would be satisfied by the performance of single rod scram testing during the Unit 2 power ascension program. Item 3) had been completed and no values were known to be in noncompliance with the recommended spring tension settings in GE SIL No. 373. This URI was considered by the inspector to be adequately resolved for fuel load but will require additional followup during the power ascension phase.

o. (OPEN) Unresolved Item (259, 260, 296/87-26-03), RHR Pump Nozzle Stress Exceeds Allowables.

The licensee's Engineering Assurance (EA) organization identified a deficiency concerning an assumption by DNE engineers that blanket approval was authorized for nozzle load calculations to result in a 20 percent overstressed condition. The actual requirement was a specific case-by-case justification, analysis, and approval of each condition. The problem was identified during a review of the IE Bulletin 79-14 calculations and was documented in EA Audit 87-13.

The NRC inspector reviewed the licensee's corrective action in response to this finding, and confirmed that the generic implications had been addressed (two additional examples were found) and corrected. Long term corrective action had been addressed through procedure changes and training. The licensee's EA organization verified acceptable implementation of the corrective action and closed this item on August 26, 1988. However, no reanalysis had been performed on the specific calculation where the deficiency was detected. The NRC will maintain this item open pending a reanalysis and/or a case specific justification for the RHR pump nozzle in question. The status of this issue is acceptable for fuel load but will require additional followup prior to restart.

p. (CLOSED) Unresolved Item (259, 260, 296/87-26-04), SNM Control.

This item identified irregularities involving a shipment of Special Nuclear Material (SNM) to another licensed facility. Violation 259, 260, 296/87-29-01, which identified the licensee's failure to perform an adequate physical inventory and follow TI - 14, was subsequently issued to address this issue. Resolution of this item will be tracked by the followup of the violation. This item is closed.

q. (OPEN) Unresolved Item (259, 260, 296/88-02-03), Long Term Corrective Action and Interim Controls for FSAR Deficiencies.

The licensee's Nuclear Safety Review Board (NSRB) had identified that the FSAR was so deficient that it could not be relied upon for the purpose of making 10 CFR 50.59 safety evaluations and Unreviewed Safety Question Determinations (USQD). The problem resulted from inadequate controls over annual FSAR updates for many years. (Refer to NRC Deviation 259, 260, 296/87-30-04 for details on this problem).

The licensee documented this condition on CAQR BFF 870088 and prepared and approved an FSAR Update and Verification Plan (B22 87088827 007) as part of the corrective action for the CAQR. The plan involved a review of many documents, including outputs from the DBVP program, to identify required changes to the FSAR. Additionally, the licensee planned to perform a review to verify the accuracy of substantial statements in the FSAR. The target completion date for this activity was July 22, 1990. During the review period, deviations identified by the program will be identified as CAQR's, if appropriate, and USQD's will be prepared and approved by the PORC and Plant Manager. Issues which are identified as being unreviewed safety questions will require approval by the NRC.

The licensee recognized that in the interim period prior to completion of the long term program, the FSAR could not be relied upon for reviews of changes, tests, and experiments per 10 CFR 50.59. On March 23, 1988, the licensee's Site Director issued a memorandum which detailed the condition of the FSAR and required verification of information by an independent source (such as DBVP design criteria documents) when performing 10 CFR 50.59 screening reviews and safety evaluations. SDSP 27.1 "Evaluation of Changes, Tests, and Experiments" was revised to provide guidance on additional documents to be used for conducting safety evaluations including TS Bases, NRC safety evaluations, licensing submittals to the NRC, Commitments and Requirements Data Base, and NRC regulatory guides as committed to in the QA Manual.

The NRC inspector reviewed the licensee's training lesson plans associated with 10 CFR 50.59 evaluations and found that the training specifically highlighted these concepts. The NRC inspector further sampled some of the more recent "Screening Review Forms for Documenting Applicability of a Safety Evaluation" per SDSP 27.1 to confirm that other sources were being appropriately referenced. Of the 25 screening review forms sampled, none referenced sources other than the FSAR and TS. During discussions with licensee engineers and management, the NRC inspector learned that many still relied solely upon the FSAR for 10 CFR 50.59 information. The TVA licensing organization acknowledged that additional corrective action was needed in this area. An impromptu training session was promptly held for upper site management. Changes were initiated to SDSP 27.1 to include further explicit guidance and controls. This URI has been reviewed in detail by the NRC inspectors over a several month time frame and has been evaluated as being adequately addressed by the licensee for fuel load operations but will remain open pending additional corrective action in the interim controls area. This issue along with the related issue discussed in paragraph 5a. will require followup evaluation prior to restart.

No violations or deviations were identified in the area of Licensee Actions on Previous Enforcement Matters.

11. Exit Interview (30703)

The inspection scope and findings were summarized on October 28, 1988, with those persons indicated in paragraph 1. 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.

Inspection Findings:

VIO 88-32-01: Failure to Follow Procedures For Equipment Tag-out and Independent Verification (paragraphs 2.b and 3.b)

IFI 88-32-02: Review of system 82, Diesel Generator, RTP Results (paragraph 7.b)

12. Acronyms and Initialisms

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|---|---|
| ARI | Alternate Rod Injection |
| ARP | Annunciator Response Procedure |
| BFEP | Browns Ferry Engineering Project |
| BFNP | Browns Ferry Nuclear Plant |

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| CAD | Containment Atmospheric Dilution |
| CAQR | Condition Adverse to Quality Report |
| CAR | Corrective Action Report |
| CRD | Control Rod Drive |
| CREV | Control Room Emergency Ventilation |
| CS | Core Spray |
| CSSC | Critical Structures, Systems, and Components |
| DCN | Design Change Notice |
| DCR | Design Change Request |
| DG | Diesel Generator |
| DNE | Department of Nuclear Engineering |
| DBVP | Design Baseline and Verification Program |
| EA | Engineering Assurance |
| ECN | Engineering Change Notice |
| EECW | Emergency Equipment Cooling Water |
| EGM | Electric Governor Motor |
| EQ | Environmental Qualification |
| ESF | Engineered Safety Feature |
| FCV | Flow Control Valve |
| FPC | Fuel Pool Cooling |
| FSAR | Final Safety Analysis Report |
| GE | General Electric |
| GET | General Employee Training |
| HCU | Hydraulic Control Unit |
| HPCI | High Pressure Coolant Inspection |
| HPFP | High Pressure Fire Protection |
| HVAC | Heating, Ventilation, & Air Conditioning |
| IE | Inspection and Enforcement |
| IFI | Inspector Followup Item |
| JTG | Joint Test Group |
| KW | Kilowatt |
| LER | Licensee Event Report |
| LRED | Licensee Reportable Event Determination |
| LOCA | Loss of Coolant Accident |
| MMI | Mechanical Maintenance Instruction |
| MR | Maintenance Request |
| M&TE | Measuring & Test Equipment |
| NPP | Nuclear Performance Plan |
| NQAM | Nuclear Quality Assurance Manual |
| NRC | Nuclear Regulatory Commission |
| NRR | Nuclear Reactor Regulation |
| NSRB | Nuclear Safety Review Board |
| OI | Operating Instructions |
| PMI | Plant Manager Instruction |
| PMT | Post Maintenance Test |
| PORC | Plant Operations Review Committee |
| QA | Quality Assurance |
| QC | Quality Control |
| RPIP | Regulatory Performance Improvement Program |
| RHR | Residual Heat Removal |
| RHRSW | Residual Heat Removal Service Water |
| RPS | Reactor Protection System |