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REGION II
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ATLANTA, GEORGIA 30323-0199

Report Nos.: 50-259/93-39, 50-260/93-39, and 50-296/93-39

Licensee: Tennessee Valley Authority
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Docket Nos.: 50-259, 50-260, and 50-296

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

Facility Name: Browns Ferry Units 1, 2, and 3

Inspection at Browns Ferry Site near Decatur, Alabama

Inspection Conducted: October 16 - November 19, 1993

Inspector: C. A. Patterson, Senior Resident Inspector — 12/9/93 Date Signed

Accompanied by: J. Munday, Resident Inspector
R. Musser, Resident Inspector
G. Schnebli, Resident Inspector
L. Watson, Project Engineer

Approved by: Paul J. Kellogg, Chief, Reactor Projects, Section 4A, Division of Reactor Projects — 12/9/93 Date Signed

SUMMARY

Scope: This routine resident inspection included surveillance observation, maintenance observation, operational safety verification, design changes and plant modifications, Unit 3 restart activities, reportable occurrences, action on previous inspection findings, site organization, and independent safety engineering group.

One hour of backshift coverage was routinely worked during the work week. Deep backshift inspections were conducted on October 17, 30 and November 7, 14, 18, and 19, 1993.

Results: Unit two operated continuously during this period and was on-line for 169 days at the end of the period, paragraph 4. A plant trip was possibly avoided during work on the condensate demineralizers due to a design change made under the scram frequency reduction program. The change modified the timing of the feed pump and feed booster pump suction trips. Efforts such as this have contributed to the continuous run time of the unit.

One violation with two examples for failure to control design changes was identified by an NRC inspector, paragraphs 5 and 6. The first example was for failure to adequately review a design change after control room drawings were issued to reflect identification number changes for fuses, handswitches, and other components. Plant operating instructions and labeling were not changed to agree with the drawings. Although, the problem was identified on system 31, air-conditioning system, the problem is applicable to several other systems being renumbered. The licensee's quality assessment of fuse labeling came to an erroneous conclusion that labeling was adequate. The second example involved failure to adequately control coordination of a design change that modified the unit separation boundary but the required configuration boundaries were not modified. These examples are indication of a programmatic problem with the coordination of design changes between engineering, operations, technical support, and other groups.

An unresolved item was identified by an NRC inspector for an potentially inadequate safe shutdown procedure revision, paragraph 4. The licensee on November 3, 1993, made 112 changes to the fire protection safe shutdown equipment compensatory measures. This plan was approved by the NRC April 1, 1993. These changes were made without Commission approval although there is a potential to adversely affect the ability to achieve and maintain safe shutdown in the event of a fire. Characteristic of these changes was revising a requirement to isolate the reactor water cleanup system, based on controlling the loss of water inventory, to permitting the establishment of a fire watch after seven days. The licensee made these changes after electrical cables to the reactor water cleanup system isolation valves were identified as being routed through the same fire zone.

An inspector followup item was identified by an NRC inspector concerning acceptance criteria for the analog trip units, paragraph 4. There is no limit specified for the difference between main steam line flow differential pressure readings.

An unresolved item was identified by an NRC inspector concerning the observation of an unfinished conduit modification, paragraph 5. A cable was not secured in a cable tray as required, a field change notice was not incorporated into a work plan that was closed, and conduit covers were not installed although indicated as completed in the work plan. These changes occurred about three years ago when unqualified cables were replaced, but has gone undetected until this time.

REPORT DETAILS

1. Persons Contacted

Licensee Employees:

*O. Zeringue, Vice President
*R. Machon, Plant Manager
*J. Rupert, Engineering and Modifications Manager
*T. Shriver, Licensing and Quality Assurance Manager
D. Nye, Recovery Manager
*E. Preston, Operations Manager
*J. Maddox, Engineering Manager
M. Bajestani, Technical Support Manager
A. Sorrell, Chemistry and Radiological Controls Manager
C. Crane, Maintenance Manager
*P. Salas, Licensing Manager
*R. Wells, Compliance Manager
J. Corey, Radiological Control Manager
J. Brazell, Site Security Manager

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

NRC Personnel:

P. Kellogg, Section Chief
*C. Patterson, Senior Resident Inspector
*J. Munday, Resident Inspector
*R. Musser, Resident Inspector
G. Schnebli, Resident Inspector
L. Watson, Project Engineer

*Attended exit interview

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

2. Surveillance Observation (61726)

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

a. 2-SI-4.7.A.3.b, Suppression Chamber - Reactor Building Vacuum Breaker Cycling

On October 22, 1993, the inspector observed the performance of 2-SI-4.7.A.3.b, Suppression Chamber - Reactor Building Vacuum Breaker Cycling. This SI demonstrates the operability of the two Reactor Building to Suppression Chamber Vacuum Breakers by cycling the valves (vacuum breakers) and insuring that less force than the maximum equivalent dp specified in the TS is utilized when unseating the valves. In addition, the vacuum breakers were tested for freedom of motion as well as inspecting the internals for debris and foreign material. Personnel performing the testing utilized a current revision of the SI and demonstrated the proper technique for independent verification. No deficiencies were noted by the inspector.

b. 2-SI-4.5.B.1.d(II), Quarterly RHR System Rated Flow Test Loop II

On October 29, 1993, the licensee performed 2-SI-4.5.B.1.d(II), Quarterly RHR System Rated Flow Test Loop II as a routine surveillance required by TS. The surveillance was completed satisfactorily. The inspector reviewed the completed procedure and verified the plant conditions were acceptable for performing the test, the acceptance criteria were met, the test equipment was appropriate, and the system was returned to the standby lineup. No deficiencies were noted by the inspector.

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

3. Maintenance Observation (62703)

Plant maintenance activities were observed and/or reviewed for selected safety-related systems and components to ascertain that they were conducted in accordance with requirements. The following items were considered during these reviews: LCOs maintained, use of approved procedures, functional testing and/or calibrations were performed prior to returning components or systems to service, QC records maintained, activities accomplished by qualified personnel, use of properly certified parts and materials, proper use of clearance procedures, and implementation of radiological controls as required.

Work documents were reviewed to determine the status of outstanding jobs and to assure that priority was assigned to safety-related equipment maintenance which might affect plant safety. The inspectors observed the following maintenance activity during this reporting period:

On November 9, 1993, the inspector observed maintenance activities associated with WO 93-14450-00, which was written to backfill the reference leg for the 2-LT-3-60, an A channel instrument. Operations had previously identified that this instrument was indicating approximately one inch higher than the instruments in B channel. The inspector verified the prerequisites were met and that the technicians had the appropriate approvals to begin work. Operations entered the applicable LCOs and the reference leg was backfilled without incident. Following completion of the work, the instrument indicated approximately one and one-half inches lower than the B channel. Engineering stated that this was expected due to the difference in elevations between the condensing pots. The inspector will continue to follow activities in this area.

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

4. Operational Safety Verification (71707)

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

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

a. Unit Status

Unit 2 operated continuously during this period without any significant problem. The unit was on-line for 169 days at the end of the period.

b. Unit 2/3 Separation Clearances

The Unit 3 electrical distribution system currently has operational restrictions required to allow operation of Unit 2. The restrictions are a result of unresolved engineering concerns involving Appendix R, station blackout, equipment qualification, or electrical separation/isolation capabilities. These restrictions are controlled by clearances, caution orders, and operating instructions. They, in turn, reference the electrical distribution system one-line drawings which use notes to restrict breaker closure. The inspector reviewed portions of clearances 3-91-095, 3-91-096, 0-91-353, and 0-91-362 which were issued to prevent Unit 1 and 3 components from affecting Unit 2 operation. These clearances contain components restricted by engineering holds and other components tagged for convenience. This could include entire systems that are tagged which are currently not needed, rather than spending resources to identify portions of systems that do need to be tagged. Boundary isolation points tagged by these clearances may be untagged for a period not exceeding forty-five days provided it is approved by Technical Support, Operations, and if an engineering hold exists, by Site Engineering. If the component needs to be removed from the clearance for a period exceeding forty-five days, the Unit interface drawings depicting these components is to be revised to indicate the new position. During this inspection the inspector noted changes to the clearance boundary which exceeded this time requirement. This was discussed with operations management and the discrepancy was adequately resolved.

c. RWCU Appendix R Safe Shutdown Procedure Concerns

On October 28, 1993, Engineering determined that a calculation used erroneous cable routing information to determine the impact of an Appendix R fire on the RWCU system. The calculation assumed that the RWCU system was unaffected to the extent that the system could be isolated by one of two isolation valves in the event of an Appendix R fire. However, it was not recognized that the isolation capability was not available for fires occurring in fire zone 2-4 and fire area 16. During a review of the calculation it was discovered that the cables to the isolation valves traverse these areas and therefore the system is not assured of being able to be isolated as required. As a result of this finding, Engineering initiated PER 93-0145. On October 29, 1993, roving fire watches were established in fire zone 2-4 and fire area 16, as compensatory measures due to this condition. On November 2, 1993, the inspector questioned why the licensee did not take compensatory measure A for the two valves, as described in the Appendix R SSP. Compensatory measure A required that action be taken in accordance with the TS referenced for that particular component, which in this case, was to isolate the system by shutting the two affected valves within four hours. Additionally on November 2, 1993, an Engineering walkdown verified that the cables for the two valves in fire zone 2-4 were located adjacent to each other and as such could not be relied on during a fire in this area. On November 3, 1993, the Appendix R SSP was revised to allow the option of taking compensatory measure A or B for the one-hundred-twelve items previously requiring only compensatory measure A. Compensatory measure B requires that the Appendix R function of the equipment be restored in seven days or an

equivalent shutdown capability be provided, which is typically satisfied by establishing a fire watch in the affected areas.

TS 6.8.1.1 requires that written procedures shall be established, and maintained covering the fire protection program implementation. In addition, license condition 14 of the Unit 2 operating license states that changes to the fire protection program can be made without approval of the NRC provided the ability to achieve and maintain safe shutdown is not adversely affected by a fire. The SSP revision approved on November 3, 1993, changed the fire protection program in such a way as to potentially adversely affect the ability of the plant to achieve and maintain safe shutdown, without first gaining prior approval from the NRC. The licensee indicated that additional information was being identified on this subject and requested a meeting to discuss this information. This item is unresolved pending the outcome of a meeting on the subject and is identified as URI 260/93-39-01. In addition, on November 4, 1993, the inspector questioned why the licensee did not establish a continuous fire watch in the cable spreading room of fire area 16 as required by the SSP. The licensee stated that a continuous fire watch should be established and corrected the problem.

d. Drywell Control Air Dewpoint

Moisture sensors which provide a control room alarm upon sensing high moisture content in the drywell control air system have been proven to be unreliable. As a compensatory measure, TACF 2-93-01-32 was written to facilitate the installation of a portable dewpoint hygrometer on the 2B control air receiver tank to allow periodic monitoring of the dewpoint temperature. The safety assessment for all TACF states that upon receipt of the high moisture alarm, Operations personnel will valve in the dewpoint hygrometer and verify that an acceptable dewpoint is obtained. The inspector questioned two UOs and two ASOSs about the actions to be taken upon receipt of this alarm. None of the operators were aware that specific action needed to be taken in accordance with the TACF. This weakness was discussed with operations management who felt it prudent to revise the ARP to include the appropriate actions.

e. Unit Operator Instrument Checks and Observations

While reviewing the Unit 2 unit operator instrument checklist, the inspector noted that the main steam line flow, indicated as a dp on ATUs, differed by as much as nine psid between steam lines. Each steam line has one flow element which feeds four transmitters. The operator records the dp from each ATU and compares the values with the others. When questioned as to what would be considered an unacceptable comparison, the unit operator did not know. He stated that there was no acceptance criteria associated with the comparison. This instrument check, performed

once per day to satisfy the requirements of TS 4.2.A, is defined by TS as a qualitative determination of acceptable operability by observation of instrument behavior and shall include a comparison with other instruments measuring the same variable. Operations management concluded that there was no requirement to define the acceptance criteria of an instrument check in terms of allowable differences between instruments measuring the same parameter. However, the licensee plans on collecting information from other utilities concerning this and possibly revising their procedures to better define acceptable comparisons. Pending resolution, this item will be tracked as IFI 260/93-39-02, Acceptance Criteria For Instrument Comparison Surveillances.

f. REX System Failure

On November 2, 1993, the inspector experienced problems with the licensee's REX system. This system is a computerized exposure tracking system for personnel working under a radiological work permit. The licensee requires that personnel have a briefing on the system use prior to entry on an RWP. In the morning, the inspector entered the RCA on the general access RWP for a tour of the reactor building. After one hour, the inspector exited, signing off the RWP using the computer. The inspector noted that the dose received was zero although one or more MR was expected after being in the reactor building for over an hour.

Later, in the afternoon the inspector attempted to use the REX system again for entry into the reactor building. The computer initially stated that entry was denied and a briefing was required for entry. A health physics technician assisted the inspector and the computer indicated that an exit entry had not been received. The inspector questioned this and noted several rows of other people were in line waiting to enter the RCA because of similar problems. The inspector discussed the systems's apparent problems with a radiological controls supervisor. Reliability of the REX system will be monitored by the inspector.

g. Housekeeping

On November 2, 1993, during a routine tour of the Unit 2 reactor building the inspector observed several large loose pieces of insulation on top of some HVAC ducting. The ducting was near the overhead of this floor elevation but appeared to be at location for some time due to the collection of dust and dirt around the insulation.

On November 4, 1993, during a routine tour of the Unit 1 and 2 diesel generator building the inspector noticed that the general cleanliness of the rooms had deteriorated somewhat. The rooms are normally clean. The rooms were dusty and a large number of spider webs had accumulated in the area.

These issues were discussed with plant management on November 5, 1993.

h. Scram Reduction Efforts

On October 13, 1993, at 6:10 a.m., the Unit 2 Condensate Demineralizer System bypassed when returning demineralizer H to service. This was apparently caused by the "E" valve on the "B" demineralizer sticking closed and then fully opening quickly at the same time the "H" demineralizer was being placed in service. The demineralizers were restored to normal and no plant transients were observed. Work request C232341 had been previously submitted on the "B" demineralizer for the failure of the "E" valve to respond properly to its air signal. The valve was subsequently adjusted and the work order closed out later that same day. Discussions with operations personnel indicated that had this transient occurred in the past it might have resulted in a plant trip. However, due to the efforts of the Scram Frequency Reduction Team, modifications had been accomplished to this system that reduced the possibility of a trip for this event. The recommendation made by the SFRT was to stagger the RFPs and the CBPs low suction pressure trips such that all the RFP's did not trip at the same point and all the CBP's did not trip as the same point. This was to be accomplished by varying the setpoints or varying the time delays of the pump trip circuitry. By varying one of these parameters a low pressure pulse would not trip either all the RFP's or all the CBP's. Therefore, loss of a single train might allow pressure to build back up to prevent tripping the other trains, thus preventing a loss of all feedwater. This modification was implemented by DCN W 16281A which was completed in May 1991. The inspectors considered the licensee's efforts in this area commendable.

i. Cold Weather Preparation

The inspector reviewed the licensee's program for cold weather protection of equipment. The licensee has an extensive program for identifying, establishing, and repairing freeze protection equipment such as heat tracing, room heaters, dampers, and space heaters. Operations completed various valve, damper, and door lineups for prevention of cold weather damage. Maintenance has the responsibility of testing the equipment and if needed performing repairs, and were approximately 90 percent complete. The freeze protection for the fire protection equipment was complete. The inspector walked down various outside areas subject to cold weather damage and verified heat tracing was established, heaters were operable, and temporary protection devices such as canvas or wood shelters, were in place. The inspector will continue to monitor activities in this area as repairs on damaged equipment is completed.

One unresolved item and one inspector followup item was identified in the Operational Safety Verification area.

5. Design Changes and Plant Modifications (37700)

a. Program Review

The inspector reviewed changes to the plant design change and modification process. The licensee has replaced BP 205, Issue Management, which described the design issue origination process, with BP 312, BFN Scope Control Process. BP 312 establishes the PID process.

Any nuclear power individual can initiate a PID. Primary responsibilities for initial review and tracking have been divided between the SMM, the ICPM, and the Site Controller. The SMM is responsible for tracking planning items, developing a BFN Long Range Plan and preparing a Fiscal Year Project List. The ICPM develops scope of work and cost benefit information. The Site Controller performs resource and funding estimates.

BP 312 also establishes two review groups, the SMART and the WCT. The SMART performs the review for all PIDs associated with scope changes and emergent work. The WCT performs the review for new work.

The PID is reviewed by the principal manager and system engineer then goes to the SMART or WCT for disposition. BP 312 provides guidance on categorizing and ranking the items and defines the managers on the SMART and WCT.

The inspector reviewed the MIL of open and closed items and the inactive MIL. Packages of selected MIL items which were open or had been placed on the inactive list were reviewed to determine if the licensee had adequate justification for the disposition of the item, i.e., delaying implementation or canceling the project. The inspector had no questions on disposition of the packages.

The inspector also reviewed portions of the following procedures which define the design review process:

SSP 9.3, Plant Modifications and Design Change Control, Rev. 10

SSP 9.4, Configuration Management/Control, Rev. 1

SSP 9.5, Design Engineering, Rev. 0

BFEP PI 89-06, Design Change Control, Rev. 9

b. QDCN and SCDN Processes

During October 1993 the licensee identified that a containment isolation valve had been replaced with a valve of a different design and the Appendix J test method had not been revised to test for stem leakage. This issue is discussed in paragraph 5.d. The valve orientation had been questioned via QDCN 23603A and the Appendix J testing error was not identified in the QDCN review.

The inspector reviewed the QDCN process described in procedures SSP 9.3 and BFEP PI 89-06. The QDCN is used to disposition questions and provide clarification of existing design output documents including DCNs. The inspector selected 20 of the 75 QDCNs closed in April and May of 1993 (end of Unit 2 Cycle 6 refueling outage) for review. The inspector noted that in five of the QDCNs generated by organizations other than Site Engineering, including QDCN 23603A, the review by the System Engineer and the Technical Support Superintendent had been marked as not applicable. Site Standard Practice, Plant Modifications and Design Change Control, Rev. 10, Section 3.2.2, requires that QDCNs that are not originated by Site Engineering be reviewed by the Technical Support - System Engineer and the Technical Support - System Engineering Supervisor. The failure to route QDCN 23603A to Technical Support for review contributed to the failure to identify the inadequate testing configuration for leak rate testing of containment isolation valve 2-FCV-064-20. Technical Support provides Appendix J reviews. In two of the cases reviewed, an FDCN had been changed to a QDCN. In these cases, there does not appear to be a review of the need to send the QDCNs to Technical Support for review.

The inspector reviewed a sample of 20 of the 63 SCDNs generated since October 1993. SCDNs are used to support documentation changes only. A case where an SDCN has been used to make changes in plant configuration as discussed in paragraph six. The inspector reviewed the use of the selected SDCN to confirm that they were used for documentation changes only. The inspector also sampled the documents that were changed to confirm that procedures and drawings had been updated. No problems were identified.

c. Fuses

On November 5, 1993, the inspector compared electrical fuse labeling in the plant against electrical equipment fuse tabulation drawings. These are primary drawings maintained in the control room. Because of previous concerns and problems with fuse labeling the licensee's quality assurance organization conducted an assessment of fuse labeling Assessment Report NQA-BF-93-154, dated November 2, 1993. The assessment team concluded that the fuse labeling program was adequate and effectively implemented. The problem associated with mislabeling of SBGT fuses discussed in the licensee's incident investigation, II-B-93-034 was considered isolated.

The inspector identified several fuse numbers that had been changed on drawing 2-45B721-50-10 under revision 4 dated April 3, 1993, for DCN S19592, but the labeling had not been changed in the electrical panel next to the fuses. This drawing was a primary drawing and a CCD. The following is an example of the labeling problems:

Fuse UNID (Per Drawing)	Fuse (As Labeled)
2-FU3-031-7206A	2-FU3-031-4099B1
2-FU3-031-7206B	2-FU3-031-4099B2
2-FU3-031-7206C	2-FU3-031-4099B3
2-FU3-031-7207D	2-FU3-031-4099B4
2-FU3-031-7208E	2-FU3-031-4099B5
2-FU3-031-7209F	2-FU3-031-4099B6

The licensee initiated a PER-930150 that identified an additional six fuses with labeling different than the drawing.

The inspector conducted an independent review of the DCN S19592 and concluded that numerous other numbers were changed in addition to fuse numbers. The change was to complete the MEL safety system project for system 31. This consisted of EMS loading of the Q-list, EQ-list, I-Tabs, Valve Tabs, and Fuse Tabs. Ten pages of fuses, handswitches, etc., were changed by this S-DCN.

The inspector checked two switches listed as being changed. 2-XSW-31-4099A1 Appendix R SDBR ACU NORM/EMER SW was changed to 2-XSW-031-7205.

The inspector obtained a copy of the system panel lineup checklist and found the checklist had not been changed. In addition, the switches located in electrical board room 2B were checked and they were still labeled as the checklist. This was compared to drawings in the control room 2-45E2749-18 and 2-45E769-18 that had the new numbers.

The inspector did not sample further but concluded that the problem was not isolated to fuses. The problem was with anything changed by the S-DCN. The S-DCN that required changing of numbering of components on plant drawings had not been coordinated to have operations or others to change plant labeling and procedures.

The inspector concluded this is a programmatic problem with engineering changing drawings without adequate coordination with other plant organizations to insure drawings accurately reflect the plant configuration. The licensee has done an excellent job in the past of issuing updated drawings to reflect plant modifications. The drawings are issued prior to the system being returned to service. However, the reverse has not been true. Control room drawings have been changed to change fuse number, handswitch number, etc., without the corresponding changes being made in the plant or procedures.

According, this is the first example of the violation concerning design control. Design control requirements are established by 10 CFR 50 Appendix B Criterion III, Design Control. This violation will be identified as 259, 260, 296/93-39-03, Failure to Control Design Changes.

On November 16, 1993, the inspector discussed these issues with the Engineering and Modifications manager and Engineering manager. Four systems have been completed for the MEL safety system project. These systems were 18, 31, 63, and 75. The inspector stressed the importance of maintaining accurate drawings to reflect the plant configuration.

d. Containment Isolation Valve Improperly Installed

As discussed in IR 259, 260, 296/93-36, paragraph 4c, the licensee discovered that CIV 2-FCV-64-20 was installed such that its shaft seals were outside of the test boundary of its periodic type C LLRT. Another issue related to this matter was noted by the inspector during the review section 5.2.3.6 (Primary Containment Venting and Vacuum Relief) of the FSAR. This section contains a statement that valves 2-FCV-64-20 and 21 (Inboard Primary Containment Isolation Valves for the Reactor Building to Torus Vacuum Breaker Lines) are air-operated and actuated by a differential pressure signal that is independent of electrical power. However, the differential pressure transmitters and solenoid valves associated with valves 2-FCV-64-20 and 21 are normally energized with electrical power. These matters were still under review at the conclusion of the last inspection period and were tracked as Unresolved Item 260/93-36-01, Containment Isolation Valve Improperly Installed.

The licensee concluded their review of this matter by completing Incident Investigation II-B-93-041. The inspectors also completed an independent review of the issue. The inspectors have concluded that the licensee failed to include a 10 CFR 50 Appendix J review in the design process which involved the installation of containment isolation valves. In not performing this review, the licensee failed to recognize the importance of valve orientation with regards to the LLRT required by TS 4.7.A.2.g. Therefore, the valve (2-FCV-64-20) was installed with its shaft seals outside of the test boundary of it type C LLRT. One of the contributing factors in this matter was the fact that DCN Q23608A was issued without first being reviewed by plant technical support. This DCN involved a question from the licensee's maintenance organization regarding the importance of flow direction when installing CIVs 2-

FSV-64-17, 18, 19, 20, and 21. The DCN Q23608A was issued by plant engineering without considering Appendix J testing of the valve. The issue of Technical Support not reviewing Q-DCNs as required by SSP-9.3 is further discussed in paragraph 5b of this report.

During the inspectors' review of this issue, the inspectors determined that a weakness existed in the installation instructions provided to the craft for the valves specified in DCN W16880A (the original DCN written to replace 2-FCV-64-17, 18, 19, 20, and 21). Simple sketches on installation instructions were not included in the DCN or WP. It appears that a great deal of engineering assistance was required during the valves installation. In regards to the inspectors concerns that FSAR Section 5.2.3.6 states that valves 2-FCV-64-20 and 21 operate independent of electrical power when in fact power is required to normally operate the valves, the licensee is in the process of changing the applicable portion of FSAR Section 5.2.3.6. The proposed change is as follows; one valve is air operated and is actuated by a differential pressure signal; upon loss of electrical power or air, the valve will fail in the open position. This change satisfied the inspector's concern in this area as the FSAR will more closely represent actual plant conditions.

The licensee has determined that valve 2-FCV-64-20 will be reoriented and tested during the next period of cold shutdown providing the applicable design documentation is completed and materials are available. These issues will continue to be tracked by URI 93-36-01, pending resolution of the valve's orientation and testing.

e. Unfinished Modification

On November 2, 1993, during a routine tour of the Unit 2 reactor building the inspector identified some conduit that appeared to be an unfinished modification. A modifications supervisor was contacted and toured the area with the inspector. On the mezzanine above the clean room at the drywell entrance there were two conduits with missing inspection covers. A single cable was in one of the conduits but the other conduit was empty. Also, the cable exited the conduit into a cable tray but the cable was dropped down out of the cable tray.

The licensee researched the problem using the conduit numbers and found that an FDCN had not been incorporated into a WP about three years ago. The licensee generated PER BF 930149 to resolve this issue. This will be tracked as URI 260/93-39-04, Unfinished Conduit Modification, pending resolution of the issue.

f. Reactor Water Level Modification

On May 28, 1993, the NRC issued Bulletin 93-03, (Resolution of Issues Related to Reactor Vessel Water Level Instrumentation in BWRs), to all licensees operating BWRs. Specifically, the bulletin deals with a generic concern in which noncondensible gases could become dissolved in the reference leg of BWR water level instrumentation and lead to a false high level indication after a rapid depressurization event. In addition to several short term compensatory actions, the bulletin requested that each licensee implement hardware modifications necessary to ensure the level instrumentation reliability for long-term operation. This hardware modification was to be implemented by all licensees in the next cold shutdown period after July 30, 1993.

TVA responded to the Bulletin with a letter dated July 30, 1993, which stated that TVA would install a continuous fill system which injects CRD water to the water level instrumentation condensing chambers through reference leg piping. Because Browns Ferry Unit 2 has not been in a cold shutdown condition since starting up from the refueling outage in June 1993 and the fact that being in this condition is a requisite for installation of the continuous fill system, the modification is not yet implemented. In preparation of their first period of cold shutdown during the operating cycle, the licensee has prepared and issued DCN W16435C to document this modification. Currently, the licensee is installing piping, piping supports, cabling and conduit to the maximum extent possible with the unit at power in order to minimize the effort once the unit has reached cold shutdown. The licensee anticipates completing this work on November 23, 1993. Once the unit reaches cold shutdown, current estimates are that modification will require 7½ days for implementation. The inspectors are currently monitoring the "pre-outage" work and anticipate monitoring the final implementation of the modification.

One violation was identified in the design change area.

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

The inspector reviewed and observed the licensee's activities involved with the Unit 3 restart. This included reviews of procedures, post-job activities, and completed field work; observation of pre-job field work, in-progress field work, and QA/QC activities; attendance at restart craft level, progress meetings, restart program meetings, and management meetings; and periodic discussions with both TVA and contractor personnel, skilled craftsmen, supervisors, managers and executives.

a. Modification of Unit Separation Clearance to Support Unit 3 Condenser Hydrostatic Test

The return to service of systems in support of Unit 3 condenser hydrostatic test and other activities required that some breakers

be released from an engineering hold. DCN S25756A revised one-line drawings based on calculations which altered various operational restrictions. These restrictions include such things as Appendix R, station blackout, equipment qualification, and electrical cable separation/isolation capabilities. For example, 3B CRD pump usage is restricted due to a cable separation violation and the HPCI pump discharge valve, 3-FCV-74-73, is restricted due to battery loading concerns. The safety assessment for the DCN states that the evaluation concluded that certain circuit breaker operating restrictions must be continued and/or revised to support continued Unit 2 operation and Unit 3 restart. The DCN specifically listed those components whose use was still restricted. On October 15, 1993, while verifying the proper positioning of the components on the list, the inspector identified the following discrepancies:

1) Breaker 602 on Battery Board 2 was being controlled by a clearance instead of a caution order.

2) Breaker 601 on Battery Board 3 was found closed, however it was required to be open.

3) Breaker 612 on Battery Board 3 was found closed and being controlled by a caution order, however it was required to be open.

After operations management was informed of these discrepancies by the inspector, operations personnel walked down all of the equipment affected by an engineering hold and found approximately two pages of needed changes to the clearances, caution orders, and Operating Instructions controlling these components. Discussion with Operations and Site Engineering management indicated that these changes were inappropriately made without the review of the operations staff. Criterion III of 10 CFR 50 Appendix B, requires that measures shall be established for the identification and control of design interfaces and for coordination among participating design organizations. These measures shall include the establishment of procedures among participating design organizations for the review, approval, release, distribution, and revision of documents involving design interfaces. Failure to coordinate the issuance of design documents with the resulting changes in plant configuration is a violation of 10 CFR 50 Appendix B, Criterion III, Design Control, and resulted in misconfiguration of the plant. This will be identified as the second example of VIO 259, 260, 296/93-39-03, Failure to Control Design Changes.

The inspector discussed with site management that close monitoring of the unit separation boundaries was essential as more Unit 3 systems are tested and returned to service.

b. Design Changes and Plant Modifications

The inspectors review selected Design Change Notice packages associated with plant modifications to support the Unit 3 recovery effort. The DCN work packages were reviewed and work in progress was observed to: ensure that the DCN packages were properly reviewed and approved by the appropriate organizations in accordance with the licensees administrative controls; verify the adequacy of the 10 CFR 50.59 evaluations performed and that the appropriate FSAR revisions were planned or completed, if applicable; ensure that the applicable plant operating procedures and design documents were identified and revised to reflect the modification; verify that the modifications were reviewed and incorporated into the operations training program, as applicable; verity that the modifications were installed in accordance with the work package (for those that could be physically inspected); ensure that the modification was consistent with applicable codes and standards, regulatory requirements, and licensee commitments; and ensure that post modification testing requirements were specified and that adequate testing was accomplished. The following DCNs were reviewed:

1) DCN W17631, Base Plate Installation

On October 21, 1993, during a routine tour of the Unit 3 reactor building the inspector observed base plate installation of supports. This was being performed under DCN W17631. The inspector noted that four anchor bolts were in place but the bolt holes were next to (about 1/4 inch) away from an existing hole. The vacant holes were grouted over but the sleeve was still in place. This in effect appeared to negate the grouting since the spacing between the sleeves was still below minimum spacing. The inspector contacted a QA supervisor to review the installation. In addition the inspector reviewed MAI 5.1 that specified the spacing between anchor bolts. Most distances required several inches between spacing. Initial review of this issue determined that F-DCN 26899 was issued to revise anchor bolt spacing dimensions because of spacing violations. A different type anchor bolt penetrating several inches into the floor was used. This design strengthened the area to be pulled out. Additionally, calculation CD-Q3064-922915 was revised to incorporate the F-DCN 26899. The inspector reviewed the calculation and concluded the issue was resolved.

2) DCN W7731A, Electrical Separation

The inspector reviewed the work associated with scaffolding in the Unit 2 reactor building spaces. The work was being performed under DCN W7731A to reroute the normal power supply cable to 250 volt reactor MOV Board 3C to correct

electrical separation concerns. The cable was a safety system cable but was routed in non-safety raceways. This was a Unit 3 DCN but involved work in Unit 2 operating spaces. The inspector reviewed the safety assessment for the DCN and unit separation issues were addressed. TS requirements, secondary containment, and system operability were each discussed. Additionally, inspection of the scaffolding in Unit 2 reactor building identified no deficiencies. The inspector concluded that the licensee had performed a thorough assessment of this modification's impact on the operating unit.

c. System SPOC's

The purpose of SPOC process is to provide a systematic method for evaluating items and issues which potentially affect the ability of Unit 3 systems and the Unit 3 portion of common systems to perform as designed. This process determines the status of each item/issue and assures completion of those which affect system return to operation for Unit 3 restart. For each system evaluated, the SPOC process may be accomplished in two phases. Phase I SPOC addresses the Restart Test Program testing milestone if that milestone exists for the system, and establishes system status control by the Operations department. Phase II SPOC addresses System Return to Operation in preparation for the declaration of system operability. Each phase ensures that open items/issues which potentially affect the phase are either completed, or reviewed and satisfactorily dispositioned. The SPOC process does not declare system operability. Rather, it is used to support a declaration of system operability which is made after other requirements for operability are satisfied (e.g., support systems available, performance of Surveillance Instructions, etc.).

The following system SPOC packages were reviewed to ensure they complied with SSP 12.55, Unit 3 System Pre-Operability Checklist, Revision 5. Minor deficiencies were resolved with the system engineer.

System 27, Condenser Circulating Water System, Phase I.

d. System Testing

On November 8, 1993, the licensee performed a static hydrostatic test on the condensate side of the main condensers. The test was conducted in accordance with SOI-19, Flooding Hotwell for Condenser Tube Leak Check. There were no major problems encountered; however, the test did identify that two condenser tubes required plugging, seventy tubes required rerolling, and four tube sheet holes required plugging because the internal baffle plates didn't allow for tube installation. This was the

first milestone in the licensee's efforts to return Unit 3 to service and it was accomplished seven days ahead of schedule.

7. Reportable Occurrences (92700)

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

(CLOSED) LER 50-296/92-006, Inadvertent Emergency Diesel Generator Start During Testing Due To A Short Circuit.

On December 3, 1992, during the performance of the 3D DG redundant start test, the 3D DG inadvertently started when test leads connected across the autostart relay shorted. The short occurred when the test lead was pinched under a movable handle on the test equipment cart and damaged. As corrective action the licensee replaced the damaged test leads and secured them in such a fashion to reduce the potential for damage. The moveable handles on the carts were also removed. Additionally, training was provided to maintenance personnel on the impact of potential degradation of test leads. The inspector reviewed the training records of the maintenance personnel and verified the test carts were modified to reduce the potential for damage to the test leads. The inspector considers this item closed.

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

a. (CLOSED) Fire Protection Weakness Identified in IR 93-19

A weakness was previously identified in IR 93-19 involving the control and timely update requirements of the licensee's FPR and the LCO determinations for fire protection features impacted by recent design changes. The licensee issued Revision 6 on September 30, 1993, to SSP 12.15, Fire Protection, addressing this weakness. The revision added Appendix A, Management of the Fire Protection Report Volume 1, providing administrative controls for the purpose of maintaining and controlling the FPR including revising and updating the document. Step 5.0 of this appendix assigns responsibility to site engineering requiring that the FPR be updated within 30 days after completion of a refueling outage or earlier if deemed necessary. The residents reviewed the licensee's corrective actions for this concern and discussed them with the regional based inspector and determined this issue closed.

b. (CLOSED) URI 259, 260, 296/93-02-02, Mislabeling of Fuses

(CLOSED) URI 259, 260, 296/93-08-01, Design Control Coordination Discrepancies

(CLOSED) URI 259, 260, 296/93-25-03, Inoperable SBGT Due to Fuse Mislabeling

These issues are considered closed with the issuance of violation 259, 260, 296/93-39-03, Failure to Control Design Changes. This violation encompasses fuse labeling issues.

9. Site Organization

On November 10, 1993, Eugene Preston reported on site as the Operations Manager to fill the vacancy left by the resignation of Max Herrell.

On November 15, 1993, Richard Machon reported on site as the Plant Manager replacing John Scalice who transferred to Watts Bar as the Operations Vice President.

10. Independent Safety Engineering Group (40500)

The inspector reviewed the status of the site's ISEG. The requirement for an ISEG was a commitment under the Nuclear Performance Plan (Volume 3) for the restart of Browns Ferry. The group was established by NUREG 0737 guidelines for multi-site utilities. The group is patterned after the Sequoyah plant that has the ISEG requirement in TS. It is not a TS requirement at BFNP.

On April 8, 1993, the licensee submitted revision three of their NQA plan that combined the site licensing and quality manager into one position. This also placed the ISEG under the licensing and quality manager on-site instead of an off-site corporate manager. Region II accepted the changes on June 9, 1993. The licensee contends that the site licensing and quality manager does not report to any site manager and the independence is maintained.

Further changes are occurring with ISEG. The group functions are being incorporated into a new group called Reactor Safety Engineering and Review commonly called RSER. A procedure for this group is being developed.

11. Exit Interview (30703)

The inspection scope and findings were summarized on November 19, 1993, with those persons indicated in paragraph one above. The inspectors described the areas inspected and discussed in detail the inspection findings listed below. The licensee did not identify as proprietary any of the material provided to or reviewed by the inspectors during this inspection.

The Site V.P. stated that a fire watch provided immediate detection of a fire and the changes to the plan, as outlined in paragraph 4 above, were justified.

Item Number	Description and Reference
260/93-39-01	URI, Inadequate Safe Shutdown Procedure Revision, paragraph 4.c.
260/93-39-02	IFI, Acceptance Criteria For Instrument Comparison Surveillances, paragraph 4.e.
259, 260, 296/93-39-03	VIO, Inadequate Design Control, paragraph 5.c and 6.a.
260/93-39-04	URI, Unfinished Conduit Modification, paragraph 5.e.

Licensee management was informed that 1 LER and 3 URIs were closed.

12. Acronyms and Initialisms

ARP	Annunciator Response Procedure
ASOS	Assistant Shift Operations Supervisor
ATU	Analog Trip Units
BFEP	Browns Ferry Engineering Project
BFNP	Browns Ferry Nuclear Power Plant
BP	Business Practice
CBP	Condensate Booster Pump
CCD	Configuration Control Drawing
CFR	Code of Federal Regulations
CIV	Containment Isolation Valve
DCN	Design Change Notice
DG	Diesel Generator
DP	Differential Pressure
EDG	Emergency Diesel Generator
EMS	Equipment Management System
FCV	Flow Control Valve
FDCN	Field Design Change Notice
FPR	Fire Protection Report
FSAR	Final Safety Analysis Report
HPCI	High Pressure Coolant Injection
HVAC	Heating, Ventilation, & Air Conditioning
ICPM	Item Coordinator Project Manager
IR	Inspection Report
ISEG	Independent Safety Engineering Group
LCO	Limiting Condition for Operation
LER	Licensee Event Report
LLRT	Local Leak Rate Testing
MAI	Modification Alteration Instruction
MEL	Master Equipment List

MIL	Master Issues List
MOV	Motor Operated Valve
NRC	Nuclear Regulatory Commission
PER	Problem Evaluation Report
PI	Project Instruction
PID	Planning Item Description
PSID	Pounds Per Square Inch Differential
QA	Quality Assurance
QC	Quality Control
RFP	Reactor Feedwater Pump
RHR	Residual Heat Removal
RSER	Reactor-Safety Engineering and Review
RWCU	Reactor Water Cleanup
RWP	Radiological Work Permit
SBGT	Standby Gas Treatment System
SFRT	Scram Frequency Reduction Team
SI	Surveillance Instruction
SMM	Scheduling Methods Manager
SOI	Special Operating Instruction
SPAE	System Plant Acceptance Evaluation
SPOC	System Pre-Operability Checklist
SSP	Site Standard Practice
SSP	Safe Shutdown Procedure
TACF	Temporary Alteration Control Form
TS	Technical Specification
UO	Unit Operator
URI	Unresolved Item
VIO	Violation
WCT	Work Control Team
WO	Work Order
WP	Work Permit