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

Report Nos. 50-259/87-42, 50-260/87-42, and 50-296/87-42

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

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

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

Facility Name: Browns Ferry Nuclear Plant

Inspection at Browns Ferry Site near Athens, Alabama

Inspection Conducted: November 1-30, 1987

Inspectors: G. L. Paulk — 12/9/87
G. L. Paulk, Senior Resident Inspector — Date Signed

C. A. Patterson — 12/9/87
C. A. Patterson, Resident Inspector — Date Signed

Charles R. Brooks — 12/9/87
C. R. Brooks, Resident Inspector — Date Signed

E. F. Christnot — 12/9/87
E. F. Christnot, Resident Inspector — Date Signed

A. H. Johnson — 12/9/87
A. H. Johnson, Project Engineer — Date Signed

NRC Contractor Assistance:

David B. Waters, Previous Enforcement Matters

Gary W. Bethke, Previous Enforcement Matters

Donald A. Beckman, Engineering Change Notices/Modifications

David H. Schultz, Engineering Change Notices/Modifications

Approved by: A. J. Ignatonis — 12/11/87
A. J. Ignatonis, Section Chief, — Date Signed
Inspection Programs
TVA Projects Division

SUMMARY

Scope: This routine inspection was in the areas of operational safety, maintenance observation, reportable occurrences, and previous enforcement matters, restart test program, and engineering change notices/modifications.

Results: No violations or deviations were identified in this report.

REPORT DETAILS

1. Licensee Employees Contacted:

H. G. Pomrehn, Site Director
*J. G. Walker, Plant Manager
*P. J. Speidel, Project Engineer
*J. D. Martin, Assistant to the Plant Manager
*R. M. McKeon, Superintendent - Unit 2
*J. S. Olsen, Superintendent - Units 1 and 3
T. F. Ziegler, Superintendent - Maintenance
D. C. Mims, Technical Services Supervisor
J. G. Turner, Manager - Site Quality Assurance
M. J. May, Manager - Site Licensing
*J. A. Savage, Compliance Supervisor
A. W. Sorrell, Health Physics Supervisor
R. M. Tuttle, Site Security Manager
J. R. Kern, Fire Protection Supervisor
D. A. Pullen, Office of Nuclear Power, Site Representative
A. Ballard, Acting Principal Engineer, DNE
R. Burt, Mechanical Modification Section Supervisor
A. Chapman, Asst. Modification Manager
L. Clardy, QA Surveillance Supervisor
P. Crabb, Work Plan Coordinator
E. Long, QC Supervisor, Mechanical
R. Martin, Asst. Modification Manager
S. McRight, PMT Supervisor
R. Young, Modification Manager

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

2. Exit Interview (30703)

The inspection scope and findings were summarized on November 6, 20, and 30, 1987 with the Plant Manager and/or Superintendents and other members of his staff.

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

*Attended exit interview

3. Licensee Action on Previous Enforcement Matters (92702)

(OPEN) Inspector Follow-up Item (259,260,296/87-02-06). This item resulted from a walkdown of the Diesel Starting Air System (System 86) in association with review of the Browns Ferry configuration management system. Concerns were noted with: an inconsistent definition of the System 86 boundary with the FSAR, Figure 8.5-2; incorrect FSAR

Figure 8.5-2; local labeling of air compressor discharge check valves 86-521B and 86-501B; actual component labels installed on the two automatic pressure operated control switches not agreeing with the designation shown on the Configuration Control Drawing (CCD) for System 86.

The licensee presented information in Commitment Closure Summary SLT 87-C049-009 consisting of completed Maintenance Requests and revised system 86 CCDs for Units 1 and 2 Diesel Generators A-D for closure of this item. MR# A-718105 was completed 3/25/87 and provided tags for all identified valves, pressure switches, hand switches and alarm indicating lights. MR# A-814518 was completed 6/28/87 and retagged all of the items tagged in MR# A-718105 due to a legend change. CCD 4-47E861-1 Rev. 3, 4-47E861-2 Rev. 1, 4-47E861-3 Rev. 1 and 4-47E861-4 Rev. 1 define the diesel generators and air starting components downstream of the "Y" strainers as out of walkdown scope. A copy of FSAR Figure 8.5-2 is included in the licensee's information package but has not been revised nor corrected.

A walkdown of Diesel Generator D Starting Air System was conducted on 11/4/87 and confirmed that tags listed in MR# A-814518 were in place and that the tag legends matched those on the plant valve and component tag request forms (BF-193) and the Nameplate Request Forms (BF-200). It was noted that all previous tags were removed. However, a check of HS-86-505 for Diesel Generators A, B, and C revealed that an old tag with a previous legend was still in place.

The licensee investigated the handswitches and agreed that the labeling should be consolidated and redundant labels eliminated. MR# A-814614 has been initiated (and completed) to correct these items. The tags for air compressor discharge check valves 86-521A-D and 86-501A-D were in place although the revised CCDs still refer to them as "not verified". This item is closed except for revision FSAR Figure 8.5-2, which remains a concern. This item remains open and will be reviewed during a subsequent inspection.

The inspector noted during the walkdown that none of the major components such as the air compressors, motors and individual starting air tanks were labeled, although the associated valves and switches have unique identification. Discussions were held with licensee personnel involved in the plant tagging program who indicated that component identification for pumps, motors and similar components was desirable but not on the same priority basis as the tagging of valves, switches and panel indications. Lack of consistent and explicit component identification prior to unit startup could potentially result in wrong train/wrong component events as described in IN 87-25. Subsequent discussions with the licensee indicated that component labels will be provided for Unit 2 (and common) components prior to Unit 2 startup, along with piping color coding and directional arrows in accordance with procedure BF-8.11.

(CLOSED) Inspector Followup Item (259,260,296/84-32-04). This item resulted from a question on the method for compensating the drywell pressure indications for atmospheric changes in pressure. Other licensees

had identified a potential problem with the drywell pressure scram sensing device not being compensated for with regard to atmospheric changes that occur. Misleading information could be provided should the drywell pressure indication system not indicate actual inplant conditions.

The licensee presented information in Commitment Closure Summary SLT 850913004 consisting of several internal memoranda and attachments for closure of this item. This information concluded that the drywell pressure sensing instruments are differential pressure devices, with setpoints in terms of psig and with the instrument continually vented to the reactor building. Thus, the parameter of differential pressure between the drywell and the reactor building is automatically compensated for changes in barometric pressure. The inspector discussed this information with licensee personnel who informed the inspector that the Static "0" Ring pressure switches formerly used at Browns Ferry Units 1, 2 and 3 would be replaced with Rosemont pressure transmitters prior to restart as part of the analog system improvement. However, the new devices would still measure differential pressure for the drywell pressure application and the conclusions in the utility package would still be valid. Discussions were also held with Region II inspectors, who concurred that the utility conclusions regarding the self-compensation of differential pressure devices was valid. This item is closed.

(OPEN) TI 2515/87 Instrumentation for Nuclear Power Plants to Assess Plant Environs Condition During and Following an Accident (Regulatory Guide 1.97). This TI requires verification that an instrumentation system exists for assessing plant conditions during and following the course of an accident that meets the criteria specified in the applicable revision to Regulatory Guide 1.97 and that the system is installed in accordance with licensee commitments and as described in the safety evaluation report (SER). Submittals were made on April 30, 1984 and May 5, 1985 by TVA regarding their conformance to Regulatory Guide 1.97. The licensee indicated that the SER has not yet been issued by NRC, and that additional information will be requested of TVA by December 1, 1987. This item remains open.

(OPEN) I.E. Bulletin 80-06, Engineered Safety Feature (ESF) Reset Controls. The subject bulletin was issued March 13, 1980 and requested action by licensees to review ESF systems and determine if resetting of Engineered Safety Features Actuation Signals (ESFAS) allowed the changing of components from their safety or emergency mode to their normal mode. The only modifications required as a result of the review was to prevent energizing the TIP withdrawal enable circuit upon reset of containment isolation. Installation of this modification was committed to be completed in December 1984 in the licensee's letter of December 4, 1981.

Several items of correspondence from NRR to TVA indicated that a safety evaluation was being prepared based on the licensee's submittals. Since a copy of the SER could not be found in the resident file or the BFNP file on IEB 80-06, a copy was obtained from the NRC TVA project manager. The conclusion of the SER was that the licensee has satisfied the concerns of IEB 80-06, subject to the successful completion of the licensee's commitment to implement and test the proposed design modifications and

verification that all equipment is consistent with reviewed schematics (via testing).

The referenced design modification is being processed under ECN P-0469. When questioned, the licensee stated that the modification was not yet complete and was not scheduled to be completed prior to Unit 2 startup. It was later determined that the modification is in a non-scheduled status, i.e., no schedule exists for completion of the modification. The licensee is currently following up on this.

Copies of the test reports, 480 volt load analysis for the diesel generators and safety evaluations justifying the exceptions to ESF reset noted in the correspondence are currently being obtained from the licensee for review. Closure of IEB 80-06 cannot be completed until the TIP modification is installed and tested.

(OPEN) Unresolved Item (259, 260, 296/87-27-02) Transportation, Control of Contamination, and Inadequate Radiological Surveys Associated With Carbon-14 Tracer Used At the Browns Ferry Biothermal Research Facility. This item to be left open for Regional Inspection follow-up and potential enforcement action. Additional information obtained as of November 3, 1987 is given below:

- The biothermal research facility (BRF) was apparently released for uncontrolled use about September 1985.

- Low levels of beta contamination were found in the August 1985 survey on blotting paper in the fume hood and in the refrigerator. The highest swipes found were 300 DPM/100 CM2 (in the refrigerator), which is less than the Muscle Shoals Administrative limit for beta emitters (1000 DPM/100 CM2).

- The fume hood and refrigerator were decontaminated between August and September 1987 - even though no contamination levels 1000 DPM/100 CM2 were found.

- Questioning of Mr. Armitage, the researcher using the C-14 at BRF, by Mr. R. Maxwell of Muscle Shoals resulted in Mr. Armitage stating that no C-14 contamination was ever flushed down the sink of fume hood drains. August 1987, work by TVA, consisting of cutting out section of sink drain pipe and fume hood exhaust pipe, seem to verify Mr. Armitage's statement. The sink drains did go to normal sanitary sewage system during the time testing with C-14 was in progress.

- The amount of C-14 source material shipped to the BRF was 10 millicuries (10,000 uci). For background purposes, the concentration and body burden limits in 10 CFR 20 App B&C for C-14 are 8.10-4 uci/ml and 100 uci respectively.

- The BRF was added to the Muscle Shoals byproduct material licensee (01-16821-02) in 1983. All tracer work at BRF should have been conducted under that license, not the nuclear plant's license.

- Although work with the C-14 apparently stopped in October 1984, the remaining C-14 was not moved to Muscle Shoals until about September 1985.

- Per Mr. R. Maxwell (Muscle Shoals), Mr. Armitage ordered the C-14 from the supplier using the nuclear plant license, thereby failing to make cognizant Muscle Shoals personnel aware of his activities and need for Health Physics coverage.

(CLOSED) Open Item (259/85-28-07) Delegation of PORC Chairmanship. This item was opened because the delegation of the PORC Chairmanship did not appear to meet the intent of Technical Specification (TS) 6.2.B.1 which only allows a representative of the plant superintendent to serve as Chairman in the absence of the plant superintendent. TVA interpreted the word "absence" as meaning from the meeting while NRC interpreted it to mean absent from the site. TVA has since revised and received NRC approval for the applicable portion of the TS. The change is part of an overall upgrade to the TSs and will be implemented in December 1987 as Section 6.5.1.2. The applicable revisions by BFNP unit are: U1 - Rev. 138, U2 - Rev. 134, U3 - Rev. 109. The applicable old section of the Technical Specifications was supported by Standard Practices procedure BF-1.10. A new procedure, PMI-7.1-01 will replace BF-1.10 at the same time the TS revisions become effective. PMI-7.1-1 clearly defines PORC membership qualifications, quorums, responsibilities, and composition. Complying with the revised TS, Appendices 1 and 2 to PMI-7.1-01 clearly define PORC Composition Requirements and list members by title, name, allowed PORC position, and phone number.

(CLOSED) Inspector Followup Item (259, 260, 296/84-32-03) Drywell Pressure Transmitter is in PSIG and Control Room Indicator is in PSIA. This item was opened when the NRC Inspector noted that one of the drywell pressure transmitters and the control room indication in the control room did not agree with each other. At present, when the transmitters are calibrated the control room recorders and meters are set to 14.7 PSIA with the sensing side of the transmitter vented to atmosphere. The licensee has performed an analysis, based on historical weather data (barometric pressure) in the area, which shows that the maximum indication error would be .435 PSI. This is less than the reading accuracy of the scales on the installed control room indicators. While this analysis does not fully resolve the initial concern, a review of the Detailed Control Room Design Review (DCRDR) results shows that additional corrective action will be taken. DCRDR Human Engineering Deficiency (HED) No. 201 addresses the same problem identified by the NRC Inspector. HED 201 has been assessed as Category 1 and the PI-64-67 and PR-64-50 instruments will be changed to PSIG scales. ECN-P 3058 will accomplish correction of HED 201. The ECN is scheduled to be completed prior to the startup of each of the three BFNPs.

(CLOSED) Inspector Followup Item (259,260,296/87-11-03) Welding PQRS Lack Adequate Referencing Information. This item was opened when the inspector noted that several PQRs have the same reference number and that one of the PQR set (GT88-01) showed a backing strip instead of an open butt. It is

TVA's practice to number like PQRs with the same reference number and then differentiate among them, based on modifications to essential variables, with the PQR date. The rational for this practice is that it avoids revising Detail Weld Procedures (DWPs) referenced PQRs as new PQRs are developed or existing PQRs qualified for new essential variables. The original IFI suggested that the licensees should number PQRs in a manner which would relate them to dependent DWPs. This would be impractical, since one PQR (or a set of PQRs) may be referenced in many DWPs. Interviews of BFNP welding engineers showed that they clearly understand the existing numbering system.

The explanation of the backing strip being shown on the PQR is as follows: Any of the GT88-0-1 PQRs which were qualified using a backing ring on the speciman must have the backing ring shown on the PQR. The use of the backing ring remains optional with respect to any dependent DWP. The use of the backing ring on a particular weld is then determined by the individual welders qualification and the details of the DWP. Per ASME Section IX,QW-402.2, the backing ring for the subject welds is a non-essential variable.

(OPEN) Inspector Followup Item (259,260,296/83-56-03) This item resulted from a review of the snubber inspection and testing program for Unit 1, in particular the failure analysis that is performed on failed snubbers. The inspector found that no specific guidance was provided for site personnel to indicate all areas to be investigated when a failed snubber was encountered, the behavior of the snubbers were not consistent with the declared cause of failure, the vendor was not contacted for assistance in analyzing snubber behavior, and previous test data was not investigated to determine if unusual behavior might be indicated.

The licensee presented information in a Commitment Closure Summary consisting of modified procedures SI-4.6.H-1, and SI-4.6.H-2, Functional Test of Hydraulic and Mechanical Snubbers. The modification to procedure SI-4.6.H-1 is not pertinent to this IFI. The modifications to procedure SI-4.6.H-2 improve the engineering evaluation of each inoperable snubber by requiring the notification of the Division of Nuclear Engineering of the failure (in case reanalysis of piping or restraint requirements are called for), asking if the vendor needs to be contacted in case the failure is not obvious and requiring the initiation of a Condition Adverse to Quality Report (CAQR). There is not, however, a requirement to review previous test data to determine if unusual behavior or marginal but acceptable performance was exhibited. Additionally, no specific guidance has been incorporated to indicate areas to be investigated when failed snubbers are encountered. The licensee committed to modify procedure SI-4.6.H-2 and Data Sheet 4.6.H-2-3 under procedure change request SI 4.6.H-2-06 to add these requirements. This item remains open pending approval of the procedure change.

(CLOSED) Inspector Followup Item (84-13-2) Improved Water Chemistry Procedures. Procedure improvements required to close this IFI were reviewed by NRC during Inspection 86-07. This item was verbally closed at the exit meeting on 21 February 1986. While the item is not on the NRC

Browns Ferry Open Item List it remains open in TVAs records because it was not formally closed in the 86-07 Inspection Report. Per discussion with cognizant Region II Inspectors, this item is closed.

(CLOSED) Inspector Followup Item (259,260,296/86-05-03) Failure of SBGT Train A and CREV Train A to Initiate during Inadvertent Trip of Containment Isolation Logic (Group 6) Caused By Reactor Building Ventilation Radiation Monitor Calibration SI. This item was opened when the SBGT and CREV Train "A"s failed to initiate at the same time as SBGT Train B & C, and CREV Train B initiation, and a Group 6 isolation. These initiations were caused by an instrument technician shorting the power supply terminals in the Reactor Building Vent Monitoring System during surveillance testing. TVA reported this event to NRC in LER 86-003-00 on February 21, 1986. The LER was closed in NRC Inspection Report 87-14 with the following statement:

"The inadvertent containment isolation was caused by the voltmeter lead slipping and momentarily shorting the power supply to ground. Corrective action consisted of a rewrite of the Surveillance Instruction (SI), critique of the event by the instrument maintenance personnel, and verifying correct functioning of the logic."

Further investigation of this event has resulted in the following additional information: Special Test No. 8606 was performed in June 1986 to demonstrate that a single downscale trip of the Reactor and Refueling zone radiation monitors will not initiate Containment Isolation, SBGT or CREV. The test showed the logic to work satisfactorily. The logic for these ESF initiations requires one high trip or two downscales or one instrument in the "out of service" mode. This last feature in the logic necessitates jumpering the relays during surveillances. Discussions with TVA I&C personnel and a review of the trip logic diagrams shows that in order for any of the ESF system start logics to have been initiated, the K37 or K38 relays in the radiation monitoring system logic would have had to de-energize. These relays cannot be re-energized without performing a manual reset. It is now believed that the shorting of the power supply caused both a downscale and a high trip on RM 90-141 and RM 90-143, even though initial investigation showed that there was no high annunciator sealed in. It is believed that the failure of SBGT and CREV trains A to initiate may have been a problem in their individual start logics. This logic was not evaluated in the TVA followup investigation of the event, but has been tested since. There is an outstanding DCN to install a banana plug panel on the radiation monitor panels to further reduce the probability for error during surveillance testing (i.e., eliminate the use of internal jumpers and alligator clips). While the root cause of the problem may have escaped the TVA investigation, the failure to initiate SBGT and CREV is not repeatable. This item is closed.

(CLOSED) Inspector Followup Item (50-259,260,296/84-52-06). This item resulted from an inspection of an operational event involving a failure of the Reactor Core Isolation Cooling (RCIC) Inboard Steam Isolation Valve (1-FCV-71-2) to open on March 21, 1984. The inspector found that procedures and/or plant equipment were inadequate to allow recharging (or

warming) of the RCIC steamline downstream of the inboard isolation valve following an isolation at full reactor pressure. The licensee presented information in Commitment Closure Summary SLT 05093005 consisting of an engineering study of the operational event and revised operating instruction OI-71 for Units 1, 2, and 3 incorporating one of the recommendations of the study. The study presents a detailed analysis of the forces seen by the valve operator in closing and opening the valve and concludes that under normal isolation conditions the existing operator would be capable of performing both funtions, even with full reactor pressure on the upstream side of the inboard isolation valve with the outboard isolation valve open. The study then goes on to look in detail at the operational event, in which the outboard isolation valve was closed first to isolate steam leaks and then later the inboard valve was closed. The study concludes that when the inboard valve was closed, condensed steam was present in the line which caused pressurization of the bonnet. When the upstream portion of the line was drained prior to warming, the temperature difference of steam upstream of the valve disc and water downstream of the disc led to a thermal binding of the disc and body due to uneven temperature distribution. Thus, the combination of bonnet pressurization and thermal binding resulted in opening forces which exceeded the capacity of the valve operator. This would not have occurred if the steamline had been drained prior to the closing of the inboard valve or if the inboard valve were closed before the outboard valve.

The following recommendations to procedures and/or hardware were made:

a. Initially isolate the line by closing the inboard isolation valve, FCV-71-2. If, for any reason, FCV-71-3 is closed first, explicitly require that the line be drained before closing FCV-71-2.

b. Modify the valve discs to relieve bonnet pressure to the upstream line. This can be done by machining the disc.

c. Replace the solid wedge with a split disc.

Recommendation a. has been incorporated into OI-71 for all three units and is acceptable. Recommendations b. and c. have not been implemented. In an internal memorandum (Tucker to Pratt, 1/13/87) justification was provided for deferring physical modifications until generic TVA studies are completed for Sequoyah related to bonnet pressurization and thermal binding and the best physical modifications determined which could then be applied to the Browns Ferry valves. This is acceptable considering the procedural modification which should prevent reoccurrence of the operational event in 1984 and the fact that the nuclear safety function of valve closure is not impacted by the problems encountered during the event. This item is closed.

(CLOSED) TVA Commitment to NRC in Letter dated March 26, 1986 to Upgrade EOIs to include ATWS and to conduct related Operator Training (TVA Action Item No. NCO-86-0142-001,002,003).

The following brief chronology outlines the final development of symptomatic EOPs and the inclusion of ATWS in the EOPs and in the operator training program:

June 1984: Procedure Generation Package (PGP) submitted to NRC (IAW GE EPGS, REV 3).

April 1985: TVA implemented EOIs (TVA specific acronym for EOPs) except for steps relating to ATWS, Secondary Containment Control, and Radiation Release Control.

March 1986: TVA letter to NRC requesting extension from April 1986 to "prior to unit startups" for implementing ATWS procedures and completing associated ATWS training.

June 1986 - June 1987: Groups 1, 2, and 3 of licensed operators completed classroom training and simulator training on EOIs with ATWS included.

March 1988: Group 4 (final group) of licensed operators scheduled to complete classroom and simulator training on EOIs with ATWS.

The steps necessary to manually insert control rods by diverse methods are included in Sections RC/Q-4 and 5 of EOI-1. The procedures for controlling reactor vessel level during an ATWS to control power are included in Section C-5 (Contingency) of EOI-1 in the Level Control section. Training records were reviewed to verify that operators have received classroom training on ATWS. Simulator training records were reviewed to verify that operators have been trained using several variations of ATWS scenarios. The Browns Ferry simulator is able to respond fairly well to ATWS events through the severity level of 100% ATWS, MSIV closure, at 80% Power. The fourth and final group of operators requiring ATWS training are scheduled to complete all training by March 1988. Although one group remains to be trained, records of past training and the training schedule support NRC considering the subject commitment to have been met.

The TVA to NRC letter dated March 26, 1986 stated that EOIs had been written without Secondary Containment Control and Radiation Release Control considerations. During the process of confirming that ATWS related EOI development and training commitments are being met, progress on the other two issues was reviewed. TVA has developed draft procedures EOI-3 (Secondary Containment Control) and EOI-4 (Radiation Release Control). EOI-3 appears substantially complete. EOI-4 will require additional development. TVA has made no commitment to implement these two procedures prior to restart of any BFNP unit. Procedure EOI-3 and EOI-4 need to be completed and implement prior to unit restart. This item will be tracked as an Inspector Followup Item (260/87-42-01).

(CLOSED) Violation (259,260,296/87-26-01). This violation resulted from inspector review of maintenance activities and failure investigations associated with the inoperability of several fire protection dampers in the diesel generator rooms. The violation stated that: 1) although the root cause of fire damper failure was attributed to excessive dust

accumulation of the damper blades, the dust was not removed from the dampers. The significant condition adverse to quality (SCAQ) was therefore not corrected; 2) as of June 15, 1987, neither the Plant Operations Review Committee nor the Plant Maintenance Superintendent reviewed the March 16, 1987, failure investigation report which documented the root cause and corrective action recommendations. The SCAQ was therefore not reported to appropriate levels of management; 3) no closure or tracking mechanism existed for corrective action recommendations documented on the failure evaluation. Thus, corrective actions to preclude repetition was not taken in a timely manner.

The licensee responded to the violation in their letter of August 27, 1987. In response to item 1, they stated that the root cause of the fire damper failures was a "collection of dust" rather than "excessive dust accumulation on the damper blades." The fire protection engineer who wrote the Failure Investigation Reports 8713, 8714, 8715 and 8716 determined that dust buildup on the damper pivot arms at the sealed bearings due to previous lubrication and infrequent cycling was the cause of the failure, rather than any dust present on the damper blades themselves (which were not cleaned at the time of the failure investigations in March 1987). However, the reports were not specific in stating this as the root cause. The licensee committed to revise the failure investigation reports to correctly state the root cause. This was accomplished by Revision 1 to the reports dated August 14, 1987, and provided by the licensee in Commitment Closure Summary package. Confirmation of the operability of all fire dampers was made on March 6, 1987, by maintenance personnel and a member of the fire protection staff through MRs A-791431 - 791446, after correction of the four inoperable dampers addressed in the failure investigation reports under MRs 761979 and 761981 - 761983. The original two inoperable dampers found on March 3, 1987 and repaired under MRs A-759311 and 785752 never had failure reports initiated for them, although they were determined operable on March 6, 1987. The licensee initiated a Condition Adverse to Quality Report (CAQR) No. BFN 870458D02 on July 7, 1987, to address the failure to take corrective actions as recommended in the failure investigation. The corrective action committed to by the Fire Protection Unit was to implement a preventive maintenance program for the emergency diesel room fire dampers by October 1, 1987. Procedure FP-0-39-PM-001 was approved on September 25, 1987. A discussion of the specifics of the PM procedure will be addressed below.

In response to items 2 and 3, the licensee state that they would revise Plant Maintenance Instruction 6.13, "Failure Investigation of Safety-Related Items" to require the Maintenance Superintendent; designee to review the investigation report for completeness and acceptability of the recommendations, and also ensure the failure investigation remains open until recommendations and corrective actions are completed or dispositioned. CAQR No. BFN 870458D01 was initiated on July 7, 1987, to ensure completion of the procedure revision as the corrective action.

Revision 0001 of PMI-6.13 was approved on August 26, 1987, and contains the items described in the licensee's letter. PMI-6.13 was actually

initiated as a new procedure on July 21, 1987, as a result of this investigation, replacing Standard Practice BF 6.18.

Discussions were held with licensee representatives concerning the failure investigation program to determine what type of failures would lead to the generation of a Failure Investigation Report or root cause determination. PMI-6.13, Section 1.2, Scope, states that failure investigations shall be performed on all failed items identified as LERs as a minimum, and when a failure occurs on a significant component which could result in load reduction or unit shutdown, or when maintenance history evaluation indicate a failure trend. Additional clarification is provided in Section 4.0, Instructions, which states that failure investigation should be performed routinely on all safety-related components where there appears to have been a materials failure or a component or item failed to perform its intended function.

Another means of determining root cause for failures would be the Corrective Action program, SDSP-3.7. This program, of which the CAQR is the key document, focuses on all conditions adverse to quality including maintenance problems.

Since fire protection systems and equipment are defined under BF 1.11 as safety-related in areas where critical equipment is being protected, a failure investigation and/or a CAQR should be generated when a failure of the equipment to operate occurs. The definition of equipment will be improved by a revised approach to Q-list definition, which is in process.

As part of the corrective action to the failure investigation program, the licensee initiated a report from Planning and Scheduling to the maintenance groups, the Maintenance Superintendent and Technical Support to show on a monthly basis the status of the Failure Investigation Reports. A report was generated November 17, 1987 for the inspector's review showing the status of all failure investigations from the beginning of 1986 to date. This is an added management tool for assessing progress on completing reports and corrective actions, but is weak in that no indication is given for a report not being completed within a reasonable timeframe, i.e., several months, nor is information provided to show the reason for the lengthy resolution time. Discussion with maintenance supervisors indicated that a followup process on failure investigations was in existence but was not formalized. The Maintenance Superintendent appears to have overall control of the program but not all of the organizations that may generate Failure Investigation Reports. There is concern that the process for resolving failures and implementing corrective actions in a timely manner may not be sufficiently detailed and rigorous. Resolution of this issue will be provided during a subsequent inspection and is being tracked as Inspector Followup Item (260/87-42-02).

One concern addressed in the previous inspection was that the Plant Operations Review Committee (PORC) had not reviewed the circumstances surrounding the damper failure. Discussions with the licensee indicated that, whereas failure investigations generated under PMI-6.13 are not routed on a routine basis to PORC, all CAQRs and LERs are reviewed and

approved by PORC along with operational critiques. The review of CAQRs and operational critiques have been instituted since March 1987 and should give added assurance that significant equipment failures are adequately reviewed by the committee.

The preventive maintenance program for the diesel generator room fire dampers was reviewed to determine if it incorporated the corrective actions noted in the failure investigation reports. Procedure FP-0-39-PM-001 is performed on a quarterly basis and "allows for the inspection, lubrication and manual cycling of the dampers to insure (sic) operability in the event of CO2 discharge." (Section 1.2) The procedure specifies the use of Aero kroil lubricant or equivalent. Aero kroil was selected by the licensee due to its cleaning and lubricating properties and the expectation that dust collection and buildup would be minimized. An "or equivalent" lubricant could not be readily defined by the licensee, leaving the possibility that an unsuitable lubricant could be used. All dirt accumulation is to be cleaned from damper blades, linkages, frame and other areas and then the lubricant is to be sprayed sparingly into and around linkages, swivels and other moving parts. Any excess lubricant is to be wiped off. Actuation of the damper is performed several times to verify all moving parts free, move easily and are properly adjusted. Any adjustments made, along with fusible link and negator clip conditions, are noted on the procedure data sheet. The licensee confirmed that the PM would be referenced on any MR that would be generated in the future for an inoperable diesel generator fire damper as the first approach to returning the damper to service.

The PM was performed 11/6/87 on all 16 fire dampers. A copy of the completed PM data sheets were reviewed and an inspection of one of the intake dampers was made. No concerns were identified, and the PM procedure addresses the corrective actions identified in the failure investigation reports. The concerns of this violation are closed.

(CLOSED) Inspector Followup Item (259,260,296/85-15-11) This item arose from inspector review of three Licensee Event Reports (LERs) that failed to include information required by 10 CFR 50.73(b). The licensee stated that a checklist would be developed to ensure all reporting requirements of 10 CFR 50.73(b) would be included in future LERs. The licensee provided Commitment Closure Summary SLT 850925009 which included a copy of Plant Operations Review Staff Section (PORS) Instruction Letter No. 15, Handling of Licensee Event Reports (LERs) and copies of the original and Revision 1 versions of the subject LERs. It was confirmed that all three LERs and their revisions have been closed out by subsequent inspections. A checklist (Attachment 7) was added to PORS 15 and appears suitable in conjunction with other guidance in the instruction to ensure that future LERs will contain the information required by 10 CFR 50.73(b). This item is closed.

4. Unresolved Items (92701)

There were no unresolved items identified during this report period.

5. Operational Safety

The inspectors were kept informed of the overall plant status and any significant safety matters related to plant operations. Daily discussions were held with plant management and various members of the plant operating staff.

The inspectors made routine visits to the control rooms when an inspector was on site. Observations included instrument readings, setpoints and recordings; status of operating systems; status and alignments of emergency standby systems; onsite and offsite emergency power sources available for automatic operation; purpose of temporary tags on equipment controls and switches; annunciator alarm status; adherence to procedures; adherence to limiting conditions for operations; nuclear instruments operable; temporary alterations in effect; daily journals and logs; stack monitor recorder traces; and control room manning. This inspection activity also included numerous informal discussions with operators and their supervisors.

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

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

6. Maintenance Observation (62703)

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

Maintenance requests were reviewed to determine status of outstanding jobs and to assure that priority was assigned to safety-related equipment maintenance which might affect plant safety. The inspectors observed the below listed maintenance activities during this report period:

a. 480 VAC and 250 VAC circuit breaker testing in accordance with EMI 7 amd MMI 6. - Unit Common

b. RPS MG set testing - Unit 2

c. Drywell Blower Damper Testing - Unit 2

No violations or deviations were observed in this area.

7. Reportable Occurrences (90712, 92700)

The below listed licensee events reports (LERs) were reviewed to determine if the information provided met NRC requirements. The determination included: adequacy of event description, verification of compliance with technical specifications and regulatory requirements, corrective action taken, existence of potential generic problems, reporting requirements satisfied, and the relative safety significance of each event. Additional in-plant reviews and discussion with plant personnel, as appropriate, were conducted. The following licensee event reports are closed:

LER No.	Date	Event
259/77-03, Rev. 1	1/4/77	Liquid Release from RHR Service Water Heat Exchanger 1A
259/80-90, Rev. 1	12/10/80	Continuous Air Monitor Declared Inoperable
259/84-25, Rev. 2	6/15/84	Inadequate Isolation of Building Heat System Between Reactor and Turbine Building
259/85-13	5/14/85	Temporary Startup Test Panel Installation
259/87-02	1/31/87	Engineered Safety Feature Actuation Because of Personnel Error
259/87-06	3/2/87	Improper Flow Testing of Control Room Emergency Ventilation Fan Leads to Prohibited Condition

259/87-11	4/24/87	Inadequate Inspection Program Results in Inoperable Check Valves in the Emergency Equipment Cooling Water System
259/87-20	8/2/87	Primary Containment Isolation Due to Reactor Water Cleanup Temperature Switch Failure
260/87-04	6/24/87	Primary Containment Isolation Due to Reactor Water Cleanup Instrument Drift
296/82-57, Rev. 1	11/28/82	Continuous Air Monitor Declared Inoperable

A licensee inspection revealed that an RHR heat exchanger (LER 259/77-03) gasket leak occurred because the stud bolts had become loose in service. In repairing the leak and replacing the gasket, locknuts were installed on each stud bolt to help prevent future loosening. This modification was made to all plant RHR heat exchangers when they were opened for maintenance or inspection. Also, the color of all process radiation monitoring annunciator windows were changed to distinguish them from other annunciators. The event was reviewed by all licensed operators. Chemical laboratory sampling procedures were revised. The results of the liquid effluent analyses which exceed limits will be given to the shift engineer in a written as well as verbal form.

The continuous air monitor (CAM) for the reactor/turbine ventilation exhaust (LERs 259/80-90 and 296/82-57, Rev. 1) was declared inoperable due to water in the sample line. Browns Ferry is purchasing new CAMs which will have a water trap on the inlet line. Until the new CAMs are installed, the CAM hoses will be checked for condensation during source background filter checks which are performed weekly.

A review of the building heat systems (LER 259/84-25) by Engineering Design during the Bulletin 79-14 inspection determined that the isolation between the turbine and reactor building heat system lines were inadequate. These six hot water penetrations now fall within the bounds of a larger program described in LER 259/86-24. The licensee stated that until this issue is resolved no unit will start up.

Problems were identified by Engineering regarding the existing configuration of the startup test instrumentation panels (LER 259/85-13). Corrective actions taken by the licensee included a modification of the panel installation, and a design change request.

Personnel performing a relay calibration (LER 259/87-02) omitted a step in the procedure which caused an engineered safety feature actuation. Disciplinary action in the form of a letter of warning was given to the responsible test engineer. All Operations and Division of Power System Operations personnel were required to review the event critique.

Confusion regarding the correct zeroing procedure for flow testing of control room emergency ventilation fans (LER 259/87-06) has been resolved, and the personnel involved have been instructed on the correct use of the instrument. The surveillance procedure was revised to use a pitot tube traverse and micromanometer in lieu of the rotating vane instrument.

An inadequate inspection program of the emergency equipment cooling water system (LER 259/87-11), which did not verify disc free movement, is the root cause for the inoperable condition of the carbon steel valves. The Unit 2 valves are stainless steel and were found to be operable. SIs have been written to inspect these valves at least once per operating cycle. A program for changing out small diameter carbon steel valves with stainless steel valves is planned.

A primary containment isolation (LER 259/87-20) was caused by a reactor water cleanup temperature switch failure. An amplifier board in the power supply of the temperature switch failed. The power supply amplifier board was replaced and the temperature switch was recalibrated. A review of the maintenance history of the temperature switch showed no previous failure of this kind.

A primary containment isolation (LER 260/87-04) was caused by a reactor water cleanup instrument drift. A spurious signal from a temperature indicating switch monitoring the water temperature at the outlet of the non-regenerative heat exchanger caused the isolation. The switch was found to be out of calibration. The temperature indicating switch was recalibrated and placed back in service.

8. Restart Test Program

a. During the performance of Restart Test Procedure (RTP) 2-BFN-RTP-082 "Standby Diesel Generator," for Diesel Generator (DG) 3A, the DG field excitation breaker tripped open and shut down the DG. The DG had been running at 2950 kilowatts (kw) output for approximately one hour prior to the field breaker opening. The breaker was removed, tested and re-installed; however, it showed an open A phase prior to performance of the test. A new breaker was obtained from stores (the breaker is a molded case type equipped with a shunt trip); however, the shunt trip coil was found to be defective. Another new breaker was obtained from stores and on November 19, 1987, the inspector witnessed the testing of the third breaker and the disassembly of the original breaker. The third breaker was tested in accordance with Electrical Maintenance Instruction 7.2 "Test Procedure For Initial Installation and Troubleshooting of Molded Case Circuit Breakers." The inspector observed that the breaker testing was performed in accordance with the approved procedure and that the procedure was being followed by maintenance personnel. The disassembly of the original breaker revealed that the open A phase was due to a mechanical failure, which appeared to be metal fatigue of the thermo trip element.

Additional test instrumentation was installed by the RTP Group and the third breaker was reinstalled. On August 14, 1987, the test was resumed; however, at 2950 KW with a .8 power factor the field breaker indicated 95 amperes and 149 degrees F temperature. This placed the breaker within the Minimum Total Clearing Time and the Maximum Total Clearing Time of the General Electric, Molded-Case Circuit Breaker, 100 line graph for the field breaker. The Test Director determined that with the temperature continuing to rise the breaker would trip at any time, therefore the test was stopped. The RTP Group and the Unit 2 plant staff will evaluate data and determine further action as needed.

b. Special Test 87-35

The inspector observed Special Test 87-35 performed to determine and verify the capability of Unit 1 and 2 of diesel generator D to accept its emergency loads in incremental steps. This was test #3 in a series of four (4) test, with each increment representing approximately 250 kw starting from a base of 2100 kw. This on-going testing and test observations by the resident inspector is discussed in NRC Inspection Report Nos. 50-250/87-33, 50-260/87-33 and 50-296/87-33. The test group was briefed by the Test Director during which the group was informed that this test was being performed to verify that Unit 1 and 2 diesel generator D would accept it's identified loads, would not overload and would not trip during the test. Personnel then reported to the assigned test stations such as DG room, switchgear rooms and the Unit 1 and 2 control room. The inspector observed that during the test the D diesel generator did not trip, accepted loads up to approximately 2400 kilowatts (as indicated by the control room meter) and did not trip. The initial review of the raw data appeared satisfactory, however, the final test results will be reviewed when available.

c. Diesel Generator Load Acceptance, Air Start and Load Reject Tests

The inspector observed the performance of portions of 2-BFN-RTP-082, Standby Diesel Generator, as it applies to diesel generator 3D load acceptance test and load reject test. The inspector also observed the performance of portions of 2-BFN-RTP-082 as it applies to diesel generator 3C air start test. Both tests were performed to approved procedures and test personnel displayed a thorough knowledge of the systems under test. Other than a test exception involving an obvious typographical error the tests were performed as is. Initial review of raw data indicated a successful performance of the tests; however, a final review will be conducted when final test results are available.

9. Inspection of Engineering Changes and Modifications

The purpose of this inspection was to ensure that Engineering Change Notices (ECNs) and resulting plant modifications required to support Browns Ferry Unit 2 restart are being adequately accomplished in

accordance with the licensee's quality and engineering assurance programs. This assessment is based upon inspection of ECN and plant modification packages that have progressed completely through the design process and are being (or have been) installed.

The inspection activities reported herein involved represent the first phase of a two part inspection. The second phase of the inspection will be conducted during another site visit in early January 1988.

The licensee has previously experienced breakdowns in the facility's configuration management as documented by prior NRC inspections, Systematic Assessments of Licensee Performance, and the licensee's Nuclear Performance Plan. The inspection evaluated a sample of ECNs and modification packages for the High Pressure Coolant Injection (HPCI) System, Reactor Core Isolation Cooling (RCIC) System, the Core Spray (CS) System, and Reactor Building Closed Cooling Water (RBCCW) System.

Specific reference material reviewed during the inspection is listed in Appendix A to this report. The inspection consisted of a review of:

1) ECN/Modification Processes - These processes have undergone progressive and major changes over the past several years as the licensee has taken action to correct prior deficiencies. This has resulted in the establishment of transitional programs for control of engineering and modification leading to an eventual "final" program to be implemented in early 1988. The engineering, construction, and administrative control processes and status of this transition were evaluated.

2) Review of ECN/Modification Packages - Engineering Change Notices, Work Plans & Inspection Records (WP&IRs), Post Modification Testing, drawings, and related documents comprising the "packages" were reviewed for completeness, technical adequacy, and conformance with plant instructions.

3) Inspection of Field Activities - Work in process and the as-constructed condition of selected modifications were inspected with respect to compliance with modification procedures and quality of the work.

4) Quality Organization Involvement - Quality Assurance, Quality Control, and Engineering Assurance overview, audit and surveillance activities were reviewed.

5) Organizational Interfaces - As part of ongoing organizational changes, expansions, and program transitions (above), the participation of each involved organization was reviewed with respect to flow of information and overall satisfaction of program requirements.

6) Program Status and Overview - The overall status of modification program activities was reviewed with respect to the overall numbers and status of ECN/Modifications and the licensee's satisfaction of

commitments made to NRC in the TVA Nuclear Performance Plan, Volume III.

This report addresses the inspection to date for items 1-3 above. Items 4-6 and the completion of inspection for items 1-3 will be reported during subsequent inspections.

Prior difficulties with loss of configuration management control and conformance with the as-licensed design basis are being addressed by a licensee Design Basis Verification Program (DVBP) as part of the licensee's commitments to NRC contained in the TVA Nuclear Performance Plan, Volume III. NRC is reviewing these activities separately. This inspection reviewed only those aspects of the program which were applicable to the specific modifications reviewed, i.e., interim programs for assuring that design inputs and outputs accurately reflect the as-constructed plant, etc.

Engineering Change Notices are the vehicle established to control the development, approval and transmittal of design information for plant modifications. Work Plans and Inspection Records (WP&IRs) are the work instructions for installation of the modifications. For the purposes of this report, the ECNs and WP&IRs are collectively referred to as "modification packages" or "work plans" respectively.

The following modifications were selected for detailed implementation review:

[NOTE: Unit 2 only; all valve and equipment numbers prefixed by 2-]

ECN	Subject
P0157	Removal of Core Spray (CS) Pump Motor Oil Coolers
--	WP 6619 & 2018-84 - Remove Coolers
P0795	Installation of Valves to Permit 10 CFR 50, App. J Testing of CS Valves 75-606, -607, -609, and -610
--	WP 2117-86 - Piping & Valves
--	WP 2118-86 - Piping & Valves
--	WP 2119-86 - Piping Supports
--	WP 2120-86 - Hydrostatic Test
P02039	Replacement of CS Solenoid Valves FSV-75-57 and -58 with environmentally qualified units
--	WP 2174-84 - Electrical
P0959	Installation of Valves to Permit 10 CFR 50, App. J. Testing of RBCCW Valves

-- WP 2121-87 - Piping & Valves
-- WP 2122-87 - Drywell Area Supports
-- WP 2123-87 - Torus Area Supports
-- WP 2124-87 - Reactor Building Area Supports
-- WP 2165-87 - Hydrostatic Test

L2003 Replacement of Stainless Steel Core Spray Piping and Safe Ends with Carbon Steel

-- WP 9244 - Same as Above

P0162 Remove Auto Initiation Opening Logic from Normally Open HPCI Steam Line Isolation Valves

-- WP 2005-86 - Test FCV 73-3
-- WP 2100-85 - Test FCV 73-2
-- WP 2144-85 - Electrical Work

P0652 Replace F[illegible]C Valve FCV 71-40 with Pneumatic Operated Soft Seat Check Valve

-- WP 2054-84 - Replace Valve
-- WP 2148-85 - Electrical Work

P0651 Replace HPCI Valve FCV 73-45 with Pneumatic Operated Soft Seat Check Valve

-- 2153-84 - Replace Valve
-- 2147-85 - Electrical Work

P0153 Reroute Cabling to Separate HPCI and Automatic Depressurization (ADS) Components (App. R)

-- WP 2166-85 - Cable Relabeling
-- WP 2005-85 - Install New Conduit
-- WP 2084-85 - Pull/Terminate New Cables
-- WP 2085-85 - Rework Cable/Internal Wiring
-- WP 2100-85 - Perform Modification Testing
-- WP 2162-85 - Install Cable Tags

P0965 Installation of Valves to Permit 10 CFR 50, App. J. Testing of HPCI Valves

-- WP 2126-87 - Install Valves & Piping
-- WP 2127-87 - Install Supports

P03061 Replace Level Switches (IEB 79-01B)

-- WP 2193-84 - Replace Level Switches
-- WP 2034-86 - Construct Access Platform
-- WP 2066-86 - Perform Functional Test

P03116 Replace Limitorque Operator Components (IEB 79-01B)

-- WP 2226-84 - Replace Components FCV 73-2
-- WP 2152-87 - Modify Pipe Support
-- WP 2166-87 - Rework Hangers
-- WP 2224-87 - Replace Gears (FCR 73-34, -44)

1) ECN/Modification Processes

Modifications design engineering is accomplished by the Division of Nuclear Engineering (DNE) and its contractors using Nuclear Engineering Procedures (NEPs) as listed in Appendix A, References. The design information is incorporated into an ECN, reviewed and approved by management, and transmitted to the construction forces using several procedures.

Prior to mid-1987, the licensee used BF8.1 for control of ECN preparation. SDSP8.1 did not fully accommodate the output of the Design Basis Verification Program nor the interim need to verify as-constructed plant configuration as part of the design input process. In mid-1987, the licensee implemented a three part transition program to accommodate these circumstances.

SDSP8.1 continued to be applied to modification packages in process prior to April 1987. NEP 6.1, Change Control, Section 2.0, Policy, was supplemented to require design engineers to "walkdown the affected systems as part of the design process" to ensure that the as-constructed configuration was properly reflected in the design inputs and outputs. The inspectors interviewed DNE responsible engineers and task engineers and performed detailed drawing reviews to ensure that this part of the process was effective.

SP 86-03 was issued to provide transitional control of new packages for which design outputs had not yet been issued. The ECNs to which SP 86-03 is applicable were identified by a DNE memo of May 5, 1987, listing approximately 80 transition ECNs. SP 86-03 requires a more formal walkdown and a series of constructability meetings with modifications construction personnel to ensure that the as-designed outputs accurately represent the facility. Although, this procedure is supposed to be limited to the specific transition ECNs, the inspectors found that as new work packages were being developed for "older" ECNs, the above provisions for configuration management were being implemented.

The licensee expects to issue the "final" ECN control procedure in late 1987 - early 1988 which will replace both SDSP 8.1 and SP 86-03, incorporating the results of the DBVP and the various features of the prior programs. These activities were not specifically inspected.

Similarly, development of the construction work plans was being controlled under a two procedure system. BF8.3, Plant Modifications, was issued in March 1986 as the procedure for implementation of work

plans, field change requests (drawing changes) and post modification testing. This procedure continues to be used for those work plans issued prior to Fall 1986. In June 1986, SDSP 8.4, Preparation and Processing of Workplan and Inspection Records, was issued to replace BF8.3 and is being applied to ECNs not subject to BF8.3. The inspectors noted that the transition to SDSP8.4 was well under way and a number of the modification packages for even "older" ECNs included WP&IRs issued per SDSP8.4.

The inspectors noted that, notwithstanding the various applicability criteria applied to the above procedures by TVA management, the procedures are variously applied to in process modifications resulting in the "newer procedures" (SDSP8.4, etc.) being applied to some of the "older" ECNs. The inspectors were further unable to determine procedure applicability for ECNs and Work Plans which predated the above procedures. The inspection for these latter cases was focused, therefore, on the technical completeness and adequacy of the as-constructed modification without regard to the details of procedure implementation.

2) Review of ECN/Modification Packages

The ECNs and work plans listed above were reviewed to determine whether the following objectives were fulfilled:

a) ECNs and Design Change Requests (DCRs) included evidence of proper management review, peer review, and Plant Operations Review Committee (PORC) approval, if required.

b) Unreviewed Safety Question Determinations (USQDs) were accomplished in accordance with applicable procedures, were technically complete and thorough, and met the requirements of 10 CFR 50.59.

c) Design inputs and outputs were reviewed to determine that provisions for other licensee programs (10 CFR 50, App. R, Fire Protection; Environmental Qualification, etc.) were addressed as and where required.

d) Design outputs (drawings, calculations, bills of material, etc.) were reviewed to determine that the as-engineered design was complete and in accordance with DNE procedures (NEPs), the licensee's QA program and applicable license requirements (codes, standards, NRC regulatory positions, etc).

The design outputs were also reviewed with respect to the work plans (below) to determine their conformance to as-found field conditions.

e) Design outputs and ECN Data Sheets were reviewed to ensure that the outputs were properly controlled, reviewed and approved. Field Change Requests (FCRs, drawing and specification changes)

were reviewed to assess the initial quality of the design outputs and the final design. Unreviewed Safety Question Determinations (USQDs), Appendix R checklists, and FCRs were also reviewed to assure a level of review and approval equal to the initial reviews, as applicable.

f) Work Plans were reviewed for conformance to BF8.3 or SDSP8.4 as applicable, including pre-work review and approval, work authorization and control provisions, accurate reflection and use of ECN design outputs, QC inspection provisions, etc.

g) Detailed aspects of work plans were reviewed, on a sampling basis, to ensure that appropriate welding, nondestructive examination, and material quality, procurement, and traceability requirements were incorporated and implemented.

h) Administrative controls and detailed implementing procedures for construction testing and post modification testing were reviewed to verify conformance with SDSP 17.2 and other applicable test guidance such as MAI-54.

   Licensee control of testing, approval of test procedures and evaluation and approval of test results was also reviewed.

i) Administrative controls for review and approval of work completion, preparation transmittal of as-constructed drawings, and final closeout of work packages were reviewed and confirmed to be adequate for the applicable work packages sampled.

d. Findings From Package Reviews

[Note: Inspection of the packages was not complete at the end of this inspection period. Review of work plan data will continue during the next scheduled site visit.]

1) SDSP 8.4, BFEP PI 86-03, and NEP 6.1 (Change Control) variously require pre-design and pre-construction system walkdowns to ensure that existing conditions are adequately reflected by the design outputs and reduce the necessity for drawing changes and rework. These requirements appear to cover modification circumstances currently being processed.

   For typical piping and electrical installations, relatively few FCRs were required to adjust as-designed drawings for field conditions. However, an inordinate number of FCRs appeared necessary to achieve adequate, constructable support designs in for numerous piping hangers and restraints.

   For example, Support No. 47B2464-27, ECN P0959, required nine FCRs to support final installation. ECN P0959, RBCCW Appendix J modifications, required 61 FCRs to correct the design of 27

hangers in WPs 2122-87 and 2123-87 during the period of August - October 1987.

Numerous FCRs were written against WP 2147-95. FCR 87-425 clearly appeared avoidable by performance of a constructability walkdown. The work plan included installation of a steel support for an electrical junction box, connecting to an existing, field-installed conduit. The arm length of the support required change because the as-designed installation was not possible.

Six (6) FCRs were necessary to implement WP 2127-87. Two (2) of the FCRs (87-966 and 87-985) were required for the same reason (increasing length of threaded portion of the support rod). FCR 1192 was required to incorporate existing field conditions not shown on the applicable drawing.

Fifteen (15) FCRs were required through October 21, 1987 for implementation of ECN P0753, Separate HPCI and ADS Cabling. FCRs for subjects such as cable penetrations and Appendix R check-off applicability appeared to be items avoidable by stronger design involvement in constructability.

Discussions with cognizant modifications engineers indicated that, notwithstanding the provisions of NEP 6.1, SP 86-03, and SDSP 8.4 requiring pre-design and/or constructability walk downs, the as-engineered support drawings did not adequately reflect construction conditions (interferences, dimensions, etc.) and were frequently "unbuildable" (e.g. inaccessible welds, etc.). This trend appeared to be continuing in early November 1987.

2) ECN L2003 involved replacement of stainless steel Core Spray piping and reactor vessel penetration safe ends with carbon steel to reduce the potential for intergranular stress corrosion cracking (IGSCC) in accordance with WP 9244.

WP 9244 was released for work on March 31, 1978 and installation appears to have been substantially completed in June 1978. However, the work package was not properly closed.

The package has been assigned to the Special Projects Group (Backlog Group) for review and closure. The inspector conducted a preliminary review of the current status of WP 9244 on November 4, 1987 finding that:

-- The package has not yet been processed by the group. The work plan file contains the original, partially executed work plan and unsorted/uncollated weld data sheets, material certification and traceability records, and miscellaneous installation records and drawings.

-- The partially executed work plan is divided into Parts I and II, addressing installation of piping from FCV 75-26 to -54 and and from FCR 26 and -54 to FCV 75-23 and -51 respectively.

-- The work plan has not been signed off for key installation and testing steps.

-- The hydrostatic test for Part I has not been signed off by QA.

-- Dye Penetrant Testing for Part I welds for FCVs 75-23, -25, -26, -51, -53, -54 have not been signed off by QC.

-- The hydrostatic test for Part II has not been signed off by either the cognizant engineer nor QA.

-- The data package for Part II, including major installation, NDE and testing steps has not been signed off by the cognizant engineer nor QA.

-- The DNE engineering files include internal TVA memoranda dated December 20, 1985, June 23, 1986, July 24, 1986, and October 31, 1986 as attachments to USQD, Revision 2. These memos discuss the apparent omission of several 3/4 inch drain and test lines from the piping replacement activities. The memos indicated that the lines would be replaced and recommended that the replacement not occur until the first outage after Unit 2 restart. The inspector was unable to determine the current status of the formal licensee disposition of the above and will continue to pursue this item during the next inspection. This item will be tracked as an Inspector Followup Item (260/87-42-03).

The work plan as-found status does not appear adequate to have supported a licensee determination of Core Spray System TS operability for periods of operations since mid-1978. Although, the work plan file may contain sufficient information to substantiate completion of the items above, the condition of the records does not permit a ready determination.

There is evidence that a prior review of the package was conducted in 1984; notes attached to various package records identify missing signatures and data. Discussions with the cognizant backlog group personnel indicated that the package is identified for closure prior to Unit 2 restart but has not yet been addressed by the current "backlog" effort. The licensee's plans include detailed verification of the package contents, physical walkdown of the system, and rework/reperformance of work elements that cannot be otherwise verified.

The inspector requested the licensee (Backlog Work Group Supervisor and Compliance Supervisor) to provide additional information for further inspection of this item.

3) Work Plan 2005-86 (Tests FCV 73-3 for ECN P0162) references WP 2100-86 as the procedure which will test the other valve affected by ECN P0162, FCV 73-2. WP 2100-86 does not appear to test FCV 73-2. However, WP 2100-85, issued under ECN P0753, appears to actually test this valve.

   The Unit 2 Startup List Report (cross reference ECNs to work plans) does not reference WP 2100-85 as required to complete ECN P0162. Further, WP 2100-85, Test Description, does not detail the fact that the test will fulfill testing requirements for ECN P0162.

   The incorrect, mutual cross references appear to provide the potential for a change in one ECN/WP causing an undesirable impact or omission on the corresponding other ECN/WP.

4) ECN P0652 replaces control air solenoids for HPCI testable check valve FCV 73-45. Report SCR BFN MEB 8502 is referenced as a potential uncorrected problem in the USQD (Sh. 9, Rev. 2) for ECNs P0651 and P0652. The report notes that control air solenoids for the HPCI valve FCV 73-45 can cycle under certain conditions to actuate or impede valve operation, representing a potential unreviewed safety question. The USQD also notes that implementation of the ECN will <u>not</u> correct the condition.

   However, the design output package is complete, the field installation of the valves is complete (work plans in work process) and neither resolves the USQD problem. Licensee plans for resolution of the above were not known at the end of this inspection and the USQD question remained unresolved.

5) Numerous administrative errors in work plan processing were noted. A few of the noted errors are listed below as examples:

   - Design Change Request No. 2546 recommended replacement of HPCI and RCIC check valves 73-45 and 71-40, respectively, with dual seat, pneumatically operated valves. DNE split the DCR into two ECNs: P0652 for RCIC and P0651 for HPCI. This was done without explanatory documentation in the design output documents. Thus the RCIC valve replacement now carries an Outstanding Work Item Number (OWIN) of 2-73-88, a system designator for HPCI.

   - Work Plan 2151-85 was referenced by WP 2053-84 (on work plan specification sheet, plan page 1 of 30) as the work plan which tests the work accomplished under WP 2053-84 (leak and hydrostatic test of FCV 73-45). Revision 13 of WP 2053-85 cancelled WP 2151-85, modified the procedure

appropriately, but did not revise the work plan specification sheet (above). A similar condition was noted for Work Plan 2054-84.

- Field Change Request 87-425 was initiated on May 11, 1987 by the responsible field engineer. The date of review by the responsible design engineer was entered as May 19, 1986.

6) Work Plan 2148-85 included a stores requisition (TVA 575 Form) for six (6) 3/4-inch expansion shell anchor bolts (SSDs). Reference to the appropriate design specification and applicable drawing showed the anchors should be seismic qualified QA Level II.

The TVA 575 Form did not specify any quality level for the anchors nor did the Data Sheet (MAI-4, Attachment A) for the anchors reflect that the proper material was used.

7) ECN P0965 installed HPCI system block and test valves to permit local leak rate testing of FCV 73-30. WP 2126-87, initiated in the field on August 17, 1987 (Shift Engineer and Unit Operator signoff), included a hydrostatic test of the installation performed on September 6, 1987.

No separate notification was made to the Unit Operator or Shift Engineer prior to the test on September 6. This is consistent with MAI-54, Pressure Testing of Piping Systems, Revision 0.

A review of MAI-54 indicates that there is no requirement to notify operators. However, SDSP 17.2, Post Modification Test Program, Section 6.3.h, does require such notification: "Test Directors shall include a prerequiste notifying the appropriate Unit Operator (UO) this test is commencing. [RPT 82-16, LER 259/82-32]."

The inspector noted that the test configuration could have communicated with plant systems, and the piping adjacent to the test volume was not vented.

8) The hydrostatic test of the ECN P0965 valves used FCV 73-30 as a test boundary. Review of MAI-54 (above), section 5.1.6, states, "Check valves, relief valves,control valves or power actuated valves shall not be used as test boundaries without DNE/Plant Engineering concurrence."

WP 2126-87 (field package for ECN P0965) did not receive specific DNE authorization for use of FCV 73-30 as the test boundary. The inspector further noted that SDSP8.4, Section 6.3.1, Preparation and Processing of Work Plan and Inspection Records, does not require the Work Plan Coordinator (WPC) to forward to DNE for review, WP&IRs that utilize approved plant

procedures such as MAIs. This latter practice is in conflict with the requirements of MAI-54, Section 5.1.6.

e. Inspection of Field Activities

Field activities were inspected to determine whether the following objectives were fulfilled:

1) Actual installations were in accordance with the work plans and design output documents. The inspectors visually inspected the installed systems, structures and components and compared the installations to the detailed design drawings, weld data sheets, and work plan provisions.

2) As constructed installation dimensions, assembly/subassembly configurations, wiring terminations, cable/conduit routing, and hanger and support details were verified to be in accordance with the design drawings and work plan requirements.

   Material certifications and traceability was confirmed by verification of heat numbers, fitting and component bills of materials, etc.

   Completion of construction and inspection activities as documented in the work packages were confirmed by direct observation.

3) Radiological controls and ALARA provisions, fire prevention, housekeeping and Quality Control inspection provisions were observed.

4) Current, approved work plans and associated procedures were in use at the point of work. Changes in work plans, procedures, and drawings were processed when construction conditions dictated.

5) Where possible, work in progress was observed for each of the above characteristics.

f. Findings From Field Walkdowns

Except as noted below, the inspection of field activities found acceptable implementation of the work plans and associated requirements.

1) During inspector walkdown of the Core Spray System modifications, the following observations were made:

   a) Although, nuclear piping coating (painting) requirements of WP 2117-86 had been signed off as complete on October 7, 1987, six welds (Nos. 44, 45, 46, 46A, and 46B) were found

unpainted. The licensee issued an immediate revision to the work plan to repaint the affected areas.

b) Piping heat number identification numbers could not be found on three short, 2-inch diameter piping sections installed in the CS Loop I local leak rate testing manifolds (downstream of Valve 75-607) per Drawing 47W2485-8. The licensee demonstrated to the inspector that the affected piping sections were portions of original plant piping retained for use in the manifold and that all modified piping was marked with heat numbers in accordance with the work plan. The inspector was unable to determine whether the original installation should have been marked with its original heat number and will verify this during the next inspection. This item will be tracked as Inspector Followup Item (260/87-42-04).

c) The testing manifold installed on CS Loop II is equipped with a new 3/4 inch test/drain tailpiece extending vertically upward. This tailpiece, containing valves 75-646 and -647 is cut into the mineral insulation of an adjacent 4-6 inch pipeline. The licensee and inspector were unable to identify the larger pipe or the existence of an engineering authorization permitting the insulation to be cut away around the tailpiece.

The inspector observed that the tailpiece appeared to clear the larger pipe (inside the insulation) by only an inch or two and further questioned the licensee regarding the minimum acceptable clearance for hot system operation.

The cognizant modification engineer advised the inspector subsequent to the walkdown that larger insulated line was a building heating pipe, that the clearances and remaining insulation thicknesses were adequate, and that no further action or engineering authorization was necessary. The inspector will review this during the next inspection visit. This item will be tracked as an Inspector Followup Item (260/87-42-05).

d) The two inch CS Loop II fill manifold modified by ECN P0795 includes a manual globe valve (75-611) and a check valve (75-610). The valves were partially covered about 1/16 to 1/8 inch thick with cable tray fire retardant, apparently oversprayed or spilled during treatment of overhead cable trays. The valves were repainted as part of the above modification without removing the fire retardant material. Further, the manufacturer's name plate on valve 75-610 had been mutilated and was attached by only one rivet.

The licensee's preliminary investigation determined that the above condition is undesirable and potentially

deleterious to the valves. Plans were in progress to remove the material and repaint the valves and repair the name plate.

g. Quality Organization Involvement

An inspection was conducted on the licensee's Quality Control (QC) involvement as applied to work plan implementation. The inspectors reviewed the assignment and implementation of QC hold points in work plans and related instructions, completed weld data sheets, and painting/coating data sheets, etc. as part of the documentation and field activity inspections discussed above. QA Surveillance and Audit Reports as listed in Appendix A were reviewed during this inspection period. For the areas inspected no deficiencies were identified. Subsequent inspections will be performed in the area of QA involvement in ECN/modifications program.

h. Organizational Interfaces

Organizational interfaces among design engineering, modification engineering and construction, QC, and craft groups were observed in conjunction with the other inspection activities above. Inspector review of the specific aspects of organizational interfaces will continue and be reported in conjunction with the next inspection.

i. Overall Program Status and Overview

The inspector began assembling program status data and held preliminary discussions with design engineering and modification management. The specific objective of this purpose of the inspection is to independently determine the status of modification completion for those modifications considered by NRC and TVA as prerequisites for Unit 2 restart.

This portion of the inspection will continue and be reported in subsequent inspection reports.

APPENDIX A

REFERENCES

--	Browns Ferry Unit 2 Technical Specifications
BF 3.2	QC Inspection Program, Revision 1
BF 8.3	Plant Modifications, Revision 10
SD SP8.1	Plant Modification/Design Change Approval Revision 5
SD SP8.4	Preparation and Processing of Work Plan and Inspection Records Forms, Revision 6
SD SP8.8	Conversion of Temporary Alteration to Permanent Plant Modification, Revision 0
SD SP8.9	Field Change Requests, Revision 0
SD SP9.8	Site Walkdown Program, Revision 2
SD SP13.3	Implementation of ASME Section XI, Revision 4
SD SP17.2	Post Modification Programs Test, Revision 2
SD SP15.7	Periodic FSAR Updating, Revision 1
MA I-54	Pressure Testing of Piping Systems, Revision 0
BF EP PI 86-03	Preparation and Control of Engineering Change Notice Modification Package, Revision 2
BF EP PI 87-27	Procedure for Origination of Configuration Control Drawings, Revision 0, with Supplements
NE P 6.1	Change Control, Revision 0, with Supplements
NE P 3.2	Design Input, Revision 0, with Supplements
NE P 5.1	Design Output, Revision 0, with Supplements
CA QR 870727	Condition Adverse to Quality Report, Additional Welding on RBCCW Pipe Support after final QC Inspection
LP 4N 45A-C	TVA Internal Audit Plan for FY88 and First Quarter Audit Schedule
BF-A-87-0013	Plant Modifications and Design Control QA Audit Report

QA Surveillance Reports:

QB F-S-87-0454	QA Surveillance Report - Operational Readiness Restart Issues
QB F-S-87-0427	QA Surveillance Report - Electrical Maintenance
QB F-S-87-0244	WP&IR Preparation, Review and Approval
QB F-S-87-0054	WP&IR Work Performance
QB F-S-87-0320	WP&IR Work Performance
QB F-S-87-0321	WP&IR Work Performance
QB F-S-87-0331	WP&IR Work Performance
QB F-S-87-0342	WP&IR Work Performance
QB F-S-87-0369	WP&IR Work Performance
QB F-S-87-0385	WP&IR Work Performance
QB F-S-87-0429	WP&IR Work Performance
QB F-S-87-0441	WP&IR Work Performance
QB F-S-87-0442	WP&IR Work Performance
QB F-S-87-0444	WP&IR Work Performance
QB F-S-87-0444	Prefabrication Workplans
QB F-S-87-0279	WP&IR Work Performance

QB F-S-87-0082	ECNs - Corrective Actions
QB F-S-87-0023	ECNs - Followup on Corrective Actions for Cancelling ECNS and Drawing Discrepancies
QB F-S-87-0013	ECNs for Drawing Discrepancies
QB F-S-87-0081	U2 HP Raw Water Fire Proection Sys. Walkdown

Drawings:
Flow Diagrams Mechanical Diagrams
Mechanical Control Diagrams
Isometric Analysis Diagrams
Wiring Diagrams
Conduit & Cable Schedules
Conduit Routing Diagrams
Pipe Support Details