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

Report Nos.: 50-259/88-21, 50-260/88-21, and 50-296/88-21

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

Docket Nos.: 50-259, 50-260 and 50-296 License Nos.: DPR-33, DPR-52, and DPR-68

Facility Name: Browns Ferry 1, 2, and 3

Inspection at Browns Ferry Site near Athens, Alabama

Inspection Conducted: July 1-31, 1988

Inspector: _____ 10/21/88
D. R. Carpenter, Senior Resident Inspector Date Signed

Accompanying Personnel: C. Brooks, Resident Inspector
E. Christnot, Resident Inspector
W. Bearden, Resident Inspector
A. Johnson, Project Engineer
J. York, Senior Resident Inspector, Bellefonte

Approved by _____ 10/21/88
W. S. Little, Section Chief, Date Signed
Inspection Programs,
TVA Projects Division

SUMMARY

Scope: This routine inspection was in the areas of operational safety, maintenance observations, surveillance testing observations, restart test program, and licensee action on previous inspection findings, Plant Operations Review Committee, reportable occurrences, corrective action program, condenser retubing, and General Electric contractor recommendations

Results: One violation was identified involving failure to perform CAQR generic reviews in a timely manner. One unresolved item was identified concerning the procedures controling keys for access to high radiation (>1000 mrem/hr) areas. Four Inspection Followup Items (IFIs) were identified involving RHRSW corrosion, deficiencies identified during the restart testing of LOP/LOCA C, vaulting of completed and approved test results, and adequacy of identifying and closing out of significant hardware test exceptions.

All of these issues are to be resolved prior to restart.

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REPORT DETAILS

1. Persons Contacted

Licensee Employees

*J. Walker, Plant Manager
P. Spiedel, Project Engineer
J. Martin, Assistant to the Plant Manager
*R. McKeon, Operations Superintendent
T. Ziegler, Superintendent - Maintenance
*D. Mims, Manager - Technical Services Supervisor
J. Turner, Manager - Site Quality Assurance
M. May, Manager - Site Licensing
*J. Savage, Compliance Supervisor
A. Sorrell, Site Radiological Control Superintendent
R. Tuttle, Site Security Manager
L. Retzer, Fire Protection Supervisor
H. Kuhnert, Office of Nuclear Power, Site Representative
T. Valenzano, Director - Restart Operations Center
*C. McFall, Compliance Engineer

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

*NRC Attendees

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

*Attended exit interview.

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

2. Operational Safety (71707, 71710)

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

The NRC inspectors made routine visits to the control rooms. 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 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.

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

The NRC inspector toured the residual heat removal service water (RHRSW) pump building on July 12, 1988, and found the condition unacceptable. The B3 pump which was in service at the time exhibited gross shaft seal leakage. This condition had been in existence since at least May 21, 1988, and was documented on MR No. 869190. Another deficient condition which was documented on a tag hanging on the pump, MR No. 865395, since May 23, 1988, was excessive vibration. The apparent reason that this pump was in service was that of the 12 RHRSW pumps, seven pumps were out-of-service for various reasons. General corrosion was evident on all piping and components with the most severe being the valve bonnets for the A1, B1, and B2 pump discharge valves. The licensee was asked to determine the minimum acceptable wall thickness for these valves and establish the acceptability of both the current wall thickness and projected end of life wall thickness. This will be tracked as an Inspector Followup Item (IFI) (259,260,296/88-21-03) RHRSW Corrosion. Licensee representatives were cautioned not to allow the flexibility and redundancy available with this system, particularly with only one unit to be placed in operation, to translate into a lack of aggressiveness in system maintenance.

On July 15, 1988, the NRC inspector observed the controls established over high radiation areas which exceed 1,000 mrem/hr per BFN Technical Specification (TS) 6.8.3.2. Fuel reconstitution activities released activated corrosion products which were deposited in fuel pool cooling (FPC) system components. As a result, a high radiation area greater than 1,000 mrem/hr was created around the Unit 1 FPC heat exchangers. Access to the area was not secured by locks with the keys under control of the Shift Engineer. This condition is allowed by TS for a period of up to 30 days provided that the area is controlled by direct surveillance to prevent unauthorized entry. The inspector interviewed the high radiation area watch and confirmed that he was knowledgeable of his duties and responsibilities. The inspector observed that the area was properly posted and confirmed by review of the surveys and by independent measurement that boundaries were properly established. However, the condition was not properly annotated on the survey maps posted at the Radiation Worker Information Boards located at the entrance to the Radiologically

Controlled Area (RCA), nor was it listed as a "Miscellaneous Problem" or "Unusual Condition" on the Shift Operating Supervisor's (SOS) Status Board. The inspector questioned whether SOS permission was required for entrance into the area since no key control was currently available. Licensee representatives responded that SOS permission was not required but would review the issue. The intent of the TS is that management, via the Shift Engineer, be made aware of and exert positive control over each individual entrance into a greater than 1,000 mrem/hr high radiation area.

The NRC inspector also assessed the licensee's routine program for control over locked high radiation areas. The program is described in Browns Ferry Standard Practice BF-19.26, Key Control and Accountability; Radiological Controls Instruction RCI-17, High Radiation Area Door Control; and Operations Section Instruction Letter OSIL-16, Keys. The instructions were found to be contradictory and confusing. OSIL-16, Section 3.1 requires high radiation area door keys to be "under the strict administrative control of the Shift Operations Supervisor" (SOS) (new designation for the Shift Engineer); however, in Section 4.1 it also allows high radiation area keys to be assigned to each control room, assistant shift operations supervisor, RadCon, and Nuclear Security Services with the SOS maintaining "accountability for them". Section 4.2 of this OSIL states that high radiation area keys may be used by "Operations, Operations Training, and NRC Resident Inspection personnel only". No allowance is made for RadCon or Nuclear Security use of their assigned keys. RCI-17 requires in Step 6.3 that authorization to unlock high radiation area doors will be obtained from the SOS, but in Step 6.4 it indicates that RadCon has pre-authorized use of their keys and need only inform the Unit Reactor Operator and not the Shift Engineer.

The NRC inspector ascertained through interviews with RadCon personnel and a SOS that high radiation area door keys were maintained by RadCon and that the SOS permission was not sought or obtained for all entries into the locked high radiation areas. The only control exercised by the SOS over the high radiation area door keys was a once per shift acknowledgement that the SOS clerk had performed a survey of all the keys and all keys were accounted for. The inspector is concerned that the procedures that control the keys controlling access to high radiation areas (¶1000 mrem/hr) are confusing and contradictory and this is identified as an unresolved item pending clarification of this issue (259, 260, 296/88-21-01).

3. Surveillance Observation (61726)

On July 12, 1988, during the performance of Surveillance Instruction (SI) 2-SI-4.9.A.2.a-2, Weekly Check for Shutdown Board C and D Batteries the licensee declared the shutdown boards inoperable. The battery electrolyte temperature was found to be in excess of the 90 F acceptance criteria on both the C and D shutdown board batteries. Later that same day, the Units 1 and 2 A, B, and C diesel generators were declared inoperable for the same reason during the performance of 0-SI-4.9.A.2.a, Weekly Check for Diesel Generator Batteries. The licensee initiated CAQR No. 880470 in order to document proper resolution of the condition. The source of the acceptance criteria on battery electrolyte temperature was the vendor

manual which included a normal operating temperature range of 60-90 F. The only adverse affect of higher temperatures is a 50 % reduction in expected life for each 18 F above 77 F in the event of continued exposure to the elevated temperature. The only limitation contained in the vendor manual with regard to temperature is an absolute limit of 170 F during battery charging. The license initiated changes to all appropriate SIs to raise the upper temperature limit accordingly. The NRC inspector followed the licensee's activities and confirmed the appropriateness of the corrective action through a review of the vendor documentation.

No deviations or violations were identified.

4. Plant Operations Review Committee (40700)

On July 14, 1988, the Plant Operations Review Committee (PORC) conducted a meeting by telephone conference call at 8:00 pm. As requested, the NRC inspector was notified and monitored the meeting by phone. TS 6.5.1.4 authorizes PORC business to be conducted by phone for expedited meetings when it is not practical to convene as a group. Several PORC members questioned the necessity of the telephone meeting given the issues on the agenda. Two CAQR's requiring PORC approval in order to release non-conforming material for testing and installation from the warehouse were discussed (BFN 880476, BFN 880481). The telephone conference was deemed necessary in order to meet schedules established for systems return to service to support the targeted fuel load date. In both cases, the deficiency which prevented release of the material was the lack of seismic qualification documentation. The CAQR's clearly documented a prohibition on considering the systems in which the components were to be installed as operable unless and until the seismic documentation was obtained and approved by design engineers. Part B of the CAQR's, however, took a contradictory position to this. The "No" block was checked on both CAQR's in answer to the question of whether the CAQR impacted unit operability. This was a point of discussion among several PORC members, but all subsequently approved the CAQR's as written. Several NRC observations as a result of this meeting were discussed with the Plant Manager and members of his staff during a routine weekly meeting. Among these concerns were, 1) Telephone conference PORC meetings should not be conducted as a matter of convenience but should be reserved for events, incidents or conditions having true safety significance; and 2) The apparent inconsistency of not checking the block marked "Yes" in response to the question of impact on operability. These observations were considered by the inspector to be isolated occurrences.

No deviations or violations were identified.

5. Maintenance Observation (62703)

During a control room observation on July 24, 1988, the inspector noted an out-of-service amber indicating light for the 4KV shutdown bus no. 2 auto-transfer lockout relay 43-2. The inspector tracked the deficiency to

MR No. 811066 dated December, 1985. The current status was reported as being in the Electrical Technical organization for engineering evaluation (since March 1988). Upon interviewing the responsible personnel, the inspector learned that the MR was being turned over to operations for post-maintenance testing and closeout following a visual inspection which found no problem. The NRC inspector accompanied the cognizant engineer on an inspection of the light and noted that the wrong light had been tagged as being deficient. The engineer had been troubleshooting the light for the 4KV shutdown board A transfer switch 43SA. The orange MR sticker with the MR number was erroneously applied to an adjacent light. The deficient condition went uncorrected for 2 1/2 years due to erroneous positioning of an MR sticker and poor trouble shooting of a properly functioning component. This is considered to be an isolated breakdown of the MR process that occurred 2 1/2 years ago. Recent improvements in the licensee's programs should preclude issues such as this in the future.

No deviations or violations were identified.

6. Reportable Occurrences (90712, 92700)

The below listed licensee event 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.

(Closed) LER No. 296/83-04 Rev. 1, Residual Heat Exchanger Tube Leak. A leaking tube was found in the RHR 3D heat exchanger. Metallurgical examination revealed a circumferentially oriented crack in an area of the tube where mechanical damage had occurred. Metallographic examinations did not reveal any evidence of corrosion assistance to the failure. Eddy current testing was performed on 380 tubes of which 12 tubes were found to be mechanically damaged (dented). All 12 tubes including the leaking tube were plugged. The NRC inspector reviewed the completed work plans.

(Closed) LER No. 259/84-08 Rev. 1, Reported Failures of the Unit 1 High Main Steam Line Flow Differential Pressure Transmitters (DPT) 1-25 A through D. The failures occurred during January and February of 1984. Following testing of the transmitter, the licensee concluded that the failure was due to the behavior of the pulse dampening devices (snubbers) installed in the instrument sensing lines. The snubbers were removed and Unit 1 was operated for approximately seven months with no further problems noted. The snubber removal occurred under Temporary Alteration Change Form (TACF) 1-84-079-1. The temporary alteration was made permanent by ECN P0126 as implemented under WP 10370. The NRC inspector reviewed the completed TACF, ECN and work plan.

(Closed) LER No. 259/84-15 Rev. 1, Removal of Under Designed Vacuum Priming Valve. During a design review of the EECW system it was discovered that the vacuum priming valve on the north header (none on the south header) was under designed for the system pressure. This valve had been isolated from the system for two and a half years and testing determined that the valve's function was unnecessary. The NRC inspectors reviewed Work Plan and Inspection Report No. 3026-87 and performed a field inspection to ascertain that the valve had been removed and the piping capped.

(Open) LER No. 296/85-17 Rev. 1, Failed Supports On the Residual Heat Removal System. Three hangers located on loop 1 of the RHR system of Unit 3 failed due to high vibration caused by the throttling action (resulting in cavitation) of an injection valve during shutdown cooling mode operation. The licensee decided to reduce the vibration to an acceptable level by replacing the original valve disc with a fluted disc. This has been completed for Unit 2 and vibrational measurements have been made on the Unit 2 torus to torus portion of the RHR system and found acceptable. However, the vibrational measurements have not been made on a part of the line that is used during the normal reactor cooling mode. The vendor does not recommend running water into the reactor vessel when the fuel is not present and the head is not on the vessel because of potential damage to some of the reactor internals. The vibrational tests for the remaining portion of the RHR piping will be made during post modification test (PMT) No. 139 and will be required prior to Unit 2 criticality.

(Closed) LER No. 296/86-07, Loop I of RHR System Inoperable After Two Damaged Hangers Discovered. Hanger H3 had a 4-inch crack at a structural tubing weld and hanger H8 had a support lug broken off from the ceiling. Both hangers are located on the 18-inch diameter RHR injection line above the torus. The high vibration of this piping is caused by the throttling action of loop I RHR injection valve 3-FCV-74-52 that occurs during the shut down cooling mode. Metallurgical examination revealed that the parts had failed due to fatigue. The NRC inspectors performed a visual inspection of the two repaired hangers. In order to reduce the magnitude of the vibration, the injection valves were to be modified by replacing the present valve discs with fluted discs. The modification of the valves has been completed on Units 1 and 2, but not on Unit 3. The modification of the valve on Unit 3 will be followed under LER No. 296/85-17.

(Closed) LER No. 296/86-09, Damaged RHR Hanger Causes Prohibited Operability Configuration. Hanger H10 (loop I) located on the 24-inch RHR shutdown cooling and low pressure coolant injection line above the torus, had a crack in a load bearing tube steel member. Metallurgical analysis of the crack showed a fatigue mode of failure. As noted in LER No. 296/86-07, this piping on Unit 3 (loop 1) has considerable vibration caused by the throttling action of injection valve 3-FCV 74-52. The NRC inspectors performed a visual inspection of the reinstalled hanger H10. The modification of the valve on Unit 3 will be followed under LER No. 296/85-17.

(Closed) LER No.296/87-02, Unplanned Reactor Water Cleanup Isolation During Testing Due to Fuse Failure. When channel A1 was de-energized during a functional test, there was an unexpected isolation of the reactor water cleanup system. This happened because channel B2 was de-energized due to a blown fuse. The functional test procedure was revised to add steps which verify that the trip relays are energized at the start of the test. The NRC inspector reviewed the revised functional test procedure and verified that it requires verification that the trip relays are energized.

(Closed) LER No. 259/87-08 and Rev. 1, Failure of Potential Transformer Fuse Contact Cause Electrical Fault and Engineering Safety Feature Actuation. During the performance of a monthly surveillance test a phase to phase short occurred between contacts in the diesel generator control cabinet for the 3ED diesel generator (DG). The fault caused a refueling zone isolation, initiation of standby gas treatment and control room emergency ventilation, and on Unit 3, a half scram and primary containment isolations. The cause of the fault was a failure of the potential transformer fuse contacts. The fuse and spring finger contact were bypassed on all eight DGs after an engineering evaluation determined that the fuse in the DG exciter potential transformer circuitry was unnecessary. The NRC inspector reviewed the engineering evaluation and the completed work plans that bypassed fuse and spring finger contacts for all eight DGs.

(Closed) LER No. 259/87-22 and Rev. 1, Engineered Safety Feature Actuation Due to Personnel Error During Switch Calibration. During calibration of a raw cooling water pressure switch, two emergency equipment cooling water pumps were inadvertently started due to a personnel error. The calibration procedure was revised to provide an improved method of isolating the switch during calibration. The instrument mechanics involved were counseled on the need for increased caution when working with energized equipment. The NRC inspector reviewed the revised calibration procedure and the documented counseling of the instrument mechanics.

(Closed) LER No.259/88-03, Inadequate Procedure Causes Inadvertent Start of Emergency Equipment Cooling Water Pumps. During an attempt to put a raw cooling water (RCW) pump into service and the taking out of service another RCW pump, the RCW header pressure dropped below the low pressure setpoint. The operating instructions for the RCW system were revised to provide instructions for alternating pumps in and out of service. A review of this event will be provided to current operations personnel. The NRC inspector reviewed the revised instruction and the event description provided to the operations staff.

No deviations or violations were identified.

7. Restart Test Program (RTP)

The inspector attended RTP status meetings, reviewed RTP test procedures, observed RTP tests and associated test performances, reviewed RTP test

results (including test exceptions), and attended selected Restart Operations Center (War Room) and Joint Test Group (JTG) meetings. The following are the RTP activities and associated activities monitored, and the status of testing during this reporting period:

a. Restart Tests Performances (RTP)

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

RTP-03 A and B, Reactor Feedwater

RTP-023, Residual Heat Removal Service Water

RTP-030, Diesel Generator and Reactor Building Ventilation

RTP-031 A and B, Control Building Heating Ventilation and Air Conditioning

RTP-077, Turbine-generator/Electro Hydraulic Control

RTP-067, Emergency Equipment Cooling Water

RTP-064, Primary Containment Isolation

RTP-070, Reactor Building Closed Cooling Water

RTP-073, High Pressure Core Injection

RTP-074, Residual Heat Removal System

RTP-082, Diesel Generators

RTP-084, Containment Atmosphere Dilution

RTP-085, Control Rod Drive

RTP-092, Neutron Monitoring (SRM, IRM, LPRM, APRM)

RTP-099, Reactor Protection System

RTP-057-1, 120 V DC Diesel Generator Batteries

RTP-057-3, 250 V DC Unit Battery

RTP-057-5, 4 KV Distribution

RTP-BVC, Backup Control

RTP-L/L C, LOP/LOCA "C" Rev. 2

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

b. Loss of Power/Loss of Coolant Accident (LOP/LOCA) Testing

During the LOP/LOCA series of tests a significant test exception was identified involving the Unit 3 diesel generators output breakers 3A, 3C and 3D. When performing LOP/LOCA "C", these breakers, which supply standby power to shutdown boards 3EA, 3EC and 3ED respectively, locked out, i.e. would not close onto their shutdown boards, when required by the LOCA signal. The licensee modified the breaker control logic circuits (See paragraph 8 of this report for details). Also during the LOP/LOCA "C" test, a major electrical switchboard (480V Reactor MOV Bd 2A) was not aligned as called for in the test prerequisites. The licensee initiated CAQR 88-0399 to document this misalignment. These two items prompted the licensee to perform a LOP/LOCA "C" re-test referred to as LOP/LOCA "C" Revision 2. This test called for turning off the incoming plant power using the plant unit startup switchboard breakers, which are located in the plant, instead of the three main power feeds coming from the switchyards. NRC Inspectors monitored this retest from various plant locations. To perform the procedure reviews, test witnessing, and post test evaluations, the inspectors used the NRC 2513 program inspection modules as guidance. No problems with the test procedures were identified.

The retest was mainly for those items that did not test satisfactorily during the original LOP/LOCA "C" Test. However, a new addition to the test was added and referred to as Section 5.5, Diesel Generator/Paralleling System. This new test was to verify a design feature that allowed for the paralleling of all eight (8) DGs with offsite power while the LOCA signal was still present. The following NRC inspector observations were made and discussed with the licensee:

(1) Two inspectors were stationed in the Unit 1 and 2 control rooms and made the following observations:

- RHR Pump 2A breaker failed to close

- The following loads could not be verified due to their breakers being racked out: 480 V Shutdown Boards 1A and 1B; 250 V Battery Charger 1; Fuel Pool Cooling Pump 1A; Drywell Blower 1A; LPCI MG Set IDN; and LPCI MG Set IEN.

- Low Pressure Coolant Injection (LPCI) MG Set 1EA should have remained energized; it's breaker was found opened.

- Control bay water chiller "B" should have load shed; it's breaker was found closed.

- Step 8.3.13 of OI-82 (restoring off-site power to the 4KV Shutdown Boards) could not be executed. The auto transfer lockout relay could not be reset (apparently due to the accident signal).

- I & C Breaker 204 inadvertently tripped during the test.

- A blown fuse in the control power prevented operation of the alternate feeder to 4KV Unit Board 2A.

- A lack of communications and proper operational coordination was observed while conducting the diesel generator paralleling operation. A major load was re-energized (480 Shutdown Board 2B) which produced a voltage transient while attempting to establish proper synchronization. This came as a complete surprise to the operator who was attempting to parallel the Unit 1 and 2 DGs.

- Improvements were noted in control room communications and control. The unit operator announced the receipt of alarms, provided reports on completed actions, established professional face-to-face communications through "repeat-backs", and verified expected indications following switch manipulations. Additionally, a prompt review of loads lost following an inadvertent trip of an I & C breaker was conducted. Although not everyone practiced these techniques, the Unit 2 operators established and projected a close team-work relationship using these methods.

- The amber indicating light on panel 9-23-8 for the 4KV Shutdown Bus 2 auto transfer lockout relay 43-2 was out of service with a December 1985 MR No. 811066 attached. This delay is considered to be unacceptable. Refer to paragraph 5 for additional information on this subject.

(2) An NRC inspector was assigned to the Unit 3 control room and made the following observation:

- Operations personnel were knowledgeable of testing requirements and plant conditions, and were attentive to their duties. The Unit 3 diesel generators were paralleled to offsite power in accordance with step 6.2 of BF 3-OI-82, Standby Diesel Generator System Operating Instructions; and step 5.5 of 2-BFN-RTP-L/L-C, LOP/LOCA testing. No discrepancies were noted.

(3) An NRC inspector was initially assigned to the Unit 2 Auxiliary Instrument Room and made the following observations:

- In order to indicate isolation signals had been transmitted to Unit 2, the inspector expected to observe that 5 HFA relays de-energized; however, only one HFA Relay (16A-K26) was de-energized. At the start of the test, the inspector was told that a test exception would be written.

- Equipment on the 4-KV Shutdown Board B performed as required - No discrepancies were identified.

- Equipment on the 4-KV Shutdown Board D performed as required. 480 V Diesel Auxiliary Board B transformer TDB ACB (compartment 13) was supposed to remain closed; however, only the yellow light was "on" indicating an open condition.

- The licensee performed these tests in a knowledgable and professional manner QA inspectors were noted at each location that the NRC inspector monitored.

(4) An NRC inspector was assigned to the 3EC shutdown board feeder breaker from the 3C DG. This breaker had malfunctioned during the LOP/LOCA "A" and "C" tests. The inspector made the following observations:

- The normal feeder breaker opened upon the initiation of the LOP signal; approximately six seconds later the the standby feeder breaker from 3C DG closed and then immediately opened; the charging motor recharged the breaker springs; and, unlike what occurred during the original LOP/LOCA "C" test, the breaker closed and remained closed. The inspector then observed the newly installed time delay relay and it indicated that it had activated, timed out and closed it's logic.

- No deficiencies were identified at the 3ED shutdown boards; diesel generator auxiliary boards 3A, 3B, A and B; and shutdown board B.

- At the Unit 3 control room, the paralleling of the Unit 3 DGs with offsite power while the LOCA simulated signal was still present were observed. No deficiencies were identified.

The failure of the RHR pump 2A breaker to close is identified as IFI 259,260,296/88-21-04, pending determination of why it failed and any necessary corrective action.

At the beginning of the LOP/LOCA test series, several site departments appeared to not give these tests the type of attention

needed; however, by the end of the testing series this attitude was turned around and the final test was conducted with the type of professionalism expected at a nuclear power facility.

c. Specific Test Witnessing - Reactor Protective System

The inspector reviewed 2-BFN-RTP-099, Reactor Protective System (RPS), and observed portions of the scram testing involving various scrams such as high reactor vessel pressure, high main steam line radiation, and low reactor vessel level. The Browns Ferry RPS consists of two trains, A and B, with each train sub-divided into two channels A_1, A_2 and B_1, B_2. Each channel has the same number of scram signals. This arrangement allows for a half scram logic if any one of the channels activate, and it takes a logic of one out of two trips occurring twice (one out of two, twice) to initiate a full scram. The overall intent of the test is to calibrate each scram parameter such as high reactor vessel pressure and low reactor level; check the functions of the reactor mode switch, such as refuel and startup; verify the scram logic from each scram signal; verify that the reactor can be scrammed remotely by turning off the power supply electrical breaker to the RPS Motor-generator sets (a total of two); and conduct time response testing for each channel.

The NRC inspector observed the performance of Section 5.5 of the test, Low Reactor Water Level. This section was verified using SIs 2-SI-A,B,C and D (one for each channel). The SIs required that a calibration device be hooked up to the Rosemount level transmitters located in the reactor building and that a calibration of the transmitters be performed. The SI also required that the Rosemount Analog Trip Units (ATU) located in the Unit 2 auxiliary instrument room be checked. The RPS testing will continue until each scram signal has been verified (a total of 39 signals). It should be noted that the inspector utilized NRC 2513 inspection program modules as guidance in performing these inspection activities. No deficiencies were observed.

d. Test Results Review

The NRC inspector reviewed the results of 2-BFN-RTP-[illegible], Standby Gas Treatment System, which were reviewed by the Joint Test Group and approved by the Plant Manager on July 6, 1988. This test was reviewed by the inspector from the Baseline Test Requirements Document (BTRD) to the final results. Actual field observations were conducted and were documented in previous resident inspection reports. Throughout the testing activities the inspector used NRC 2513 Inspection Program Modules as guidance in performing the inspection activities.

It was noted by the inspector that a total of twenty-one (21) test exceptions (TE) were documented by the Test Director, with five outstanding when the test results were approved by the Plant Manager. The following outstanding TEs were reviewed in depth:

(1) TE-7 involved the zonal dampers located in stairwells and elevator shafts separating the three unit reactor building zones from the refuel zone. The test director initiated MR No. 779834 to repair these dampers. However, the FSAR indicates that credit is not taken for these zonal dampers because credit is taken for the entire secondary containment, i.e, the three reactor building zones plus the refueling zone. The BTRD was revised to reflect this.

(2) TE-8 involved the four (two per train) dampers located in the equipment access area located between Units 1 and 2. The test requirement called for the dampers to close in ten (10) seconds, however, all four dampers closed in approximately twenty-eight (28) seconds. The test director initiated CAQR-880177 to document this deficiency.

(3) TE-11 involved the attempt to document stack effect by blowing smoke into the duct work located in the stack. This item was discussed in a previous resident inspection report. The test director initiated CAQR-880304 because of possible seismic considerations with an unmonitored ground release.

(4) TE-20 involved a hold order on both A and B DG auxiliary electrical boards, breakers numbered 5C, which supply power to the dilution fans.

(5) TE-21 involved an NRC Violation (259,260,296/88-05) which documented the performance of iodine testing of the filters.

The inspector noted, on July 26, 1988 (20 days after final review and approval), that the approved test results were not turned over to the QA vault in a timely manner. This item is identified as IFI 259,260,296/88-21-05, Vaulting of Completed and Approved Test Results, pending the development of more timely administrative controls.

e. Test Exceptions

The inspector continued to followup the licensee's handling of the TEs identified by the RTP. As of mid July 1988, four hundred forty four (444) TE's were identified, of which one hundred and one (101) were still outstanding. The restart testing group has reviewed the TEs and have categorized them into six areas as follows.

(1) Equipment Deficiencies, which is subdivided into equipment malfunction (1.1) and equipment performance (1.2).

(2) Procedural difficulties, which is subdivided into procedure errors/editorial (2.1); procedure method/performance (2.2); plant condition/equipment availability (2.3); and prerequisite/initial conditions (2.4).

(3) Personnel Errors, which is subdivided into test director errors (3.1) and support personnel errors (3.2).

(4) Partial Release (4.0), which is used when the JTG releases a particular section of a test for performance.

(5) Calibration Deficiencies, which is subdivided into measuring and test equipment (5.1) and process instruments (5.2).

(6) Other

The inspector reviewed selected TEs from Test Procedures 052-4, 057-5, 065, and 082 against this categorization. It was noted that of the four hundred forty four TEs, ninety-two (92) involved equipment malfunction (category 1.1) and forty nine (49) involved equipment performance (category 1.2). The inspector observed that maintenance requests (MRs), procedure changes (intent and non-intent), and CAQRs were used to document and close out TEs. The inspector expressed concern that these MRs, procedure changes and CAQRs be given appropriate consideration in the final review and approval of each test results package. The inspector will verify this in his review of the approved test results packages. This is identified as IFI 259,260,296/88-21-06. Adequacy of Closing Out of Significant Hardware Test Exceptions (Categories 1.1 and 1.2.).

No deviations or violations were identified.

8. Modifications (37700)

The NRC inspector reviewed CAQR 880394 which documented the locking out of 3EA, 3EC and 3ED DG breakers during the initial LOP/LOCA C test and subsequently generated a design change request. This change involved the installation of time delay relays manufactured by ASEA Brown Boveri in each of the eight (8) shutdown boards in order to allow the DG output breakers to close on to their respective shutdown boards during a loss of power followed by a loss of coolant accident. The inspector also reviewed the design change implementation which was accomplished by Work Package No. 0095-88, Engineering Change Notice (ECN) E-0-P7150, Test Scoping Document for test no. PMT-195, and Post Modification Test 195 (PMT-195). The inspector observed the installation process involving shutdown boards C, B, and 3EC. The licensee did not install the time delay relay into shutdown board D, because that board would be deliberately disabled for the LOP/LOCA retest. The inspector observed the presence of QC inspectors monitoring the installation of the time delay relays. The review of the test scoping document and the PMT-195 indicated that the delay was to be set for 2.5 + or - 1.35 seconds, because the time to recharge the closing springs of the breakers was to be two (2) seconds or less. However, the closing spring recharging time on some of the breakers was greater than two (2) seconds. The Post Modification Testing Group initiated a CAQR which resulted in changing the time delay to three (3) seconds. During the LOP/LOCA C retest the inspector noted that the modification worked as designed. The inspector will follow up on the installation of this design on shutdown board D.

No deviations or violations were identified.

9. Generic Applicability of Conditions Adverse to Quality

The NRC inspector selected various CAQRs for review and identified four CAQRs that had not received a Browns Ferry generic review within the time frame required by the NQAM Part I Section 2.16, paragraph 10.5 which requires that potentially affected organizations complete a generic review within 70 calendar days from the origination of the CAQR. Specifically the inspector noted the following generic reviews that were performed late:

CAQR	Due	Performed
SQN 871347	11/1/87	2/9/88
SQP 871003	8/10/87	2/24/88
SQP 871066	7/15/87	4/5/88
SQT 871347	10/15/87	2/23/88

The inspector determined that the problem with late generic reviews has existed for some time. The inspector reviewed various Site Quality Manager memos which identified many overdue generic review items. The memos identifying overdue generic reviews have been issued on a routine basis. The inspector was informed by the Site CAQR Coordinator that the number of outstanding late generic review items has decreased from 137 during January 1988, to 24 during July 1988, and that NQAM Part I, Section 2.16 is in the process of being revised to re-structure the time limits to a more workable structure. The NQAM will also require escalation to higher management based on late generic review items. SDSP 3.7 will be revised to reflect the new NQAM revision within 90 days of NQAM issue.

The NRC inspector believes the problem of late generic reviews has received insufficient management attention. This has occurred inspite of the fact that licensee QA personnel have documented the problem in 2 different CAQRs during 1987 (BFP 87 0830 and BFP 87 0326). Additionally, separate violations in this area are documented in NRC Inspection Reports 86-43 and 87-41. Although the late generic reviews were identified by the licensee, timely and effective corrective action has not resulted. This constitutes an apparent violation for failure to follow procedure, Violation 259, 260, 296/88-21-02, Failure to Perform CAQR Generic Reviews.

10. Condenser Retubing

During July 18-19, 1988, an NRC inspector observed ongoing work associated with DCN M0075A, Unit 2 Main Condensor replacement of Admirality Brass tubes with Allegheny Ludlum AL-6XN stainless tubes. The observed work was being accomplished in accordance with WP 2165-88. The inspector reviewed WP 2165-88 and noted that it included a special installation instruction which stated various requirements intended to control the tubing installation work and prevent damage to the new tubes prior to installation. The inspector noted various poor work practices which were not in

accordance with the special instruction included in the work plan. Specifically the inspector observed the following:

Unsupported sections of new tubing lengths of 25 to 30 feet during installation and storage at the work location. Special instruction required no more than 10 feet of unsupported span.

Failure to clean the full length of new tubes with freon during the actual installation process.

Workers were observed walking on new tubes located in the tube storage rack.

Sections of heavy wood lumber used as temporary walkways were drug across new tubes located in the tube storage rack.

The inspector discussed the observed poor work practices with licensee management and the licensee agreed to investigate the event and take corrective action.

The licensee responded by way of a Modifications Manager memo dated July 26, 1988 (R06 88 0725 876). As corrective action the licensee agreed to revise the workplan, counsel the craft involved with the job, and take additional measures to protect the new tubes during the ongoing work.

Although the condenser retubing work is not associated with CSSC components, there exists a significant concern that similar poor work practices could exist on other jobs associated with safety-related systems or components.

No deviations or violations were identified.

11. Licensee Action on Previous Enforcement Matters (92702)

(OPEN) Violation 259,260,296/88-04-02, Procedures improperly encouraged use of temporary "scratchpads" before making formal operating log entries. The licensee admitted that this practice did not comply with the Nuclear Quality Assurance Manual and recently revised Standard Practice 12.24, Conduct of Operations, to eliminate this practice. Currently, the procedure requires log entries to be made directly into the official record at the time of the event. The inspector observed several operators over the course of this inspection period and in all except two instances they were found to be in compliance with the new guidelines. In these two instances occurrences were not recorded in the log, even though significant time had elapsed since they had occurred, and operators were still using temporary "scratchpads." The operators did not appear to be involved in activities that would hinder them from making prompt log entries. These cases were reported to the Operations Superintendent and are considered to be examples of the continuing violation. This item will remain open pending furtner corrective action by the licensee and additional sampling for compliance by all operators.

(CLOSED) Violation 259,260,296/84-34-04, Failure to adhere to procedures and inadequate procedures which contributed to an incident involving overpressurization of the Core Spray System on August 14, 1984. One example of this violation involved an SI step which erroneously designated MOV Board 2A or 2B vice the correct MOV Board 1A or 1B. This SI has been corrected. The second example involved failure on the part of an operator to properly open a circuit breaker as required by the SI. In response to this violation, the licensee conducted operator training on proper breaker manipulation, clarified the step in the procedure, and added second party verification steps throughout the instruction in order to assure that similar operator errors would be minimized. Further, the licensee initiated a change to the TS which would eliminate the need to perform these types of tests at power and allow testing to be performed during outages. Thus the potential for a recurrence of this type of event and possible interfacing system loss of coolant accident is diminished. The NRC inspector reviewed the associated procedure changes and the TS change. The TS change was approved by the NRC on February 12, 1988. This violation is considered closed.

(Closed) Deviation 259,260,296/87-02-01, Curbs and floor drains in battery rooms and battery board rooms. This item concerned Section 10.11 of the FSAR requiring that each battery room and battery board room contain drains and curbs. A tour by a NRC inspector was made of the battery and battery board rooms for all three units to ascertain that the curbs had been installed. It was also noted that (a) Unit 2 had floor drains in both rooms, (b) Units 1 and 3 had floor drains in the battery rooms, and (c) Units 1 and 3 have a 2 inch x 6 inch opening in the wall between the battery room and the battery board room to allow the water to flow from the battery board room to the battery room floor drain. This deviation in considered closed.

12. Followup of Open Inspection Items (92701)

(Closed) Inspector Followup Item 296/83-19-03, Shutdown board room conduit damage. During a tour of the Unit 3 shutdown board rooms, the inspector noted that several electrical conduits extending through the 3ED room south wall to the outside area were displaced 2 inches lower from the position as indicated on the applicable TVA drawing 45N888-12RA.

The inspector reviewed the TVA drawing discrepancy report associated with this item. The licensee evaluated the condition and determined the installation as existing as acceptable and that the physical location of the conduit was incorrectly shown on the original drawing. The inspector reviewed the revised drawing modified to comply with drawing discrepancy package No. 3-86-0772 which shows the correct conduit locations. This item is closed.

(Closed) Inspector Followup Item 260/84-41-04, Relocation of HPCI EGM control boxes. The licensee had identified the need to relocate the HPCI EGM control boxes (Panel 25-49 Turbine Control Panel) due to the harsh environment of high temperature and high humidity in which the control box

was located. This item was opened to track corrective action until a design change request (DCR) could be approved and the EGM control box relocated.

The inspector reviewed ECN P3184, associated with DCR 2349, which called for replacement and/or relocation of various components of the HPCI system and determined that the ECN was field complete and that the EGM control box for the Unit 2 HPCI system had been relocated to a new location away form the HPCI turbine skid. ECN P3184 had been written to upgrade the environmental qualification of the HPCI System for Unit 2. The inspector concludes that adequate licensee corrective action has occurred to resolve the original concerns as identified in the original inspection report for Unit 2 only. This item is closed for Unit 2 but will remain open for Units 1 and 3 pending future review of licensee actions.

(Closed) Unresolved Item 259,260,296/85-15-01, NDT curve out of date; no surveillance required, and TS discrepancy between units. The inspector had identified three concerns during a review of Unit 1 TS. The concerns identified were:

- Out of date Figure 3.6-1 contained in TS

- No documented verification contained in GOI-100-1, Integrated Plant Operations, that reactor vessel shell temperatures were at or above the temperature of curve no. 3 of Figure 3.6-1 as required by TS 3.6.A.2

- Inconsistency between Unit 1 and Unit 2 TS and Unit 3 TS for TS 3.6.A.1.

The inspector reviewed the documentation associated with the licensee's reply to the identified concerns. The inspector noted that the following corrective action had occurred:

- TS for all units have been revised to remove the inconsistency identified above and to update Figure 3.6-1 for use until 12 EFPY (effective full power years) of irradiation

- GOI-100-1 has been revised to require that vessel shell and primary water temperatures are greater than 180 F on all working indicators.

The inspector feels that the licensee's corrective actions are sufficient to address the concerns as identified in the original inspection report. This item is closed.

(OPEN) Unresolved Item 259,260,296/86-06-08, Inadequate slope on instrument sensing lines. This item documented a failure to comply with the targeted slope of 1 inch per foot during installation of Workplan No. 2040-35. Although 1 inch per foot downward slope is the goal, Construction Specification G-60 allows as little as 1/8 inch per foot slope when it is impractical to achieve the desired 1 inch per foot. The NRC

inspector accompanied the modification engineer on his final walkdown inspection and witnessed selected slope measurements in the field. Revision 14 to the Workplan added a QC holdpoint to verify slopes on all sensing lines. Revisions 21, 23, and 25 to the Workplan documented rework necessary to meet the sloping requirement. Only one line failed to meet the minimum slope criteria of Specifications G-60. Field change request (FCR) 86-178 documented zero slope on line B-5 shown on drawing 47W600-58. Approval of the FCR on April 16, 1986 accepted the zero slope.

The instrument line having zero slope serves three instruments; PT-64-160A High Range Drywell Pressure, PT-64-57B High Drywell Pressure, and PT-64-58B High Drywell Pressure. The NRC inspector located the instrument line on July 20, 1988, and found that it was not impossible to re-route the line and achieve some downward slope. No further engineering analysis or safety evaluation was performed by the licensee to justify acceptance of this deficient condition. This aspect of the unresolved item will remain open pending further justification to be provided by the licensee. All other aspects of the unresolved item are considered closed.

(Closed) Unresolved Item 259/260/296/86-14-02, Tunnel inspections. During an April 1986 tour of the CST tunnel, NRC inspectors noted corroding support base plates, pipe clamps with missing or loose fasteners, and a general deterioration of the tunnel. During an NRC tour of this tunnel in June 1988, it was noted that the area had been cleaned, painted and some repairs completed. However, in a discussion with the licensee and a review of the area drawings it was determined that this tunnel and related piping are not safety-related and this unresolved issue is closed.

(Closed) Inspector Followup Item 259,260,296/86-16-03, Usage of non-licensed operators. The inspectors had identified a concern about the use of unlicensed operators in the control room. Although unlicensed operators had sometimes been used to maintain the unit operator log book, BF-12.24, Conduct of Operations, specifies that the Shift Engineer is responsible for insuring that proper records and logs are maintained and that periodic reviews are performed. These reviews shall be at least once per shift, and documented by initials or signatures on each logsheet. In the present plant condition with all units defueled, a licensed unit operator is not required in the control room but only to be on site.

This area was reinspected in NRC Inspection Report 259,260,296/86-28 where the inspector noted that implementation of the shift engineer reviews was only being done 50% of the time. This item was left open in 86-28 pending further review.

During recent routine tours of the control room, the inspector noted that the unit operator log indicated proper reviews in all cases. Additionally the inspector was informed by the licensee that the practice of using unlicensed operators for this purpose has been stopped and will no longer be necessary. This item is closed.

(CLOSED) Unresolved Item 259,260,296/86-40-08, Failure to verify automatic initiation of primary containment isolation valves during surveillance testing. This deficiency was identified by the licensee's SI review and upgrade program concurrently with the NRC's identification of this finding. Valves 74-57, -58, -60, -61, -71, -74 and -75 were not in TS Table 3.7.A which TS 3.7.D.1 references as listing the isolation valves required to be operable. These valves were listed in TS Table 3.7.F, Primary Containment Isolation Valves located in Water Sealed Seismic Class 1 Lines. Table 3.7.F is not referenced in the text of the TS which is silent on the operability and surveillance requirements for these valves. The licensee has revised their SIs to asure that these valves will be tested for automatic initiation and closure. The licensee has submitted a revised TS table 3.7.A including all primary containment isolation valves required to be operable.

No violation will be issued for this item since the TS did not clearly identify the operability requirements for these valves, and the licensee has taken action to revise the TS and include the valves in the SI test program.

This item is closed.

(Closed) Unresolved Item 259,260,296/86-43-02, Adequacy of CAQs Reviewed This item was associated with the adequacy of generic review of CAQs to other nuclear facilities which TVA has in operation or under construction. The concern was associated with 4 CAQs documented at Bellefonte and Browns Ferry Nuclear Plants and possibly applicable to other TVA sites. At the time of the inspection there did not appear to be evidence of an adequate generic review for the identified CAQs due to a lack of documentation to verify performance.

The inspector has reviewed the licensee's response to the concerns identified in the original inspection report and determined that a violation did not exist. Subsequent to the December 1986 inspection licensee DNE personnel have provided documented evidence that potential generic reviews had been performed or that the conditions were site specific and did not apply to other sites. This item is closed.

(Closed) Inspector Followup Item 259,260,296/87-09-07, Security lighting DG building walkdown. This item had been opened to identify various concerns observed by the inspector during a walkdown performed in the security lighting diesel generator building. The inspector reviewed a licensee's Fire Protection Engineer memorandum dated August 24, 1987 (R43 87-324 882), with the attached supporting documentation. The NRC inspector conducted a tour of the building and noted the following:

- There was no noticeable diesel fuel odor.
- DG mounting frame bolts and battery mounting bolts were secure.
- Fire extinguishers indicated current inspection.
- There has been a general improvement in housekeeping in the building.

The inspector noted that the applicable fire extinguishers and fire protection valves have been added to the respective fire protection instructions to insure routine verification. The NRC inspector concluded that sufficient actions have occurred to address the concerns as identified in the original inspection reports. This item is closed.

(Closed) Inspector Followup Item 260/87-09-08, Deficiencies identified during a review of the operating instruction (OI) upgrade program. A rather extensive list of procedure deficiencies and enhancements were

noted during the inspection and tracked as a single IFI. The licensee addressed each item and initiated procedure changes as necessary. The inspector reviewed the changes and considered this item closed.

(Closed) Inspector Followup Item 259,260,296/87-30-01, Missed surveillance due to discarded chemistry composite samples. On August 26, 1987 after completion of the monthly surveillance on radiation monitor filter activity, the sample was inadvertently discarded. The sample would normally have been retained for a quarterly strontium composite as required by TS 4.8.8.3. This resulted in an unrepresentative quarterly sample to characterize the third quarter of 1987. The licensee determined that the error was due to personnel error i.e., failure by a chemist to properly table the samples resulting in another chemist mistaking it for waste. This resulted in the issuance of LER 87-023 to report the missed surveillance.

As corrective action to prevent reoccurrence of this event, the licensee has revised SI 4.8.B.2-2, Airborne Effluent - Particulate Filter Analysis (Monthly Gross Alpha), to include the requirement to properly table the sample and store in a proper storage location. The inspector was informed by licensee management personnel that there have been no similar recurrences of inadvertently discarding samples since SI-4.8.B.2-2 was revised. The inspector concluded that the licensee's actions were adequate to address the concern as identified in the original inspection report. This item is closed.

(Open) Inspector Followup Item 260/87-42-05, Insulation cut away around tailpiece valves 75-646 and 647. A testing manifold installed on Containment Spray Loop II is equipped with a three-quarter inch test/drain tailpiece containing the above two valves and comes so close to a longer insulated pipe that part of the insulation had to be removed. The clearance between the two pipes had been questioned. The inspector performed a visual inspection of the two pipes and noted that the larger pipe had only dead weight supports and could be easily moved horizontally by hand. The inspector questioned whether in a seismic event the larger non-safety-related pipe could move horizontally and shear off the smaller safety-related pipe. A CAQR was written to have design evaluate this condition. This condition will remain open.

13. General Electric Contractor Recommendations

The NRC inspector continued to review the status of the licensee's resolution/implementation of recommendations made by General Electric as part of the system review program as discussed in NRC Inspection Report 88-16. In particular the inspectors reviewed the adequacy of the licensee's classification of closed and outstanding open items for proper determination as to whether the item was required to be resolved prior to Unit 2 restart. Items classified as Category A are required to be resolved prior to unit startup while items classified as category B through E are not required to be completed prior to restart or may not be completed at all.

The inspector selected system 63, standby liquid control system, for review. The GE systems review punchlist includes a total of 36 items classified as category B, C, D or E. No items were classified as Category A. The classification of each item was compared to the restart criteria as defined in TVA system engineering memorandums dated August 10, 1987, and November 12, 1987 (R40 870810 976, R40 87110 997). The inspector did not identify any punchlist items which did not appear to be properly classified. However two concerns were identified and discussed with the licensee.

- Several items associated with revisions to system operating instructions, system drawings, and surveillance instructions were classified as category B and showed no completion date under status column indicating that the items were not complete. The licensee stated that the significance of the revisions had not rated a classification of category A for the items; however, the corrective action in each case was complete. The open status was to reflect pending management review of each item.

- Items 63-30, 63-33 and 63-35 were associated with control of GE Design Specifications. Design Specifications are vendor supplied design information provided for each GE system during plant construction. TVA has not updated or controlled the GE Design Specifications for Browns Ferry since initial construction. This concern exists for other GE systems also and is documented as outstanding items on other GE systems. The items are closed on the system review punchlist and status is stated as "TVA does not Control GE Design Specifications and will not be changed." The licensee has no plans to revise the Design Specifications and licensee management stated that 10 CFR 50, Appendix B, Criteria III, Design Control, requirements are met by separate design basis information resulting from the Browns Ferry Design Baseline and Verification Program (DBVP). The licensee further stated that the GE Design Specifications were used only as information and not as basis for design work. The inspector will review this area during the upcoming inspection periods.

14. Exit Interview

The inspection scope and findings were summarized on August 1, 1988, with those persons indicated in paragraph 1 above. The inspectors described the areas inspected and discussed in detail the inspection findings listed below. The licensee did not identify as proprietary any of the material provided to or reviewed by the inspectors during this inspection. Dissenting comments were not received from the licensee.

Item Number	Description and Reference
URI 88-21-01	Administrative control of high radiation area door keys as required by TS 6.8.3.2
VIO 88-21-02	Failure to perform CAQR Generic Reviews in timely manner as required by NQAM Part 1, Section 2.16.
IFI 88-21-03	RHRSW Corrosion
IFI 88-21-04	Deficiency identified during the retest of LOP/LOCA C
IFI 88-21-05	Vaulting of Completed and Approved Test Results
IFI 88-21-06	Adequacy of identifying and closing out of significant hardware test exceptions; Licensee's TE Categories 1.1 and 1.2.

15. Acronyms and Abbreviations

ATU -	Analog Trip Unit
BFN -	Browns Ferry Nuclear
BTRD -	Baseline Test Requirement Document
CAQR -	Condition Adverse to Quality Report
CSSC -	Critical Structures, Systems, and Components
DCR -	Design Change Request
DG -	Diesel Generator
DNE -	Department of Nuclear Engineering
DPT -	Differential Pressure Transmitter
ECN -	Engineering Change Notice
EECW -	Emergency Equipment Cooling Water
EFPY -	Effective Full Power Years
FCR -	Field Change Request
FPC -	Fuel Pool Cooling
IFI -	Inspector Followup Item
JTG -	Joint Test Group
LER -	Licensee Event Report
LOP/LOCA -	Loss of Power/Loss of Coulant Accident
MG -	Motor Generator
MR -	Maintenance Request

NQAM -	Nuclear Quality Assurance Manual
OI -	Operating Instructions
OSIL -	Operations Section Instruction Letter
PMT -	Post Modification Test
PORC -	Plant Operations Review Committee
QA -	Quality Assurance
RCA -	Radiologically Controlled Area
RCI -	Radiologically Control Instruction
RCW -	Raw Cooling Water
RHR -	Residual Heat Removal
RHRSW -	Residual Heat Removal Service Water
RPS -	Reactor Protection System
RTP -	Restart Test Program
SDSP -	Site Director Standard Practice
SI -	Surveillance Instruction
SOS -	Shift Operations Supervisor
TACF -	Temporary Alteration Change Form
TE -	Test Expectations
TS -	Technical Specifications