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

Report Nos.: 50-259/88-18, 50-260/88-18, and 50-296/88-18

Licensee: Tennessee Valley Authority
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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 Conducted: June 12-30, 1988

Inspector: for W. S. Little 9/15/88
C. R. Brooks, Acting Senior Resident Inspector — Date Signed

Accompanying Personnel: E. F. Christnot, Resident Inspector
W. C. Bearden, Resident Inspector
A. H. Johnson, Project Engineer

Approved by: W. S. Little 9/15/88
W. S. Little, Section Chief — Date Signed
Inspection Programs
TVA Projects Division

SUMMARY

Scope: This routine inspection was conducted in the areas of operational safety, maintenance observation, independent audits, and restart test program.

Results: One violation (260/88-18-02) was identified for failure to document an inadvertent overload condition on the "D" Diesel Generator on a Condition Adverse to Quality Report (CAQR) (see paragraph 5.c.) (Restart item)

The following inspection followup items were identified:

IFI 260/88-18-01, Undocumented cable conductor splice. (Restart item)

IFI 260/88-18-03, Test deficiencies identified during LOP/LOCA Test "A". (Restart item)

IFI 260/88-18-04, Documentation of offsite voltage during LOP/LOCA Test "B". (Restart item)

IFI 259,260/88-18-05, Major discrepancies identified during LOP/LOCA Test "C". (Restart item)

IFI 260/88-18-06, Deficiencies identified during LOP/LOCA Test "D". (Restart item)

REPORT DETAILS

1. Persons Contacted

Licensee Employees

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

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

Acronyms and abbreviations used throughout the report are listed in the last section of this report.

*Attended exit interview

2. Operational Safety (71707, 71710)

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

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

During a routine tour on June 27, 1988, the inspector observed a compensatory fire watch posted in the "B" Shutdown Board (SDB) room who appeared to be inattentive to his duties. The inspector initiated a discussion with the firewatch in order to ensure that he was alert and then notified a licensee representative who initiated appropriate disciplinary action.

During a routine tour on June 29, 1988, the inspector observed an unapproved flammable liquid container in a storage cage on the Refuel Floor. Upon interviewing Refuel Floor personnel, the inspector learned that a glass reagent-type bottle was being used to transport acetone from the Chemistry Lab to the Refuel Floor for use in cleaning fuel channels and other fuel related components. The empty bottle was stored for future use. The use of the bottle is contrary to the requirements of Standard Practice 14.38, Hazardous Chemicals/Flammable or Combustible Liquids, which calls for UL listed cans. The inspector contacted a site fire protection engineer who removed the bottle and instructed refuel floor and chemistry lab personnel on the use of proper containers. Since the bottle was not being used, it was immediately removed, and personnel were reinstructed, no violation or deviation will be issued.

3. Maintenance Observation (62703)

Plant maintenance activities on 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 Technical Specification limiting conditions for operations were met; activities were accomplished using approved procedures; functional testing and/or calibrations were performed prior to returning components or systems to service; quality control records were maintained; activities were accomplished by qualified personnel; parts and materials used were properly certified; power tagout clearance procedures were adhered to; and radiological controls were implemented as required.

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

a. Mechanical Maintenance Instruction (MMI) 190, Control Rod Drive Unlatching.

b. MMI 7, Control Rod Drive Changeout.

c. Operating Instruction (OI)-85, Control Rod Insert and Withdraw Timing, portions performed as post-maintenance testing.

d. Diesel Generator (DG) "D" troubleshooting and corrective maintenance activities related to the inadvertent overload on June 21, 1988 (refer to paragraph 5.c, of this report for details).

e. MR 880074 dated May 10, 1988, was indicated as being outstanding during a tour conducted on June 27, 1988. This MR documented a need for cleaning the Units 1/2 "B" DG batteries. The inspector observed a heavy layer of dust and white particulate material on top of the batteries. In addition, the inspector noted that on four of the battery cells, the spark arrestor vent covers were missing and the particulate material had entered into the battery electrolyte through the openings. The inspector informed a licensee representative of the deficient condition and also pointed out to the plant maintenance supervisor that the more than seven week turnaround on completing a routine "request for cleaning" MR was excessive. The batteries were immediately cleaned and a small quantity of the material in the electrolyte was evaluated as having no impact on battery performance.

Within this area no violations or deviations were found.

4. Independent Audits (40704)

a. Independent Safety Engineering Group (ISEG)

The licensee implemented the ISEG in early 1988 as discussed in the BFNP Nuclear Performance Plan. The inspector reviewed the last three monthly reports and found them to be well focussed on current industry problems as well as specific site weaknesses. The reports were professional and thorough. The ISEG review topics included: (1) the Licensee Event Report (LER) program; (2) Control Air System; (3) NRC Information Notices; and (4) Plant Operations Review Committee (PORC) review of plant modifications. With continued diligence, the ISEG should satisfactorily fulfill its role in providing appraisals of the quality and safety of operation to the licensee's upper management.

b. American Nuclear Insurers Nuclear Liability Insurance Inspection

A nuclear liability insurance inspection was conducted by the American Nuclear Insurers during February 1988. The inspection report and licensee's response to the report were made available to the resident inspectors. The inspection focused on Onsite Safety Review, Operations, Radwaste Management, Health Physics, and Plant Water Chemistry. Three recommendations were made and two suggestions were offered. The inspection findings did not point to any programmatic weaknesses not already known to the NRC. The resident had attended the exit and considered the most significant weakness identified to be in the area of root cause analysis and recurrence control.

c. Safety System Functional Inspection (SSFI)

The licensee's QA organization coordinated an SSFI performed primarily by an outside contractor during the period of June 6 - July 1, 1988. The inspection was patterned after the NRC's SSFI as described in NRC Inspection Manual Chapter 2515. The team identified 24 concerns documented as "observations". Fourteen of the observations were judged by the team to be safety significant or potentially significant depending upon the outcome of further review and analysis to be performed by the licensee. The team reviewed the RHRSW and EECW systems. The majority of the observations were found to have been previously identified by the licensee or the NRC and in some formal tracking program for closeout.

One deficiency was found to impact system operability under the existing plant conditions. Due to recurring failures of valves 67-50 and 67-51, the potential for failure of these valves was judged to be significant. These are cross-connect valves between the safety-related EECW system and the non-safety-related RCW system. Should these valves fail to close as required under certain circumstances, insufficient cooling water to the diesel engine coolers could result. The licensee suspended fuel handling operations on June 30, 1988, until the valves were tagged closed on a clearance hold order to eliminate this concern.

The SSFI team made several conclusions from the characterization of the findings. The maintenance area was judged to be the weakest of those evaluated. Weaknesses were also noted in design calculation verification and licensee review of calculations performed by contractors. The team further detected a lack of sensitivity among plant personnel as far as respect for the operational status of the system. This may be attributed to the extensive shutdown during which the system status was not maintained and the mind-set that accompanies such a shutdown.

The final SSFI report will be reviewed and TVA's response to the findings will be monitored.

No violations or deviations were identified.

5. Restart Test Program (RTP)

The inspector attended RTP status meetings; reviewed RTP test procedures, test specifications, and baseline test requirement documents; observed RTP tests and associated tests performances; and reviewed RTP test results including test exceptions. The inspector also attended selected Restart Operations Center (War Room) and JTG meetings.

a. Summary of RTP Activities

(1) RTP-03A, Reactor Feedwater

This test was approved by the JTG for performance on June 16, 1988. It consisted of verifying various signals, including reactor level and pressure signals to the reactor protective system and high reactor level trip signals to the HPCI and RCIC injection system turbines. The test was completed with the exception of section 5.16 which, in part, requires verifying high reactor pressure trip of the recirculation pumps.

(2) RTP-23, RHRSW

Outages continued throughout most of this reporting period with the majority of effort aimed at completing the replacement of Dresser Couplings and completing the hydrostatic test and restoration of the "A1", "A2", "B1", "B2", "C1", "C2", "D1", and "D2" pumps. Flood level switch replacement continued to be a problem, in that a dedicated switch has not been found.

(3) RTP-03B, RXFW

This test was released for performance by the JTG on June 30, 1988. It deals mainly with the RXFW pumps and their trips including low main condensor vacuum and high reactor water level. Monitoring of this test will be covered in future NRC reports.

(4) RTP-024, RCW

This test was completed during this reporting period with 47 total tests exceptions identified. Of these exceptions, seven had hardware significance. MRs were written to correct six of them; however, TE-046, which deals with check valve 1-24-852, will require DNE resolution.

(5) RTP-030, Diesel Generator and Reactor Building Ventilation

This test continues to be delayed by parts requirements, especially flow switches and damper motors. Approximately 24 test exceptions have been identified with the majority involving the failure of DG fans to stop or start as required. CAQR 88-0524 was issued as a result of this problem.

(6) RTP-31A, Control Building Heating, Ventilation, and Air Conditioning

Section 5.1 of this test was released by the JTG for performance in order to support the LOP/LOCA series of tests. This section mainly involved the CREVS. Additional tests will start after the completion of the LOP/LOCA tests. No specific test exceptions have been identified.

(7) RTP-31B, Control Building HVAC

Before this test can start, DNE must issue revision number three (Rev. 3) to BTRD-014. As of the end of June 1988, this revision was issued and the RTG group was in the final stages of writing and approving RTP-031B.

(8) RTP-39, CO2 Storage and Fire Protection

This test was completed during this reporting period with no specific test exceptions identified. The test consisted of smoke detector activation, CO2 logic testing in DG rooms, and loss of control power tests.

(9) RTP-057-1, 120 Volt DC Battery

This test involves the 8 DGs and covers a total of 16 battery chargers and battery racks. Prior to each test on the battery chargers, the filter capacitors must be changed. As of the end of June 1988, DNE was revising the BTRD and the RTP group was changing the system test specification and the RTP procedure. No test exceptions were identified.

(10) RTP-57-2, 120 V AC Distribution (120V DIST)

This test was virtually completed during this reporting period. One outstanding test exception was written involving section 5.2 dealing with the performance of Test Instruction (TI)-73B, which could not be performed because of the fire damage in the Unit 2 Drywell.

(11) RTP-57-4, 480 Volt Distribution (480V DIST)

This test was virtually completed; however, a total of 53 test exceptions were identified. Of these test exceptions, ten had hardware related issues.

(12) RTP-57-5, 4160 Volt Distribution (4 KV DISTRIBUTION)

This test is ongoing and has identified approximately 20 test exceptions; however, only two have hardware significance. One of the major restraints deals with relay setpoints and tolerances, which were to be supplied by DNE through a contractor. The six special tests involving the Unit 1/2 DG voltage regulators and speed governors have all been completed and are awaiting approval by DNE.

(13) RTP-067, EECW

This system, as with system 23 (RHRSW), was also involved with the replacement of Dresser couplings and the system hydro's. The testing in progress involved mainly section 5.8 which deals with various chillers such as the control room emergency chillers and the SDB room chillers. One hardware test exception was identified involving temperature control valve 3-TCV-67-83.

(14) RTP-069, RWCU

This test was released by the JTG on June 17, 1987, and the majority of effort so far has been directed at completing section 2.0, Prerequisites, and section 5.1, Valve Stroke Tests. No significant test exceptions have been identified

(15) RTP-070, RBCCW

This test was partially released as reported in NRC report 88-10. Since it started, approximately 27 test exceptions have been identified with 8 having hardware related issues.

(16) RTP-073, HPCI

The JTG approved this test; however, it has not released the test for performance.

(17) RTP-074, RHR

This test is being impacted in Section 5.5 which requires testing of the drywell spray upper and lower header, by the hydrolazing of the lower header. This activity was set to start on or about July 5, 1988. No significant hardware test exceptions have been identified.

(18) RTP-082, Diesel Generators

This system is undergoing a series of special tests involving all eight DGs. These are being performed by the system engineers working with vendor representatives and onsite DNE personnel. A total of ten special tests are involved and will result in the DG's speed governors (hydraulic actuators) and voltage regulators being calibrated as well as optimized for plant operations. This system test has identified over 50 test exceptions; however, only three have significant hardware issues. See Section 5.c. of this report for additional information on DG testing.

(19) RTP-084, Containment Atmospheric Dilution

This test was released by the JTG for performance on May 25, 1988. The majority of effort has been directed at completing the prerequisites. Section 5.1 deals with verifying the logic on a series of valves and replacing Class 1E splices with qualified Raychem splices.

(20) RTP-085, Control Rod Drive

This test was delayed by the changeout of 82 rod drive mechanisms and the repairs to several rod position indicators. Section 5.2, Rod Position Indicating System Function Test, has started and no significant test exceptions have been identified.

(21) RTP-092, Neutron Monitoring

This test involves the intermediate range monitors (IRMs), source range monitors (SRMs), and the average power range monitors (APRMs). The test was released for performance by the JTG on May 23, 1988; however, a switch involved in the inop bypass for the SRMs, IRMs, and APRMs were not easily obtained and contributed to delays. No significant hardware test exceptions were identified.

(22) RTP-99, Reactor Protective System (RPS)

This test was released by the JTG for performance on June 17, 1988, and involves all four channels of the RPS. One significant hardware test exception was identified involving Section 5.27, which requires that the reactor mode switch contacts be verified, and concerns wires in the panel not being marked.

(23) RTP-BUC, Backup Control Test (BUC)

This test procedure is over 800 pages long and includes the 4KV SDBs, 480 V SDBs, DG auxiliary boards, 480V reactor motor operated valve (MOV) boards, and the 250 V reactor MOV boards (also any electrical system that has backup controls). The test was released for performance by the JTG on June 13, 1988, and has been performed on a continuous basis. No significant test exceptions were identified.

The above tests were in progress and in various stages of completion at the end of this reporting period. The inspector will continue to observe these activities on a day-to-day basis.

b. RTP-075, Core Spray

During the performance of RTP-075, Core Spray, the RTP group was using SI-4.2.B-39A(II) to meet the requirements of step 5.6.5 of the RTP test. This was to prove that the CS system Loop II would perform automatically when called upon to do so. However, during the performance of the SI, the "2D" CS pump automatic circuit breaker failed to close when required. MR A-860183 was written to troubleshoot the problem and the RTP personnel wrote TE #9 to document the failure. During troubleshooting, maintenace personnel discovered an undocumented splice in the white lead of cable ES-2554-II, which was in a cable tray surrounded by flamemastic and located in the Unit 2 reactor building. The maintenance personnel wrote CAQR BFP-88-0375 to document this discovery. This item is identified as inspector followup item (IFI) 260/88-18-01, Undocumented Cable Conductor Splice in Core Spray Logic Wiring. This item is required to be closed out prior to restart. The disposition of this CAQR will be reviewed in a future inspection.

One significant test exception involving the stroke time for four flow control valves was identified.

c. RTP-082 Diesel Generators

DG "D" was inadvertently overloaded on June 21, 1988, during the conduct of Special Test (ST) 88-09, Diesel Generator Governor and Voltage Regulator Calibration. For a period of about 30 seconds, control room indications of KW, KVAR, and AMPS were off scale high. The incident was caused by unintentional shorting of test leads by an electrician who was directed to take confirmatory measurements of a parameter being monitored on a recorder. The shorting caused fuses to blow resulting in a loss of the voltage regulator and governor. Since the DG was in parallel with the grid during the test, it immediately accepted this additional load. An apparently pre-existing and undetected failure of an SBM switch (cell switch) in the DG breaker compartment prevented an automatic clearing of the overload, thus increasing the potential damage to the DG. The event was terminated when the control room operator tripped the DG breaker.

Following the event, the electrical integrity of the DG was confirmed by the satisfactory performance of a high potential test at 10,000 VDC for one minute. Visual inspections were conducted of the insulation system by an experienced vendor representative. Some hairline cracks in the insulation were detected which were determined to have been recently created. These cracks were indicative of coil movement relative to the fixed components and could have resulted from either the overload or some of the more recent stressful restart testing. The vendor representative recommended followup high potential testing at 6, 12 and 24 months in order to detect any further degradation of the insulation system as a result of the overload. No mechanical inspections were performed on the diesel engine. The licensee relied upon the expertise of the vendor representative who was present at the DG during the overload to make this judgment.

The licensee prepared a critique on the event which concluded that the capability of the DG to perform its intended function had not been affected. Following a review by the PORC on June 25, 1988, the DG was declared fully operable.

The inspector reviewed the event and asked for additional information on: 1) the cause of failure for the SBM cell switch and whether the other DG cell switches had been tested; 2) an approximation of the magnitude of the overload; 3) a comparison of diesel operating parameters such as vibration, exhaust/cylinder/cooling water temperature, oil pressures, oil analysis, and oil and fuel consumption before the event with those after the event; and 4) other diagnostics and/or inspections contained in the vendor technical manuals.

During the course of the review, the inspector learned that the incident was not documented, as required, on a CAQR. This may have been a contributing cause to the fragmented approach used to evaluate and document the condition of the diesel. Part I, Section 2.16, Step 2.1.1.F of the Nuclear Quality Assurance Manual requires that a condition adverse to quality (CAQ) be documented on a CAQR for items which have been subjected to conditions for which they have not been designed such as overpressure, overvoltage, overheating, overstressing, or environmental conditions hazardous to their function. Failure to initiate a CAQR for the overloaded DG is considered a violation of 10 CFR 50, Appendix B, Criterion V (259,260,296/88-18-02).

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

The licensee's RTP group and the system engineering group, with support from the other major Browns Ferry site organizations, performed a series of four tests referred to as LOP/LOCA testing, which started on May 29, 1988, with LOP/LOCA Test "A" and ended at approximately midnight on June 8, 1988, with LOP/LOCA Test "D". Numerous pretest meetings were held such as LOP/LOCA thrice weekly punchlist, daily RTP status, JTG as needed, and a daily morning meeting with the major site organizations to update the LOP/LOCA testing status. A total of nine inspectors, five region based and four residents, observed the performance of the four tests, with LOP/LOCA Test "C" receiving the most coverage. The procedures used for the tests were numbered: 2-BFN-RTP-L/L A (RTP-L/L A), 2-BFN-RTP-L/L B (RTP-L/L B), 2-BFN-RTP-L/L C (RTP-L/L C), and 2-BFN-RTP-L/L D (RTP-L/L D). Revision 0 of the procedures were reviewed by the JTG and approved by the Plant Manager shortly before the performance of the actual tests. Last minute changes to the tests were reviewed and approved the day the tests were conducted and RTP-L/L C and RTP-L/L D were revised approximately two days before the actual tests.

Each test procedure was reviewed and found to be adequate.

(1) LOP/LOCA Test "A"

This test was basically a LOP test, sometimes referred to as a "blackout" test, and was performed on May 29, 1988. Section 5.3 of the test called for the simultaneous tripping of the three main power feeds coming into the plant from the switchyard. The inspectors used NRR inspection modules 70306B, Loss of Offsite Power, and 70441B, Emergency/Standby Power Supply System Test, as guidance before, during and after the test. Test section 5.4, Plant Performance, required that all eight DGs start and tie onto their respective SDBs; that various 4-KV and 480 V load alignments be in accordance with the applicable procedure steps; that all three units receive a scram signal from loss of RPS Motor-Generator (M/G) sets; and that reactor MOV electrical boards and ventilation boards be in the alignment specified by applicable procedure steps. The inspectors made the following observations:

(a) Unit 3 DG "3C" did not appear to close onto its SDB (3EC). However, post test review of the visicorder traces indicated that the breaker closed and stayed closed for only five cycles, approximately one-tenth of a second, and then tripped off. The licensee's representatives informed the inspector that an item of M&TE used to time the closure of the breaker shorted out, picked up the DC trip coil of the breaker, and caused the breaker to trip shortly after closure.

(b) Unit 2 RPS M/G set "2B" did not trip off upon loss of power (the breaker supplies the power to the motor). However, other scram signals from the RPS did initiate a full scram for Unit 2.

(c) Several individual items such as recirculating M/G set oil pumps "2A-1", "2A-2", "2B-1", and "2B-2" and control and service air compressor "D" were exceptions to the test as far as their supply breakers tripping.

The above items are tracked as IFI 260/88-18-03, Test Deficiencies Identified During LOP/LOCA Test "A". This item is required to be closed out prior to restart. The conduct of the test by the licensee's representatives was not a well run test in that the control room activities appeared at times to be very hectic and a large number of personnel, more than what seemed necessary, were in the control room during the test. Improvement in professionalism by all personnel and better crowd control was needed.

(2) LOP/LOCA Test "B"

This test was basically a LOCA test using offsite power (the grid) as the power source and was performed on June 1, 1988. Section 5 of the test called for the application of a high drywell pressure signal of approximately 3.0 psig using temporary test tubing and a pneumatic calibrator installed at transmitters PT 64-58B and PT 64-58D. This signal in conjunction with the already atmospheric pressure of the reactor vessel initiated a LOCA signal. The basic requirements of the test were that all eight DGs start (they were not to close onto their respective SDBs); all four Unit 2 CS pumps start in the proper sequencing and at the proper times; all four RHR pumps start in the low pressure core injection mode in the proper sequencing and at the proper times; and all four RHRSW pumps used for EECW start in the proper sequencing and at the proper times. Additional auxiliary equipment such as reactor recirculation discharge valve closure, RHR pump cooler fans start, HPCI logic trips, etc., were verified as required to meet test acceptance. The inspectors made the following observations:

(a) An inspector was stationed at the Unit 3 control room DG electrical panel to observe Unit 3 DG operation upon initiation of the LOCA event. Test procedure requirements were that upon initiation of the LOCA event, Unit 3 DGs ("3A", "3B", "3C", and "3D") start automatically, attain rated voltage and frequency, and remain unconnected to their respective 4-KV SDBs, and the SDB mode switches transfer from "auto" to "manual". Observations of annunciator and alarm panel indicator lights confirmed the aforementioned test requirements for the Unit 3 DGs.

(b) An NRC inspector was stationed at the Unit 2 control room station and proceeded to the SDBs upon initiation of the LOCA events. The inspector observed TVA data takers and QA inspectors as well. The inspector observed no deviations or violations of procedure and test requirements.

(c) An NRC inspector observed the control room activities inside the Unit 2 main control panel "horsehoe." The inspector used Section 5.2 of the test to verify and document the performance of the four CS pumps, the four RHR pumps in the low pressure core injection mode, and the four RHRSW pumps used for EECW. Verification of operation involved recording motor amperage, system flows, and system pressures. All 12 pumps indicated electrical current flow, with the CS trains indicating approximately 6200 gallons per minute (gpm) and approximately 250 psig each, and the RHR trains indicating approximately 12000 gpm and approximately 275 psig each. Overall evaluation was that the major items of equipment responded as required by the test.

Upon post test review by all the inspectors involved, it was noted that the facilities test personnel appeared somewhat confused by what was occurring at different stages of the test. The control room activities appeared to be more controlled than during LOP/LOCA "A". It was noted that no direct monitoring of the incoming voltage from the grid was made by the licensee's representatives during this test. The inspector was informed that this will be documented by calculation. This item is identified as IFI 260/88-18-04, Documentation of Offsite Voltage During LOP/LOCA Test "B". This item is required to be closed out prior to restart. The results of the test will be reviewed and commented on in a future NRC report.

(3) LOP/LOCA Test "C"

This test was a combination of a LOP signal followed six seconds later by a LOCA signal with the "B" DG disabled and was performed on June 5, 1988. Section 5.3 of the test called for simultaneously opening the three main power feeds coming into the plant from the switchyard and after approximately a six second time delay initiating the triple low (Lo-Lo-Lo) or Level 1 reactor vessel level trip; thereby setting up a LOP followed by a LOCA. The inspectors used NRR modules 70316, Loss of Offsite Power Test, 70434, Engineered Safety Features Actuation System Test, and 70441, Emergency/Standby Power System Test, as guidance before, during and after the test. Test section 5.4, Plant Performance, required that all eight DGs with the exception of Unit 3 DG "B" start and tie onto their respective SDBs; that the "2A" and "2C" CS pumps start; that the "2A", "2B", and "2C" RHR pumps start; that the applicable RHRSW pumps start; and that all three units receive a scram signal. Additional auxiliary equipment as outlined in the procedure were to activate, including Standby Gas Treatment and Control Room Emergency Ventilation as required to meet test acceptance criteria.

The inspectors made the following observations:

(a) Unit 2 Control Room Observations

After the test was initiated at 2:16 p.m., the inspector observed that a scram signal was received. RHR pumps "2A", "2B", "2C" and CS pumps "2A" and "2C" started as planned. Power was lost to the "2A" reactor MOV board which did not transfer as expected. The transfer was manually performed at 2:29 p.m., and the CS inboard injection valve (2-FCV-75-25) opened (procedure step 5.4.9.1).

Prior to the test, RCIC was inoperable and procedure step 5.4.12 could not be met. The "RCIC Relay Logic Power Failure" annunciator was observed. The "2B" reactor MOV board is supplied by SDB "D" and DG "D" was not tested. Accordingly, procedure steps 5.4.9.2 and 5.4.19.2 could not be checked due to an absence of power.

RHR System II flow and pressure could not be checked because I&C Bus "B" had no power (procedure step 5.4.24). At 3:02 p.m., 4160 V SDB "D" was restored and these were checked.

The alarm and annunciation table, Appendix E, had the following alarms received prior to the test:

Unit Preferred Supply Abnormal
Reactor Auto Scram Channel A
Reactor Auto Scram Channel B

The "Turbine Tripped Electrical Trouble" annunciator was not received.

(b) During LOP/LOCA Test "C" an inspector observed Unit 3 control room activities with regard to alarm and annunciator panels and noted that the Unit 3 DG started and closed on the 4160 V SDBs. In addition, selected circuit breakers on the Unit 3 480V SDBs "3A" and "3B" were checked to verify and obtain their LOP/LOCA positions. The following problems were identified during the test:

- DGs "3A", "3C" and "3D" came up to rated speed and voltage but did not tie-on to their respective 4160 V SDB. Troubleshooting and review of logic circuitry indicated that the DG "3A", "3C" and "3D" output breakers closed at ~6.5 seconds and at ~6.75 seconds a LOCA signal opened the 3 DG breakers. During recharging of the breakers, the anti-pump relay (52Y) had sealed in preventing the reclosure of the DG breakers (i.e., DG breaker opened ~0.3 seconds and the anti-pump relay responded as designed to a fault condition on the SDB although the LOCA signal was the reason for the DG breakers being opened). An operator was sent to the switchgear room where a remote panel is available for local operations. The operator slowly placed the normal/emergency switch (43) to the emergency position. The switch is a break before make design and thus de-energized the 52Y anti-pump relay setting up the circuitry for closure of the three DG breakers.

- The inspector also identified that the data recorder did not have the latest data sheets for recording breaker positions on 480 V SDBs "3A" and "3B".

(c) An NRC inspector was stationed in the Unit 1 control room and proceeded to the SDBs upon initiation of the LOP/LOCA event. The inspector observed TVA data takers and QA inspectors as well.

(d) An NRC inspector observed the initiation of the LOCA signal in the Unit 2 auxiliary instrument room and noted that relay 16A-K29 in panel 9-42 did not de-energize as required for the test. This relay is part of the primary containment isolation logic.

The inspector then proceeded to SDB "D" and there observed a licensee representative performing manual manipulations on 480V reactor MOV board "2A". It was later determined that this board was not initially lined up per section 5.2, Initial Lineup of the Test. These two items were considered to be major concerns for the test.

(e) An NRC inspector was positioned in front of SDB "3EC" near DG breaker "3C". After the LOP initiation, the inspector noted that the open/closed flag on the breaker indicated momentarily closed then immediately indicated open and stayed in that position. The inspector then proceeded to SDBs "3EA", "3EB", and "3ED" and noted that the breakers for DGs "3A" and "3D" were both open. Of the Unit 3 DGs, only the output breaker for "3B" was closed on SDB "3EB". It was later discovered that the following had occurred:

Sequence of Events

Time	
1.5 seconds	Loss of Power - 4KV SDB De-energized; DGs Start
5 seconds	Board undervoltage device times out and closes contacts in breaker closure circuit (2-211-4A3)
~6.5 seconds	DGs "3A", "3C", and "3D" come up to speed and close contacts in breaker closure circuits (VSR1, VSR2)
6.55 seconds	DG breakers close ("3A", "3C", "3D")
6.55 seconds	Accident signal injected, DG breakers "3A", "3C", and "3D" tripped via LOCA signal, CASA-2 contacts closed, 52a (contact 4,4c) closed on breaker closure. This energized 52T and tripped the breaker

~6.55 seconds — When breaker closed, 452M and 52Y energized to recharge breaker. Energizing 52Y isolates 52X (closure coil) from circuit.

Note: It takes 2 seconds to recharge the breaker. The undervoltage device did not have time to reset; therefore, a closure signal is sealed into the circuit. This energizes 52Y through contacts 7-1. This keeps the breakers locked out. The inspector considered this to be a major concern for the test.

(f) An NRC inspector observed the control room activities mainly involved with the Unit 1/2 DG board. No deficiencies were identified.

Because of the problems that occurred a major part or all of Test C will be rerun.

The licensee documented the locking out of the Unit 3 DG output breakers on CAQR BFP-88-0394, and initiated action to write a LER. These items are being tracked as IFI 259,260,296/88-18-05, Major Discrepancies Identified During LOP/LOCA Test "C". This item is required to be closed out prior to restart.

(4) LOP/LOCA Test "D"

The test was a combination of a LOP signal followed at three seconds by a LOCA signal. Due to the technical/hardware problem encountered with the six second LOCA signal discovered during LOP/LOCA Test "C", the time for initiation of the LOCA signal was changed from six seconds to three seconds. This test was performed on July 8, 1988, with the Unit 2 DC battery disconnected. Section 5.3 of the test called for the simultaneous opening of the three main power feeds coming into the plant and, after approximately a three second time delay, initiating the triple low (Lo-Lo-Lo) or level 1 reactor vessel trip; thereby, setting up a LOP followed by a LOCA. Section 5.4, Plant Performance, required that all eight DGs start (due to the disconnection of the DC battery the Unit 3 "D" DG would not tie onto SDB "3ED"); that RHR pumps "2A", "2C" and "2D" start; that trains "A" & "B" of the SBGTS and CREVS start; and that all three units receive a scram signal due to the loss of all six (two per unit) RPS M/G sets. Additional auxiliary equipment as outlined in the procedure was to activate to meet the test acceptance criteria.

The inspectors made the following observations: all eight DGs started and tied onto their respective SDBs with the exception of DG "3D"; the three units each received a scram signal due to the loss of RPS M/G sets; and equipment required to perform in order to meet the test acceptance criteria appeared to do so.

This test appeared to be the smoothest run of all four tests. However, on initial review it appears that like LOP/LOCA Test "C", some equipment was not lined up initially as required by section 5.2 of the test. This item is identified as IFI 260/88-18-06, Deficiencies Identified During LOP/LOCA Test "D". This item is required to be closed out prior to restart.

General observations made and comments about the LOP/LOCA series of tests are as follows:

The inspectors were concerned that the large number of test exceptions make it difficult to determine if in fact they have met the baseline program.

With the exception of the operations manager and some of the senior personnel, the operations staff did not give the appearance of proper attitude or professionalism in conduct and attire. Also, the inspectors could not readily tell who was in charge of the operations shift.

Operator logs did not contain enough detail to recreate operational events.

Proper professionalism was not displayed when the pre-op LOP/LOCA briefings were delayed in starting: cat calls and jokes by attendees were noted. Also there appeared to be a lack of seriousness and interest shown by some individuals involved in this series of RTP testing.

On positive side, the RTP technical staff appeared to be very professional and concerned. The compliance staff and site licensing staff appeared to be very professional and had been responsive to all of the NRC concerns and requests.

These observations have been discussed with various department managers at the site. It is the conclusion of the inspectors that the series of tests were not as successful as they could have been, especially LOP/LOCA Test "C". In the area of pre-test preparations, a vast improvement is called for; in the area of operator activities, improvement is needed; and in the area of test exceptions, clarification is needed. The licensee subsequently decided to rerun much of LOP/LOCA "C".

6. Review of Quality Assurance Monitoring (QAM) Surveillances

During the reporting period, the inspector observed QAM personnel in the field actively monitoring RTP testing. The following QAM surveillances were reviewed by the inspector:

Number	Description
QBF-88-0352	This surveillance documented the activities involving the RTP daily status meetings.

Number	Description
QBF-88-00436	This item documented the activities involving the ST 88-17 on DG "B" and indicated that the DG was started with the cylinder vents open.
QBF-88-00455	This item documented the activities involving ST 88-17 and RTP-082 on DG "B" and indicated that the DG was started with the load limiter on zero.
QBF-88-0465	This item documented activities involving ST 88-07 on DG "A".
QBF-88-0475	This item documented activities involving ST 88-07 on DG "A" and indicated that MR 860262 was performed to troubleshoot the DGs droop circuit.
QBF-88-0610	This item documented activities involving LOP/LOCA Test "C" and indicated that the "3A", "3C", and "3D" DGs failed to perform as expected and that relay 16A-K29 did not de-energize. Additional test exceptions were also documented.
QBF-88-0637	This item documented activities involving LOP/LOCA Test "D" and did not indicate any major equipment failure; however, several test exceptions were documented.
QBF-88-0643	This item documented activities involving ST 88-08 on DG "C".
QBF-88-0653	This item documented the followup activities of QBF-88-0455 and indicated that CAQR-BFP-88-0403 was initiated by RTP personnel. Operations personnel initiated Critique 88-027.
QBF-88-0666	This item documented the activities involving RTP-075, Core Spray, and indicated that Loop II failed an SI due to an undocumented cable splice.
QBF-88-0667	This item documented activities involving ST 88-08 on DG "D" and indicated that RHR pump "2B" tripped during the test and that an immediate temporary change (ITC) was initiated to lift a lead to prevent this from happening again.
QBF-88-0669	This item documented activities involving ST 88-09 on DG "D".

Number	Description
QBF-88-0674	This item documented activities involving LOP/LOCA Test "A" and indicated that the DG "3C" output breakers did not perform as required due to measuring and test equipment (M&TE) problems.
QBF-88-0680	This item documented activities involving ST 88-09 on DG "D".
QBF-88-0687 QBF-88-0691	These items documented activities involving the daily RTP testing status meetings.
QBF-88-0695	This item documented activities involving RTP-057-4, 4 KV Distribution.

The above reviews indicated that QAM personnel were adequately performing surveillances and identifying and documenting problems.

7. Exit Interview

The inspection scope and findings were summarized on July 8, 1988, with the Plant Manager, Superintendents, and other members of his staff. New items identified were:

a. Inspector Followup Item (IFI) 260/88-18-01, Undocumented Cable Conductor Splice.

b. Violation 259,260,296/88-18-02, Failure to Document Overload Condition on the "D" Diesel Generator on a CAQR.

c. IFI 260/88-18-03, Test Deficiencies Identified During LOP/LOCA Test "A"

d. IFI 260/88-18-04, Documentation of Offsite Voltage During LOP/LOCA Test "B".

e. IFI 259,260,296/88-18-05, Major Discrepancies Identified During LOP/LOCA Test "C".

f. IFI 260/88-18-06, Deficiencies Identified During LOP/LOCA Test "D".

The above items are all identified as restart items.

8. Acronyms and Abbreviations

BFN	-	Browns Ferry Nuclear
BTRD	-	Baseline Test Requirements Document
CAD	-	Containment Atmospheric Dilution
CAQR	-	Conditions Adverse to Quality Report
CREVS	-	Control Room Emergency Ventilation System
CS	-	Core Spray
DG	-	Diesel Generator
DNE	-	Department of Nuclear Engineering
ECCS	-	Essential Core Cooling Systems
EECW	-	Emergency Equipment Cooling Water
FI	-	Flow Indicator
HPCI	-	High Pressure Coolant Injection
HVAC	-	Heating Ventilation and Air Conditioning
ISEG	-	Independent Safety Engineering Group
JTG	-	Joint Test Group
LER	-	Licensee Event Report
LOP/LOCA	-	Loss of Power/Loss of Coolant Accident
M/G	-	Motor Generator
MR	-	Maintenance Request
M&TE	-	Measuring & Test Equipment
PORC	-	Plant Operations Review Committee
QA	-	Quality Assurance
QAM	-	Quality Assurance Monitoring
RBCCW	-	Reactor Building Closed Cooling Water
RCW	-	Raw Cooling Water
RCIC	-	Reactor Core Isolation Cooling
RHR	-	Residual Heat Removal
RHRSW	-	Residual Heat Removal Service Water
RPS	-	Reactor Protection System
RTP	-	Restart Test Program
RWCU	-	Reactor Water Cleanup
RXFW	-	Reactor Feedwater
SDB	-	Shutdown Board
SGTS	-	Standby Gas Treatment System
SI	-	Surveillance Instruction
SSFI	-	Safety System Functional Inspection
ST	-	Special Test