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

Report Nos.: 50-259/90-08, 50-260/90-08, and 50-296/90-08

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 Units 1, 2, and 3

Inspection at Browns Ferry Site near Decatur, Alabama

Inspection Conducted: March 16 - April 16, 1990

Inspectors:	for D. R. Carpenter, NRC Site Manager	5/3/90 Date Signed
	for C. A. Patterson, NRC Restart Coordinator	5/3/90 Date Signed

Accompanied by: E. Christnot, Resident Inspector
W. Bearden, Resident Inspector
K. Ivey, Resident Inspector
R. Bernhard, Project Engineer

Approved by:	W. S. Little, Section Chief, Inspection Programs, TVA Projects Division	5/3/90 Date Signed

SUMMARY

Scope:

This routine resident inspection included surveillance observations, maintenance observations, operational safety verifications, reportable occurrences, action on previous inspection findings, implementation of nuclear quality assurance plan, high potential testing of electrical cables, and site management and organization.

Results:

One violation was identified for failure to take a chemistry sample, paragraph 4. Although an employee brought this error to the attention of management there have been several missed chemistry samples during the past two years. A NCV was identified for failure to adequately control test measures during high potential testing of electrical cables, paragraph 8. The licensee took prompt corrective action to resolve this issue. The closeout of two unresolved items resulted in two non-cited violations related to HVAC design and drywell beams design, paragraphs 6.g., and 6.h.

REPORT DETAILS

1. Persons Contacted

   Licensee Employees:

   O. Zeringue, Site Director
   L. Myers, Plant Manager
   *M. Herrell, Plant Operations Manager
   *R. Smith, Project Engineer
   J. Hutton, Operations Superintendent
   A. Sorrell, Maintenance Superintendent
   G. Turner, Site Quality Assurance Manager
   P. Carier, Site Licensing Manager
   *P. Salas, Compliance Supervisor
   J. Corey, Site Radiological Control Superintendent
   R. Tuttle, Site Security Manager

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

   NRC Employees

   *D. Carpenter, Site Manager
   C. Patterson, Restart Coordinator
   *E. Christnot, Resident Inspector
   *W. Bearden, Resident Inspector
   K. Ivey, Resident Inspector

   *Attended exit interview

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

2. Surveillance Observation (61726)

   The inspectors observed and/or reviewed the performance of surveillance testing during this reporting period. The inspections consisted of a review of the SIs for technical adequacy and conformance to TS, verification of test instrument calibration, observation of the conduct of the test, confirmation of proper removal from service and return to service of the system, and a review of the test data. The inspector also verified that limiting conditions for operation were met, testing was accomplished by qualified personnel, and the SIs were completed at the required frequency. The following SIs were observed/reviewed:

   - 2-SI-2, Instrument Checks and Observations. This SI ensures that instrument checks and observations required by the TS to be performed on a once per shift, daily, or semi-weekly frequency are performed.

This SI is performed by the operations staff. The inspector noted that there were only a few checks/observations required for the current plant conditions. No deficiencies were identified.

- 2-SI-3.7.A-1(A), Suppression Chamber Narrow Range Level Instrumentation Channel A Calibration. This SI was also validated as it was performed. This included performing all steps in the procedure even if the procedure itself did not require all steps to be performed. This method ensures that all of the procedure steps are validated by performance. The SI was performed as written and no procedural changes were required for the validation. No deficiencies were identified.

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

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 limiting conditions for operations were met; activities were accomplished using approved procedures; functional testing and/or calibrations were performed prior to returning components or system to service; quality control records were maintained; activities were accomplished by qualified personnel; parts and materials used were properly certified; proper tagout clearance procedures were adhered to; Technical Specification were met; and radiological controls were implemented as required. Maintenance requests were reviewed to determine the status of outstanding work activities and to assure that priority was assigned to equipment maintenance which could affect plant safety. The inspectors observed the following maintenance activities during this report period:

- Preventive maintenance conducted on the potential transformer for the 3ED shutdown board normal feeder breaker.

- High Potential Testing of Cables

At the beginning of this reporting period, the licensee implemented a new system for the conduct of maintenance. This system is detailed in SDSP 7.6, Maintenance Management System. The new system includes changes to the methods for requesting maintenance from the old paperwork MRs to a WR card. Valid WRs are processed into WOs which provide the necessary information and instructions to perform the work. The new WR cards are individually numbered and have two sections which can be detached. One section is a large, bright orange tag which is affixed to the equipment requiring work. The tag also includes two small detachable stickers that can be placed on small devices or equipment. The other section is a detailed request for work that includes the component identifier; location; problem description; method of discovery; Operability, LCO

entry, and CAQR evaluation signoffs; and work planning sections. The inspectors noted several of the new tags and stickers on equipment in the plant. The inspectors considered this new program an example of the attention that the new maintenance organization is giving to establishing good work practices, reducing personnel errors, and eliminating the various maintenance problems which have occurred at Browns Ferry in the past.

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

4. Operational Safety Verification (71707)

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

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

General plant tours were conducted. Portions of the turbine buildings, each reactor building, and general plant areas were visited. Observations included valve 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.

a. Scram Frequency Reduction Program

The licensee has established a SFRP. The inspector reviewed the status of this program. The program is defined in SDSP 12.10, Scram Frequency Reduction Program. A long term goal for BFNP has been established of one unplanned scram or less per unit per reactor year. The instruction outlines the function, affected organizations, and goals of the Scram Frequency Reduction Team and the mechanism the SFRP will use to pursue the scram reduction goal.

The first action taken by the team was to evaluate, review, and determine corrective actions of all BFNP scrams from January 1978, to March 1985, on an individual system by system basis. Another action

was to evaluate industry recommendations primarily from the BWR Owners Group for applicability to BFNP. From these reviews and evaluations, 122 recommendations were prepared by the SFRC and approved. The status of the recommendations is as follows: 7 of the 36 hardware related recommendations are closed, and 42 of the 86 administrative related ones are closed.

One of the improvements made was to develop a procedure to manually verify continuity of MSIV solenoids prior to testing. Some of the recommendations requiring modification are planned for next cycle. A modification to install continuity indicators will be performed during the next refueling outage.

Although the benefits and success of the SFRP can only be proven during plant operations, the program is a positive step by the licensee.

b. Information Notice 89-69, Loss of Thermal Margin Caused by Channel Box Bow.

This information notice was intended to alert BWR sites of potential problems involving loss of thermal margin caused by excessive bowing of fuel channel boxes. This channel bowing resulted in a modeling error in the plant process computer, and fuel failures at one foreign BWR facility was attributed to this cause. The impact on actual versus calculated MCPR values is expected to be much greater (about 15%) for reactors operating with channels being used in a second bundle lifetime. The licensee will complete all planned actions associated with this information notice prior to Unit 2 restart.

During this reporting period the inspector met with members of licensee management to identify additional information, if any, that may have been discovered by the licensee during the fuel bundle reconstitution activities that occurred at Browns Ferry during 1988. The inspector was informed of the following:

- Browns Ferry site and corporate standards do not allow reuse of fuel channels with exposure greater than one fuel bundle lifetime. This requirement has existed since initial fuel loading and was also applied to channels used on reconstituted fuel bundles in that exposure greater than one effective bundle lifetime would not be exceeded.

- No problems with channel box bowing were identified during the reconstitution activities, however no special effort was made to look for evidence of channel box bow.

- The licensee plans to modify the associated software to provide additional MCPR margin prior to Unit 2 restart.

The NRC inspectors will follow this issue as part of a future inspection associated with recently issued NRC Bulletin No. 90-02.

c. Missed Sample

The inspectors were informed on April 5, 1990, that the licensee had failed to perform two consecutive compensatory samples on the Unit 1 Raw Cooling Water System. These samples were required by Technical Specification 3.2.D, Note D, at least every eight hours, due the Radiation Monitor, 1-RM-90-132D, having been declared inoperable on April 1, 1990. This failure to sample was discovered by licensee personnel at 1:50 p.m. on April 4, when upon taking the required samples it was determined that the previous samples obtained at 7:45 a.m. did not identify that the spare RBCCW Heat Exchanger sample was obtained. After further investigation, the licensee determined that the required sample at 1:58 a.m. on April 4 had also been missed. The licensee has determined that the failure will be reportable to the NRC in accordance with 10 CFR 50.73. Other, almost identical, examples of failure to perform required samples are documented in LERs 259/88-41, 259/88-51, and 296-88-06. Although this event was discovered by the licensee, it does not meet the criteria for a non-cited violation since the violation is similar to the violations identified in the LERs. Violation 259/90-08-01, Missed RCW Samples, will be issued for this violation of Technical Specification requirements.

One violation was identified in the Operational Safety Verification area.

5. Reportable Occurrences (92700)

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

CLOSED LER 259/89-15, Momentary Loss of Secondary Containment Caused by Failure of Welds on Door Lock Mechanism.

A breach of secondary containment occurred when personnel were leaving the refuel floor to the control building roof and both doors of the airlock were opened simultaneously. This was caused by the failure of two welds which attach the bracket that holds the lockset in the door on the refuel floor side of the airlock. The inspector reviewed the licensee's closure package for this LER. The lockset bracket was repaired. Signs were placed at the airlocks concerning proper usage of the doors. The design of the system interlock was reviewed and determined to be acceptable. There was no history of similar failures for the doors. The failure was caused by high usage during the outage. These actions were appropriate to resolve this item. This LER is closed.

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

a. (CLOSED) IFI 260/84-41-02, Stress Analysis of HPCI Discharge Pipe.

This open item concerned the failure of Unit 2 HPCI discharge pipe supports R-23 and R-24 in the 1984 time frame and the need for a stress analysis on the associated piping. This item was addressed in the Design Baseline Verification Program and the extensive Bulletin 79-14/02 modifications effort performed by the licensee. The associated pipe stress problem N1-273-06R, Pipe Support Calculations CD-Q2073-883430 (R23) and CD-Q2073-891001 (R24), and Support Design Drawings 2-47B455S0019 (R23) and 2-47B455R0024 (R24) have been issued to document the system integrity.

While these specific calculations have not been reviewed by the NRC for closure of this particular item, the NRC has conducted numerous inspections of the DBVP and IB 79-14/02 engineering calculations and has found the licensee's programs fully acceptable. This item is closed based on the calculation programs inspection efforts.

b. (OPEN) IFI 259/85-06-02, IRM Noise

This item was opened against Unit 1, but has applicability for all three units. It involves erroneous high reading on IRM channels and was believed to have been caused by electrical "cross talk" of IRM, cables at containment penetrations. For Unit 2 the SRM and IRM, two-shield coaxial cables from the detector connector to the preamplifiers were replaced with a new improved three-shield coaxial cable as recommended by GE-SIL #192. The modification was performed under ECNs 5485 and 5534. The inspectors have reviewed the ECNs, observed surveillance, and observed the SRM operation during the recent defueling of Unit 2. No problems were identified. The work performed will be tracked for Units 1 and 3 modifications, but this item is closed for Unit 2.

c. (CLOSED) IFI 260/85-51-01, Inspection of Existing Cable Tray Support Systems.

This item concerns the fact that in 1985 evidence could not be found that indicated that the cable tray support systems had been inspected to an approved procedure for verification of as built condition.

The seismic qualification of cable tray and cable tray supports at Browns Ferry Nuclear Plant Unit 2 was reviewed by NRC as documented in the Safety Evaluation dated February 5, 1987 (Ref: NRC Letter, D.R. Muller (NRC) to S.A. White (TVA), "Transmittal of Safety Evaluation Concerning the Interim Acceptance Evaluation of Seismic Qualification of Cable Tray/Supports," February 5, 1987). Issues related to cable tray and cable tray supports are covered in this safety evaluation and are closed based on that evaluation for Unit 2 only.

d. (CLOSED) IFI 259, 260, 296/86-32-03, Reactor Protection System Calibration Frequency.

This item concerned a discrepancy between the safety analysis which supported TS changes for the new RPS Analog Transmitter and Trip Units and actual plant practice. The item was reviewed in IR 88-16 and remained opened pending resolution of outstanding discrepancies. The discrepancies were that an 18 month calibration cycle was not supportable for TOBAR transmitters, and that calculations for the calibration frequency of PT-68-95 and PT-68-96 were not completed. The inspector reviewed the licensee closure package for this item. TS amendment number 167 was issued July 7, 1989, to change the calibration frequency for instrument lines containing transmitters manufactured by TOBAR to six month intervals. The inspector reviewed the TS change and SI, and they had been changed to six month intervals. The inspector reviewed the Setpoint and Scaling Calculation for PT-68-95 and PT-68-96. The calculation compared the loop accuracies to the required accuracies, setpoints, safety limits, and/or operating limits, and concluded the accuracy of the loops was acceptable for the intended function. The inspector concluded that the TS changes, revised SIs, and calculation resolved the outstanding issues from IR 88-16. This items is closed.

e. (Closed) IFI 259, 260, 296/90-05-01, ECP Corrective Actions

This IFI identified an example in which a procedure change that was made as a corrective action to a valid employee concern was subsequently deleted from the revised procedure. The ECP initiated CATDs to ensure that corrective actions were implemented for valid employee concerns. In June 1988, licensee procedures were revised to require a note with each procedure step associated with CATD corrective actions. The inspector expressed concern that procedural corrective actions completed prior to June, 1988 could still be deleted, since they did not include identifying notes. Licensee management stated that an action plan would be developed to review this concern. This item was opened to follow the licensee's actions.

The inspector discussed this issue with licensee personnel and reviewed the licensee's action plan and findings. The action plan included a review of all CATDs closed before October 10, 1988, to ensure that procedure changes were still in place. There were 14 restart CATDs and 23 non-restart CATDs identified which included procedure changes. The licensee reviewed all of the procedure changes required for the restart CATDs and a sample of nine of the non-restart CATDs. In each instance, the changes were in place in current procedures or had been revised after subsequent licensee reviews. None of the reviewed corrective actions implemented by the

CATDs had been voided. The licensee also added notes to several of the procedure steps to indicate that they were associated with ECP corrective actions.

The inspector noted that the action plan was detailed and the reviews performed were indepth. In addition, the inspector noted that the licensee's action in response to this issue was timely (this issue was identified during the previous reporting period). This item is closed.

f. (CLOSED) IFI 260/89-20-02 for Unit 2 only, CRD Seismic Analysis.

The licensee identified an apparent discrepancy between the moment of inertia (stiffness) used in a recent seismic reanalysis for the CRD housings and the moment of inertia used in the original stress evaluation. This item required extensive modifications of the CRD housing supports which was followed by the NRC-HQ staff.

This issue was identified during the NRC inspections performed from April 26 to June 28, 1989 as reported in NRC Inspection Report 50-260/89-31 dated July 17, 1989. The staff and its consultant identified three open issues in this report.

On August 14-16, 1989, the staff performed an inspection (IR 50-260/89-39 dated October 13, 1989) to review the resolution of the open items identified in IR 89-31. As a result of this NRC inspection, two items were still open.

These two items were subsequently closed in NRC IR 50-260/89-62, dated February 16, 1990.

This issue is closed for Unit 2.

g. (CLOSED) URI 260/86-06-02, for Unit 2 only. Reactor Building Control Bay HVAC Inadequate Design.

This item concerned the licensee's identification of inadequate design of HVAC supports. Interim followup of this item was reported in IR 50-259, 260, 296/89-20, paragraph 7.C.

The seismic design of the HVAC duct and supports was reviewed by the staff in its inspections of the BFNP Unit 2 Seismic Design Program. As stated in NRC Inspection Report 50-260/88-38, the staff and its consultants identified several issues relating to this item. However, all of these issues were closed in subsequent NRC inspections. The following are the open item numbers and the inspection reports where these issues were closed.

CSG-24	IR 50-260/89-42	dated February 26, 1990
CSG-29	IR 50-260/89-29	dated September 20, 1989
CSG-30	IR 50-260/88-38	dated April 19, 1989

The staff has extensively reviewed this issue under its inspections of the Seismic Design Program. There are no open items remaining for this issue. This unresolved item is closed for Unit 2 based on the above inspections. This licensee-identified violation is not being cited because criteria specified in 10 CFR 2, Appendix C, V.G.1 were satisfied . This item is closed for Unit 2, and identified as NCV 260/90-08-03, Reactor Building Control Bay HVAC Inadequate Design.

h. (CLOSED) URI 260/86-14-03 for Unit 2 only, Overstress of Drywell Beams.

This item involves licensee identified discrepancies in the drywell platform design calculations. These discrepancies included: 1) some eccentric loads were not included, 2) some uplift loads were not included, 3) some calculations were not second checked, and 4) the structural behavior of the overall platform under combined loads was not analyzed.

The structural evaluation of the drywell steel platforms were covered under the Browns Ferry Unit 2 Seismic Design Program. During the inspection of the TVA calculations for the evaluation of the drywell steel platforms, the staff and its consultants identified several items as stated in NRC Inspection Report 50-260/88-38 dated April 19, 1989. These items were numbered as CSG-10, CSG-11, CSG-12, and CSG-14. All of these issues were closed satisfactorily in later NRC inspections. The following are the open item number and the inspection report where these issues were closed.

CSG-10	IR 50-260/89-42	dated February 26, 1990
CSG-11	IR 50-260/89-32	dated November 8, 1989
CSG-12	IR 50-260/89-29	dated September 20, 1989
CSG-14	IR 50-260/89-21	dated June 15, 1989

The staff has reviewed this issue under its inspections of the Seismic Design Program. There are no open issues remaining for this item. Therefore, this unresolved item is closed for Unit 2 based on the above inspections. This licensee-identified violation is not being cited becuase criteria specified in 10 CFR 2, Appendix C, V.G.1 were satisfied. This item is closed for Unit 2, and identified as NCV 260/90-08-04, Overstress of Drywell Beams.

i. (CLOSED) URI 260/87-26-03, RHR Pump Suction Anchors and Nozzle Load Allowables are Possibly Exceeded.

This item concerns RHR load allowables as identified by the licensee in deficiency number 87-13-6 of Engineering Assurance Audit 87-13. The licensees extensive IB 79-14/02 design verification and modification program dealt with the specific problem.

The RHR anchors and nozzle qualifications are within the jurisdictional boundary of the Long Term Torus Integrity Program (LTTIP). These anchors serve as a boundary between the 79-14 stress problem N1-274-9R and the LTTIP stress problem N1-273-5R. The overlapping loads from the 79-14 stress problem have been combined with the LTTIP pipe stress problem N1-273-5R (calculation CD-Q2073-883012). This calculation properly documents the anchor loads and the pump nozzle qualification. The pipe support structural anchors are within the LTTIP program.

Because of the actions taken under these programs as part of the overall NPP activities, it is not clear that a violation existed at the time of IR 87-26. The efforts of the licensee and the review effort by the NRC staff of the calculation program have addressed this concern, and this item is considered closed for Unit 2.

j. (CLOSED) VIO 260/85-41-01, Inadequate Design Controls for Safety-Related Cable Tray Supports.

This item concerns:

(1) Cable tray supports in the control bay area were not seismically designed.

(2) Diesel generator building cable tray supports were improperly designed.

(3) Cable tray support calculations in the reactor building showed lack of thoroughness, clarity, consistency and accuracy.

(4) Design verifications had not been implemented in an acceptable manner.

The seismic qualification of cable tray and cable tray supports at Browns Ferry Nuclear Plant Unit 2 was reviewed by NRC as documented in the Safety Evaluation dated February 5, 1987 (Ref: NRC Letter, D.R. Muller (NRC) to S.A. White (TVA), "Transmittal of Safety Evaluation Concerning the Interim Acceptance Evaluation of Seismic Qualification of Cable Tray/Supports," February 5, 1987). Issues related to cable tray and cable tray supports are covered in this safety evaluation and are closed for Unit 2 only based on that evaluation.

7. Implementation of Nuclear Quality Assurance Plan (35502)

The inspector reviewed the status of implemention of the new NQAP. This plan replaces the Quality Assurance Program Description (Topical Report) TVA-TR75-1A. Included in this change is a transition from the current

NQAM to the Nuclear Procedures System. The NQAP is to be fully implemented by June 30, 1990. The licensee developed a matrix to show where NQAP requirements are implemented. Fourteen procedures were identified that will require changes. Each change has been assigned a responsible site organization for making the change. A schedule for completing the changes has been developed with the last scheduled change to be completed by June 1, 1990. The NQAP is described in TVA document TVA-NQA-PLN89-A.

8. High Potential Testing of Electrical Cables - Work Observation and Procedure Review (51061, 51063)

The inspectors followed ongoing licensee activities associated with Special Test ST-90-01, Special Test Procedure for High Potential Testing of Low Voltage Cable. The licensee's engineering organization had identified the ten conduit that had the greatest possibility for damage to cable during "pull bys" These conduit were selected for wet high potential testing to determine if any cable damage could be detected that may have caused by pull-by cable installation problems similar to those identified at the Watts Bar facility. The inspectors observed portions of the preparations and setup for the testing, and actual high potential testing for selected cables in conduit 3ES-1676-IB. Most of the cables included in this conduit were multi-conductor cables routed from the 3EB, 4KV Shutdown Board to the Unit 3 Control Room Panel 9-23. Testing was directly observed for the following cables:

3ES-2007-IB
3ES-2071-IB

The testing process consisted of determinating both ends of the cable conductors, injecting tap water into various junction boxes located in the Unit 3 Reactor Building and applying voltages up to 7200 volts D.C., to the individual conductors. The actual lifting and relanding of conductors was controlled by Work Order 90-02259 and accomplished in accordance with the requirements of MAI-3.3, Cable Termination and Splicing for Cables Rated Up to 15000 Volts. On April 5, during testing on Cable 3ES-2061-IB, the personnel performing the test noted from indications on the testing equipment that the cable being tested appeared to be shorted. After investigating the problem, the licensee determined that the opposite end of the affected conductor had not been determinated at the control room panel. The testing activities were stopped and an investigation initiated to determine the facts associated with the failure to verify that the conductors were determinated. The licensee determined that two additional conductors other than the above mentioned conductor were also not determinated at the control room panel. No evidence of damage to any equipment or conductor has been attributed to this event.

The inspector reviewed the work order and met with licensee personnel to discuss the event. The inspector determined that the work order did not uniquely identify the specific conductors to be lifted, only the cable number. Cable 3ES-2061-IB had a total of 12 conductors, all of which were

to be tested. Of the 12 conductors, seven were associated with a single terminal block and shown on a common drawing. Two conductors were spares, and the remaining three conductors were associated with a separate terminal block. These three conductors (B11G, B11R, B11RG) were the conductors that had not been determinated and are actually identified on another drawing, 45N32655-4, which had not been referenced by the work order.

The licensee personnel involved in the testing were counseled by management and cautioned on the necessity for attention to detail in the performance of assigned duties. The licensee's incident critique will be included in the pre-test briefing for future cables to be tested under ST-90-01. The test director was instructed to personally verify conductor determinations prior to performance of future high potential testing.

This failure to maintain adequate test control measures constitutes a violation, NCV 296/90-08-02, High Potential Cable Test Control Problems, 10 CFR 50 Appendix B, Criterion XI, Test Control. Due to the fact that the failure was identified by the licensee and prompt corrective action was immediately initiated, this failure satisfied the criteria specified in Section V.G.1 of the NRC Enforcement Policy for a NCV. An NOV will not be issued and a response will not be necessary.

The inspector observed that, for the initial portions of the testing conducted until April 5, there was no QA or QC participation in the ongoing activities. After the testing resumed on April 8, the inspector noted that a member of the Quality Monitoring Group was observing the testing activities.

One NCV was identified concerning High Potential Cable Test Control Problems.

9. Exit Interview (30703)

The inspection scope and findings were summarized on April 13, 1990 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	Description
259,260,296/90-08-01	VIO, Missed RCW Samples, paragraph 4
259,260,296/90-08-02	NCV, High Potential Cable Test Control Problems, paragraph 8
260/90-08-03	NCV Reactor Building Control Bay HVAC Inadequate Design, paragraph 6.g.
260/90-08-04	NCV Overstress of Drywell Beams, paragraph 6.h.

10. Acronyms

BFNP	Browns Ferry Nuclear Plant
BWR	Boiler Water Reactor
CAQR	Condition Adverse to Quality
CAR	Corrective Action Report
CATD	Corrective Action Tracking Document
CFR	Code of Federal Regulations
CRD	Control Rod Drive System
DBVP	Design Baseline and Verification Program
ECN	Engineering Change Notice
ECP	Employee Concerns Program
GE	General Electric
HPCI	High Pressure Coolant Inspection
HQ	Headquarters
HVAC	Heat, Ventilation, & Air Conditioning
IB	Enforcement Bulleting
IFI	Inspection Followup Item
IR	Inspection Report
IRM	Intermediate Range Monitor
KV	Kilovolt
LCO	Limiting Condition of Operation
LER	licensee Event Report
LTTIP	Long Term Torus Integrity Program
MAI	Modification Alteration Instruction
MCPR	Minimum Critical Power Ratio
MR	Maintenance Request
MSIV	Main Steam Isolation Valve
NCV	Non Cited Violation
NOV	Notice of Violation
NPP	Nuclear Performance Plan
NQAM	Nuclear Quality Assurance Manual
NQAP	Nuclear Quality Assurance Plan
NRC	Nuclear Regulatory Commission
PT	Pressure Transmitter
QA	Quality Assurance
QC	Quality Control
RBCCW	Reactor Building Closed Cooling Water
RCW	Raw Cooling Water
RHR	Residual Heat Removal
RPS	Reactor Protection System
SDSP	Site Director Standard Practice
SFRC	Scram Frequency Reduction Coordinator
SFRP	Scram Frequency Reduction Program
SI	Surveillance Instruction
SIL	Service Information Letter
SRM	Source Range Monitor
ST	Special Test
TS	Technical Specification

TVA	Tennessee Valley Authority
URI	Unresolved Item
VIO	Violation
WO	Work Order
WR	Work Request