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

Report Nos.: 50-259/90-37, 50-260/90-37, and 50-296/90-37

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: November 17 - December 18, 1990

Inspector: Ralph H Berhard for C. A. Patterson, NRC Restart Coordinator — 1/7/91 Date Signed

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

Approved by: Paul Kellogg, Section Chief, Inspection Programs, TVA Projects Division — 1/7/91 Date Signed

SUMMARY

Scope: This routine resident inspection included surveillance observation, operational safety verification, modifications, essential design calculations, post modification testing, power ascension testing, cold weather preparations, SPOC, NRC bulletins, temporary instructions, TMI action items, reportable occurrences, action on previous inspection findings, resolution of open items in IR 89-44, and RPIP.

Results: A URI was identified concerning problems encountered during performance of the SLC functional test, paragraph 2. An SLC pump did not develop flow during the SI. The licensee is conducting an incident investigation of this problem which will be reviewed by the inspector.

The Unit 1 Torus closure was completed, paragraph 3. Unit 1 Torus is needed to support the RHR Unit crosstie capability. Additional progress was made toward drywell closure and completion of SIs necessary for fuel load.

All remaining open items associated with the RPIP were closed, paragraph 16. This completes closure of all items necessary to close Confirmatory Order EA 84-54.

Thirteen of 14 open items in IR 260/89-44 concerning piping systems were closed, paragraph 15.

REPORT DETAILS

1. Persons Contacted

Licensee Employees:

O. Zeringue, Site Director
*L. Myers, Plant Manager
*M. Herrell, Operations Manager
J. Rupert, Project Engineer
R. Johnson, Modifications Manager
*B. McKinney, Technical Support Manager
R. Jones, Operations Superintendent
A. Sorrell, Maintenance Manager
G. Turner, Quality Assurance Manager
P. Carier, Licensing Manager
*P. Salas, Compliance Supervisor
*J. Corey, Radiological Control Manager
R. Tuttle, Security Manager

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

NRC Personnel:

*P. Kellogg, Section Chief
*C. Patterson, Restart Coordinator
*E. Christnot, Resident Inspector
*W. Bearden, Resident Inspector
*K. Ivey, Resident Inspector
*G. Humphrey, Resident Inspector

*Attended exit interview

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

2. Surveillance Observation (61726)

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

a. 1-SI-4.2.A.10, Reactor Building and Refueling Floor Ventilation Radiation Monitor Calibration and Functional Test.

The inspector observed a portion of this SI being conducted in the Unit 1 control room on December 10, 1990. This procedure had been validated during a previous performance. No deficiencies were identified during the observation of this SI.

b. 2-SI-4.2.B-1 (D), Core and Containment Cooling Systems Reactor Low Water Level Instrument Channel D Calibration.

The inspector observed portions of this SI being conducted in the Unit 2 auxiliary instrument room and reactor building. This procedure was being validated during this performance. No deficiencies were identified during the observation of this SI.

c. 2-SI-4.2.D-2A, RHR Service Water Radiation Monitor (2-RM-90-133D) Calibration/Functional Test.

The inspector observed a portion of this SI in progress in the Unit 2 control room on December 8, 1990. This SI was being validated. All sections of the SI were being performed. No problems were observed in the control room. The inspector went to the reactor building to observe installation of the radioactive test sources at the monitor but the SI was stopped. The technicians found a hold order at the monitor local panel which had not been in place the previous day when the "as found" readings were taken. The hold order was 2-90-1038-1. This was discussed with the maintenance manager. The details concerning the SI are being reviewed. The licensee plans to provide a written review of this SI performance to the inspector.

d. 2-SI-4.1.A-8(F), RPS High Water Level in Scram Discharge Tank Functional Test.

The inspector observed a portion of this SI in the Unit 2 control room on December 10, 1990. This SI had been previously validated. No problems were encountered during performance of the SI.

e. Procedure 2-SI-4.4.A.2, Standby Liquid Control System Functional Test, is performed once per cycle to verify the adequacy of the SLC system. This test demonstrates the ability of the pumps to pump the solution from the SLC tank. This test also checks the setpoint of the system relief valves and injects demineralized water to the vessel after firing one of the two explosive valves.

On November 20, 1990, during the performance of this SI, the 2A SLC pump did not develop any flow. Investigation determined that the pump was air bound following an inadequate flushing process. Prior to the performance of the SI, the ASOS directed the AUO to flush the test tank and system piping with demineralized water. It was during this flush of the suction piping that air was allowed to collect at the pump suction. The flush was not required prior to performance of the SI. The ASOS had directed this be done in an effort to insure

that the boron concentration in the test tank would be low enough to allow injection of demineralized water into the reactor vessel.

The licensee initiated an incident investigation to determine the root causes of this event and to provide corrective actions. This issue will remain open as an unresolved item URI 260/90-37-01, Problems Encountered During SLC Surveillance, pending review of the licensee's completed incident investigation report.

Additionally, the inspector noted that the pump developed a packing leak during the performance of the SI. The inspector examined the old packing after its removal from the pump. Approximately half of the compenents of the multi-piece seal were damaged. On November 29, the inspector observed performance of 2-SI-4.4.A.2, Rev. 7, through step 7.10. When the A pump was run, there was a chattering noise in the common discharge line near or at the B pump discharge valve 2-63-516. When the B pump was run, the A pump discharge line was quiet. In addition, the B pump discharge isolation valve 2-63-517 was observed to have a packing leak. The inspector identified these items to the operators performing the SI.

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

3. Operational Safety Verification (71707)

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

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

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

controls. Informal discussions were held with selected plant personnel in their functional areas during these tours.

a. Fuel Load Surveillance Testing Status

The licensee continues to make progress toward fuel loading in early 1991. Surveillances required for fuel load are being completed. A total of 223 have been completed and 130 remain.

b. Unit 2 Drywell Work Completion

The inspector performed inspections of the Unit 2 drywell to evaluate the licensee's preliminary efforts to complete all modification work and clean debris from that area for a close-out inspection. The licensee's preliminary inspections have concentrated on the upper 3 levels of the drywell. Each inspection identified various areas with deficiencies and evaluated corrections made as a result of previous inspections. Notable improvements were made in this area during the reporting period.

c. Unit 1 Torus Closure

During the reporting period, the inspector made 3 tours of the Unit 1 Torus in an effort to evaluate the licensee's final inspection and closure. Each tour identified areas that required additional clean-up and areas where the protective coatings were deficient. During the third and final inspection, the debris identified at that time was removed, the torus was accepted by the plant quality organization, and was then closed and locked by plant security. The licensee contended that the Unit 1 Torus would not be needed for steam dumping and therefore the structures which had exposed areas (missing coatings) where acceptable. These areas consisted of pieces of angle steel utilized to support conduit and areas on the structural steel where Monorails had been removed.

No violations or deviations were identified in the Operational Safety Verification area.

4. Modifications (37700, 37828)

The inspectors continued to follow modification activities to support the restart of Unit 2. This included reviews of scheduling and work control, routine meetings, and observations of field activities.

The inspector reviewed and observed the licensee's activities involved with DCN W14030, Slow Bus Transfer. This modification was installed as a result of CAQR BFP890279 which documented the overloading of 4KV Shutdown Buses 1 and 2 under LOCA conditions when preceded by a transfer of two 4KV Shutdown Boards. This results in Shutdown Boards A, B, C, and D being fed from the same shutdown buses. The 4KV Shutdown Bus is the

normal source to two boards and the alternate to the other two boards. Automatic delayed transfer from the normal to the alternate power source is initiated by an undervoltage on the normal source monitored by time delay undervoltage relays. The FSAR assumes this transfer does not occur under accident conditions and two 4KV Shutdown Boards will connect to their respective diesel generators. However, as documented in the CAQR, the accident signal which provides the blocking function is not generated until after the transfer occurs. This modification is to assure that, for any pre-accident alignment of the Auxiliary Power System, whether automatic or manual, the final alignment is consistent with the FSAR assumptions.

The modification affected the following:

- 4KV Unit Boards 1A, 2A, 1B and 2B, two breakers on each board;
- 4KV Unit Start Board 1, four breakers;
- 4KV Shutdown Boards A, B, C and D, two breakers on each board and change the load shed time delay on 480V shutdown boards 1A, 1B, 2A and 2B.

All activities were controlled by WPs, monitored by QC and reviewed by design.

No violation or deviations were identified in the Modification area.

5. Essential Design Calculations

Inspectors reviewed the following calculations during this reporting period:

ED-Q2003-880177, Revision 5, supports a new setpoint of 539 inches for RPV water level instrument 2-LT-3-203A. This instrument provides one of four redundant channels for Reactor Building and PCIS isolations and SBGT actuation. Additionally this calculation was recently revised as part of an ongoing program for verification of approximately 450 scaling and setpoint calculations.

ED-Q2003-880178, Revision 5, supports a new setpoint of 539 inches for RPV water level instrument 2-LT-3-203B. This instrument provides another of the redundant water level channels described in the above paragraph.

ED-Q2003-880179, Revision 5, supports a new setpoint of 539 inches for RPV water level instrument 2-LT-3-203C. This instrument provides another of the redundant water level channels described in the above paragraphs.

ED-Q2003-880180, Revision 5, supports a new setpoint of 539 inches for RPV water level instrument 2-LT-3-203D. This instrument provides another of the redundant water level channels described in the above paragraphs.

PGC-002-064-0, Revision 1, "Change in Torus Free Volume Per One Inch of Water Level". TS 4.7.A.2 requires monitoring of primary containment nitrogen consumption to determine the average daily nitrogen consumption for the last 24 hours. This surveillance is accomplished via procedure 2-SI-4.7.A.2.a, "Primary Containment Nitrogen Consumption and Leakage", wherein corrections are made for Suppression Chamber level changes and Drywell/Suppression Chamber venting that may occur. The licensee determined that the correction factor from inches of change in torus water level to the equivalent standard cubic feet of nitrogen, as used in the SI, was not supported. Consequently, calculation PGC-002-064-0 was developed and issued by Technical Support on 6/26/90 and its results were incorporated in the acceptance criteria correction factor for torus level change in 2-SI-4.7.A.2.a, Rev.2 issued 7/3/90.

MD-Q0064-890045, Revision 0, "Torus/Reac.Bldg. Vacuum Breaker". TS 4.7.A.3.b states that the vacuum breaker disk must open with a differential pressure of 0.5 psid. This calculation provides the acceptance criteria utilized in Revision 1 to 2-SI-4.7.A.3.b, "Suppression Chamber - Reactor Building Vacuum Breaker Cycling" for static weight required to produce the necessary opening force equivalent to 0.5 psid.

MD-Q2075-890109, Revision 0, "Core Spray Acceptance Criteria for Technical Specification Operability Surveillance". TS 4.5.1.A.d requires that each Core Spray loop be able to deliver at least 6250 gpm against a system head corresponding to a 105 psi differential pressure between the reactor vessel and the primary containment. This calculation determines the core spray pump minimum discharge pressure at rated loop flow of 6250 gpm equivalent to the TS 105 psi differential.

No violations or deviations were identified in the Design Calculations area.

6. Post Modification Testing (37828)

The inspector continued to observe and review the licensee activities in the performance of PMT. The specific PMT reviewed was PMT-BF-0.008, Appendix J, Test Requirements Matrix. This PMT was written to test DCN W10017A, Black Snake Cable Replacement. This DCN and modification implementation WPs were issued to replace electrical cables that did not meet 10 CFR 50.49 requirements. In IR 90-33, the inspector documented the observations of the performance of the PMT for RHR valve 2-FCV-74-53. The valve did not function correctly in the Appendix R mode. The licensee corrected this deficiency.

The inspector obtained documentation from NE which listed all of the Appendix R cables affected by the Black Snake issue. The inspector compared this list for System 74, RHR against the Black Snake PMT and determined that each Appendix R cable and System 74 valve was listed as tested or scheduled to be tested. Based on additional observations of the licensee's PMT implementation, each Appendix R cable and end device such

as MOVs, TSs, and SOVs are included in the PMT for the Black Snake modification.

The inspector will continue to monitor and review the licensee's activities in addressing the Black Snake/Appendix R interface.

Additional reviews of the licensee's PMT program indicated the TDs were not being written when a TD is discovered while performing a testing WP. When a deficiency is discovered during the performance of a testing WP, a WR is written without a corresponding TD. The inspector considered this a weakness in the program in that the total number of PMT TDs are not known. Consequently the number of actual TDs is not apparent and cannot be reviewed or tracked by TVA BFNP management. This item was discussed with TVA management.

7. Power Ascension Testing

The inspector reviewed the licensee's status in preparation for the power ascension of unit 2. This included installation of the TARS to evaluate designated system performances. Of the 92 analog channels to be utilized in the testing, 80 channels have been calibrated. In addition, efforts are now in progress to set up a program to check operation of the feedwater level control system and to develop a plan for the initial spin up of the reactor feedpumps and the main turbine. All efforts are being performed to support plant startup schedules.

8. Cold Weather Preparations (71714)

The NRC inspector reviewed the licensee's program to protect plant systems and equipment important to safety from cold weather conditions. The BFNP areas subject to cold weather include the intake structure which houses the RHRSW pumps (the ultimate heat sink) the CCW pumps and the fire pumps, the reactor building roof which supports the condensate transfer system head tank, the condensate storage tanks located near the unit three turbine building, the two diesel generator buildings located on the east and west sides of the reactor building, the fire protection system valve pits, the five cooling water towers, and the diesel driven fire pump buildings.

The inspector reviewed the completed procedure 0-GOI-200-1, Freeze Protection Inspection, and observed the licensee's field activities. The inspector noted that portable heaters were prestaged in the intake building, the Unit 3 DG carbon dioxide room and various other plant areas. The inspector observed that, unlike previous years, plastic instead of tarpaulin was being used to cover the grating over the RHRSW intake structure. The plastic appeared to be torn. This and additional observations were discussed with the licensee.

9. System Pre-Operability Checklist (71707)

The inspectors continued to monitor the licensee's activities to evaluate and upgrade both plant equipment and documentation as necessary to insure that plant systems are in compliance with applicable standards and commitments to support their required functions. As of December 17, 1990, 26 of the 35 systems required to support Unit 2 fuel re-load had been completed and 43 of the 80 systems required to support the unit 2 plant operation were completed.

Those systems reviewed by the inspectors during this reporting period are listed as follows:

a. Residual Heat Removal Service Water (System 23)

The licensee's SPOC process was completed for this system on December 5, 1990. There were no exceptions to the SPOC program taken at the time of completion. Six deferrals were taken. Disposition of each was assigned to a plant start-up milestone or a design change closure. A review of each revealed that the system would be operable within the time frame assigned as necessary to support plant operation.

Major portions of the system were walked down by the inspectors and the status of the equipment was determined to be acceptable. In addition, the inspectors reviewed the SPAE and SPOC package and found each to be completed. No adverse conditions were noted within the areas reviewed.

b. Diesel Generator and Reactor Building Ventilation (System 30)

The inspector reviewed the completed SPOC package for System 30. The SPOC was completed on November 12, 1990. Four deferrals were taken against the System. For two of the deferrals the work was complete for system 30 but work remained under the applicable ECNs. One deferral was tied to System 31, Control Bay Heating, Ventilation and Air-Conditioning System, because the chill water components were tied to that system. One deferral was taken against the final JTG approval of restart test 2-BFN-RTP-030. Final approval of this was being deferred until all system test requirements were completed. No deficiencies were noted in the completed SPOC package.

c. Secondary Containment (System 64C)

The SPOC for this system was completed on December 10, 1990. The inspector accompanied the system engineer, Plant Manager, and members of the operations and maintenance organization on a final walkdown of the system on November 21, 1990. The system appeared to be in good condition and only minor work items were identified. The inspector began review of the completed SPOC package on December 17, 1990. The SPOC included three exceptions and three deferrals. Two of the

exceptions included the performance of SIs as part of the PMT process. The third exception involved the completion of hanger modifications for piping which penetrates secondary containment. The exceptions affect operability of the system and must be closed prior to fuel load. No deficiencies were identified with the SPOC package by the end of the reporting period.

d. Primary Containment Isolation (System 64D)

The inspector reviewed the completed SPOC package for system 64D. The SPOC was completed on November 17, 1990. Four deferrals were taken against the system. All of the deferrals were tied to startup. Three of the deferrals were for ECNs or workplans for which all system 64D work was field complete but awaiting closure of other system work. One deferral was against a restart test procedure, 2-BFN-RTP-064A includes system 64D equipment along with a number of other plant systems which were not available for testing. Completion of the RTP is a NPP restart commitment. No deficiencies were noted in the completed SPOC package.

e. Emergency Equipment Cooling Water (System 67)

The SPOC was completed for the EECW on November 13, 1990. A total of 7 exceptions were taken to the completed system. Based on a review of these exceptions, none were determined to affect the system operability. In each case, the work related to the EECW system was completed and the deferrals were for common DCNs which remained open as a result of outstanding work associated with other systems. Since the work efforts were related to multiple systems, the DCNs remained open until the remaining work on the other systems was completed.

In addition, special operating conditions were reviewed which identified certain areas requiring specific valves to remain closed to isolate portions of the system associated with Units 1 and 3 that were not seismically qualified. Further review revealed that the issues identified during walkdowns and each of the applicable CAQRs were addressed.

A brief review of the SPAE was performed and determined to be acceptable. Within the areas reviewed, no adverse conditions were noted.

f. Residual Heat Removal (System 74)

The inspector accompanied licensee personnel during selected portions of the system preliminary and final walkdowns associated with System 74. During the walkdown various material deficiencies such as

missing screws, loose/broken/missing valve handwheels, missing labels, missing/damaged HVAC temperature elements, improper/damaged motor and conduit grounding straps, packing leaks, inability to operate manual valve due to interference from installed pipe support, missing condulet and switchbox cover plates, missing electrical raceway covers and conduit straps, and overall generally poor housekeeping were noted. In addition various incomplete work activities were ongoing.

The system checklist was completed on December 3, 1990. The inspector reviewed the SPOC package with the system engineer on December 4, 1990. After acceptance of the system by the plant staff RHR was placed in system configuration control but not declared operable due to the large amount of uncompleted work orders and DCNs. The portion of the Site Master Punch List related to the RHR system included approximately 300 pages of items. Outstanding work on Unit 2 RHR and the portions of Unit 1 RHR Loop II required to support Unit 2 restart resulted in a total of two exceptions and 19 deferrals being taken as part of the RHR SPOC process. Exception 74-01 dealt with the Unit 1 crosstie capability which will not be available until Unit 2 hydro due to the large amount of ongoing work that must be completed to support that function.

10. NRC Bulletins

a. (CLOSED) Bulletin 260/88-03, Inadequate Latch Engagement In HFA Type Latching Relays.

The inspector reviewed the licensee submittal dated July 12, 1990, which indicated that BFNP completed the inspections of the Unit 2 HFA relays. The inspector also reviewed the licensee's documentation and observed that the HFA relays were removed and calibrated per EMI-100, Replacement of HFA Relay Components and/or Calibration of HFA Relays. Based on these observations, reviews and discussions with licensee representative, the inspector determined that the licensee has adequately addressed this bulletin.

b. (CLOSED) Bulletin 260/88-10, Nonconforming Molded-Case Circuit Breakers.

In IR 90-27, the inspector documented the licensee activities involved with their submittal dated December 15, 1989. The licensee subsequently submitted a revised response dated November 29, 1990. This response indicated that traceability was determined on additional procurement contracts. Various procurement contracts previously listed were outside the scope of the bulletin. Some plant equipment was determined to not contain molded case breakers and all nonconforming breakers installed in all three units were removed and/or replaced.

The inspector observed the plant equipment that indicated no molded case breakers installed, observed breakers in Unit 1 that had been

removed and not replaced, and observed breakers in Units 2 and 3 which had been removed and replaced. Based on this review and previous reviews the inspector concluded that the licensee adequately addressed this item.

11. Temporary Instructions

(CLOSED) TI 259, 260, 296/2515/93, Quality Assurance Request Regarding DG Fuel Oil.

The inspector reviewed the licensee's QA program with regard to DG fuel oil as identified in Multi-Plant Action Item A-15. The inspector noted that the licensee's procedure SDSP 16.11, Bulk Chemistry Control, Attachment B, lists under Specification 3 that Diesel Fuel Oil is a QA Level III Bulk Chemical. The inspector also reviewed licensee records which indicated that the licensee has included DG fuel oil in its QA program.

12. TMI Action Items

(CLOSED) 259, 260, 296/TMI III.D.3.4.3, Control Room Habitability

This item evolved from the Three Mile Incident and addressed a hazard involving habitability in the main control room following a postulated chemical release. The issue was addressed in a previous inspection report, IR 50-259,260,296/90-29, which referenced correspondence between the NRC and Licensee identifying 5 chemicals that were barged past the BFNP on the Tennessee River with no compensatory measures in place. The licensee had previously evaluated the chemicals stored onsite or offsite within a 5-mile radius or transported near the site by barge, rail, or road within the 5-mile radius and determined that only chlorine could present a hazard to personnel in the control room. However, chlorine shipments were below 50 per year as allowed per RG 1.78, Assumptions for Evaluation the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release, and therefore the plant was in compliance with the requirement. The licensee contended that the NRC request to evaluate the a postulated accident involving the remaining 5 chemicals represented a backfit and should not be required.

The NRC staff agreed that the Licensee's assessment met the requirements and found the licensee's completed actions acceptable to close the issue. Based on the letter to Oliver D. Kingsley, November 20, 1990, TVA Backfit Claim Regarding the NRC's Safety Evaluation of the Potential Impact of Hazardous Chemicals Transported By Barges Upon Habitability of the Control Room at Browns Ferry, this item is closed.

13. Reportable Occurrences (92700)

The LERs listed below were reviewed to determine if the information provided met NRC requirements. The determinations included the verification of compliance with TS and regulatory requirements, and

addressed the adequacy of the event description, the corrective actions taken, the existence of potential generic problems, compliance with reporting requirements, and the relative safety significance of each event. Additional in-plant reviews and discussions with plant personnel, as appropriate, were conducted.

a. (CLOSED) LER 296/88-03, Inoperability of Diesel Generators Due to Seismically Unqualified Battery Racks.

This LER is associated with the licensee's August 26, 1988, declaration of inoperability of DGs 3A, 3B, and 3D due to concerns over the seismic qualification of their respective battery racks. The licensee discovered that the front cross-braces were not installed on these racks as required by the vendor drawings when the racks were initially installed in 1980. Condition Adverse to Quality Report 880614 was initiated after the condition was discovered.

The following LERs are listed in LER 296/88-03 as "Previous Similar Events":

259/85-14 - All 8 DG battery racks could not meet seismic qualification requirements due to shims not being installed.

259/85-41 - All 8 DG battery racks could not meet seismic qualification requirements due to hold-down stud material problems and a missed surveillance.

259/85-49 - All 8 DG battery racks could not meet seismic qualification requirements due to hold-down stud material problems.

From the findings addressed in LER 296/88-03 it is apparent that the licensee's earlier investigations associated with the above referenced similar LERs had fallen short of fully identifying the complete scope of battery rack seismic qualification irregularities involving discrepancies between the installed field conditions and vendor drawings.

Although LER 296/88-03 represented a fourth example of seismic qualification problems associated with the "as built" configuration of the diesel generator battery racks, the "Determination of QA Programmatic Deficiency" performed on September 2, 1988 as part of the processing requirements for CAQR 880614 concluded that the CAQR did not represent a QA programmatic deficiency. Subsequent to the visual inspection discussed in the next paragraph, a QA programmatic deficiency was declared in CAQR 880924 due to the eight discrepancies found.

In LER 296/88-03 the licensee committed to visually inspect all DG and main battery racks by November 1, 1988 to verify that all rack

components required by the vendor drawings are installed. During this activity additional discrepancies were identified by the licensee and resulted in the initiation of CAQR 880924 on October 31, 1988. This CAQR contained a November 3, 1988 determination that the CAQR documents a QA Programmatic Deficiency and that the seismic calculations did not consider cross braces which are configured in the plant.

In DNE Calculation CD-Q0999-886564 approved November 4, 1988, the licensee concluded that the discrepancies documented in CAQR 880924 had no adverse impact on the seismic qualification of the battery racks.

Root cause of the seismic qualification problems addressed in the subject LER and its associated CAQRs was ascribed to failure to follow vendor drawings as a result of work processes in effect at the time which did not require vendor drawings to be included in the installation work packages. Plant procedure SDSP-8.4, "Modification Workplans" has since been revised to require all work plans to contain marked-up drawings to be utilized and an associated drawing list which references all drawings required to implement and/or inspect the modification.

As corrective action for LER 296/88-03 and its CAQR 880614 precursor, DCN H2042A was approved on September 2, 1988, to provide replacement specifications for the cross braces and mounting hardware. Field installation of these braces was visually verified on December 5, 1990 by the inspector for Batteries 3A and 3B. The original 3D battery and associated rack addressed in LER 296/88-03 was observed to have been replaced by a new battery/rack assembly.

Based on the inspector's review of the event in association with the above related issues and referenced documents no further followup action is required.

b. (CLOSED) LER 259/89-08, Design of Primary Containment Hydrogen and Oxygen Analyzers Does Not Meet Single Failure Criteria

This LER was associated with the licensee's determination that the design of the primary containment hydrogen and oxygen analyzers cooling water discharge did not meet single failure criteria and was therefore outside the design basis for the plant. For each Browns Ferry unit the cooling water discharge lines from the hydrogen and oxygen analyzers connect to a common EECW discharge header containing a single downstream check valve. Failure of this valve in the closed position would deprive both trains of the hydrogen and oxygen analyzers, for the respective unit, of the cooling water necessary to perform their function. There are two discharge paths for each unit.

Although the units had operated in this configuration prior to the 1985 shutdown, at the time of the discovery of this condition with

Units 1 and 3 defueled and Unit 2 in Cold Shutdown, operability of the analyzer system was not required by the TS.

Condition Adverse to Quality Report #BFP890344, where the licensee documented the initial findings, indicates that drawing 47W859-1 R18 shows that the hydrogen and oxygen analyzers and associated EECW piping were installed under ECN L2079. It appears that although the intent of ECN L2079 was to make analyzers redundant, redundancy was only considered for the monitoring function of detecting hydrogen and oxygen concentrations in the drywell. Redundancy of the EECW side had not been considered.

The deficiencies noted in the LER were corrected via replacement of the EECW check valves with blind spectacle flanges. These check valves were originally installed to provide a secondary containment boundary for the EECW discharge headers exiting the reactor building. The requirement for these check valves to provide a secondary containment boundary was removed via Change Request to Licensing Document CRLD No. BFEP-NTB-89001 RO and supported by Safety Evaluation No. SEBFSAR890100 RO, "Change to FSAR Section 5.3.3.5".

The following table provides the numbers of the design documents that implemented the corrective action:

Unit	Check Valve Replaced	DCN
1	1-67-556	H5120B
2	2-67-556	W6260A
3	3-67-656	H5122A

On November 16, 1990, the inspector visually verified that all three blind spectacle flanges had been installed and were in place in each of their respective RHR Service Water Tunnel locations.

Based on the inspector's review of the event in association with the above related issues and referenced documents, the actions were adequate and were implemented in a timely manner.

c. (CLOSED) LER 260/89-12, Failure to Meet Technical Specifications Because of Miscommunications of Special Requirements of an Electrical Alignment.

On April 4, 1989, TS 3.5.A.5 and 3.5.B.9 associated with the RHR System were not met while Unit 2 had fuel in the reactor because special requirements for an abnormal electrical alignment were not adequately communicated to shift operations personnel. The event resulted when, during an electrical alignment not normally used, both loops of RHR were rendered inoperable at the same time. Special requirements which would have provided continuity of necessary power supplies had been discussed prior to the scheduled electrical outage and a draft of the formal memorandum detailing the special

requirements was prepared. However, actual written guidance was not provided to shift operations personnel prior to the activity. This resulted in the 4160 Volt Shutdown Board B being inoperable during an ongoing outage associated with the A Diesel Generator. When the written instructions were made available to the control room, operations personnel immediately restored the electrical lineup and declared a loop of RHR operable.

The inspector reviewed the LER and verified that it met the reporting requirements of 10 CFR 50.73. Subsequent to the above event, the licensee issued SDSP 12.11, Special Requirements and Compensatory Measures. The inspector reviewed section 7.1.8 and Attachment B to SDSP 12.11 and determined that it provides adequate controls for controlling special requirements and compensatory measures. Additionally the inspector was informed by the licensee that licensed personnel have been instructed not to enter abnormal electrical alignments to support scheduled electrical outages without written instructions. The inspector determined that the licensee's corrective actions have been adequate to preclude recurrence of this event.

d. CLOSED LER 260/90-05, Deenergization of RPS due to MG Set Operation Caused by the Motor Overload Relay Misoperation.

On October 2, 1990, the Unit 2 RPS bus deenergized when the associated MG set tripped. Investigation revealed no failed components but the phase A motor overload relay was found to have an abnormally high resistance. This high resistance caused a MG motor starter coil control relay to drop out and trip the MG set.

The inspector reviewed the LER and verified that it met the requirements of 10 CFR 50.73. The inspector noted that all ESF functions which were not tagged out operated as designed upon the RPS power loss. The licensee cleaned the motor overload relay contacts and tested the system to ensure that no other damage had occurred. No similar events of this failure type have occurred within the last two years. The root cause was attributed to random component failure.

e. (CLOSED) LER 259/90-12, High Pressure Fire Protection System in Violation of Technical Specifications Because Functional Test Not Performed.

During an August 1, 1990 review of the fire protection technical specifications the licensee discovered that the HPFP System fire pump start logic pressure switch 0-PS-26-44 was being calibrated at 100 psig and therefore did not meet the requirements of TS 4.11.B.1.f(4) which had become effective via Amendments 162, 159 and 133 to the Browns Ferry Plants' TS issued by the NRC on December 27, 1988.

This new TS required verifying that after initial high-pressure fire pump actuation each subsequent high-pressure fire pump starts sequentially to maintain the HPFP System pressure greater than or equal to 120 psig.

Further investigation by the licensee disclosed that the HPFP system fire pump start logic was not being functionally tested to verify that the fire protection system pumps maintain 120 psig after an initial pump startup.

The inadequacies in the Fire Protection System Surveillance Instructions that resulted in this reportable event were cited as Violation 50-259,260,296/90-25-01.

In their August 30, 1990 report of the event, the licensee indicated that an investigation would be performed to determine the root cause of the event and to establish corrective actions. A supplemental report detailing the outcome of the investigation was issued on October 15, 1990 as Revision 1 to the original LER.

The licensee documented the results of their root cause analysis and incident investigation efforts in Final Event Report No. II-B-90-085. This report identifies other potential problem areas found during the investigation and establishes various corrective action items each of which has been assigned a Responsible Section for implementation and a completion date.

The root cause of the event was identified as personnel error during the initial preparation of SI 0-SI-4.11.B.1.f, "Simulated Automatic and Manual Actuation of the High Pressure Fire Pump System". The purpose of this procedure is to verify TS 4.11.B.1.f. Revision 0 of this procedure was issued on January 25, 1989, without incorporating the new requirements of the above referenced TS Amendments.

The inspector confirmed that subsequent to the LER event date, the following procedural changes were implemented:

- Revision 5 to Surveillance Instruction 0-SI-4.11.B.1.f added a Pressure Switch Setpoint Calibration Verification to the instruction.

- Non intent change NIC-08 to 0-SI-4.11.B.1.f clarified which procedure calibrates Pressure Switch 0-PS-26-44.

- Revision 9 to procedure 0-SIMI-26B, "High Pressure Fire Protection Scaling and Setpoint Documents", changed the setpoint for 0-PS-26-44 incorporating the 120 psig figure.

- Revision 10 to procedure 0-SIMI-26C, "High Pressure Fire Protection Calibration Data Sheets", also changed the setpoint of 0-PS-26-44 incorporating the 120 psig figure.

Also, Pressure Switch 0-PS-26-44 was recalibrated to the new setpoint in 0-SIMI-26B/C on August 21, 1990 via Work Order #90-11585-00.

Based on an overall review of procedure 0-SI-4.11.B.1.f (Rev.6) the inspector identified that the procedure, as written, did not accomplish the purpose stated in its Section 1.1. This was communicated to the licensee during a December 3, 1990 meeting with responsible licensee system engineers. During this meeting the licensee agreed to review the current wording in the procedure and to make the necessary changes that would provide a clear and accurate statement of how this surveillance instruction meets the specific requirements of TS 4.11.B.1.f(4). Completion of licensee action on this issue will be tracked as part of the followup and closure of item VIO 259,260,296/90-25-01, "Inadequate Fire Protection Surveillance Instruction".

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

a. (CLOSED) IFI 260/89-06-07, Reactor Vessel Level Setpoint.

This IFI concerned the new setpoint and scaling calculations that were required to support proposed new setpoints associated with RPV water level instruments 2-LT-3-203A, 2-LT-3-203B, 2-LT-3-203C, and 2-LT-3-203D. These instruments provide redundant channels for Reactor Building and PCIS isolations and SBGT actuation. The licensee had committed to resolving this issue prior to Unit 2 restart.

During a review of the licensee's closure documentation associated with this item as documented in IR 259, 260, 296/90-25, the inspector identified a concern associated with the licensee's essential calculation program. The inspector noted that the existing revisions (Revision 4) to the four calculations ED-Q2003-880177, ED-Q2003-880178, ED-Q2003-880179, and ED-Q2003-880180 which were to support a new setpoint of 539 inches for each of these instrument channels did not include an adequate margin of error. The setpoint selected by the licensee was 539 inches which complies with the TS requirement, but each of the calculations did not support closure of the open item since the calculated "allowed value" in each case was less than 538 inches. All four of the calculations used the same setpoint methodology and due to conditions unique to the individual instruments resulted in a different resulting value of PV3. PV3 was defined as the calculated allowable value and varied among the four calculations from 537.8 to 537.9. Since these values of PV3 included only those margins based on normal operating conditions and not accident conditions the calculated allowable value would be that value that the instrument channel could be expected to reach prior to periodic functional testing and calibration.

The inspector reviewed new revisions (Revision 5) to these calculations. The inspector determined that the new calculations supported a new setpoint of 539.2 inches above vessel zero. This corresponded to a new calculated value of PV3 of 538 inches above vessel zero for each channel of instrumentation. Since this new value does not conflict with the TS allowable value the inspector determined that the licensee's actions have been adequate to resolve the concern.

b. (CLOSED) VIO 260/89-10-01, Apparent Failure to Comply with Technical Specifications (TS) 3.5.A.5 During Unit 2 Core Reload.

This violation was identified during a special reactive inspection performed on Feb. 20 - March 22, 1989. Contrary to the requirements of the technical specifications, the licensee proceeded with the Unit 2 core reload with the Core Spray System and the motor operated valves in the Unit 2 Standby Coolant Supply flowpath inoperable due to the presence of nonseismically qualified vitrified clay piping in the EECW discharge flowpath. The clay piping which was present in three separate EECW discharge flowpaths was not identified on a CAQR until February 3, 1989, even though licensee Nuclear Engineering personnel had knowledge of the condition as early as January 11, 1989. Plant operations was not made aware of the condition until issuance of the CAQR.

The inspector reviewed the licensee's responses to the violation dated June 14, 1989, and August 28, 1989, along with TVA's letter providing additional information concerning the technical resolution of the clay pipe issue dated June 5, 1989. In those responses the licensee attributed the failure to a lack of sensitivity among NE personnel regarding Browns Ferry becoming operational, TS, and the necessity of timely problems identification and documentation at an operating plant. The TVA letter providing additional information provided the licensee's technical resolution for the problem including proposed modifications.

The portions of the licensee's corrective actions related to the required sensitivity training for NE personnel and other site personnel was verified by an inspector as part of the followup associated with related violation 260/89-10-02 and documented in IR 90-25. During that review the inspector determined that those corrective actions were adequate to prevent recurrence. Additionally the inspector reviewed the licensee's completed modification work and determined that it provides an acceptable design which corrects the original problem. Based on these reviews the inspector determined that the licensee's corrective actions have been adequate to address this violation.

No violations or deviations were identified during the Followup of Open Inspection Items.

15. Resolution of Open Items In Inspection Report 50-260/89-44

a. (CLOSED) Unresolved Item EMG-005, Emergency Condition Allowables

IR 89-15 identified a concern with TVA's use of an allowable limit of twice the manufacturer's normal allowable value for standard component supports in the emergency and faulted load combinations. TVA was requested to provide calculations to demonstrate that these allowable limits for standard component supports did not exceed the basic allowable limit of .9Sy used for structural steel evaluations. TVA provided calculation CD-Q0999-890366 to address the concern with the allowable limits for standard component supports. The results of the NRC's inspection team review of this calculation were documented in IR 89-44. Based on the results of calculation CD-Q0999-890366, TVA revised the allowable limits for some standard component supports and incorporated these revised allowable limits in design criteria document BFN-50-C-7107, Revision 3, October 16, 1989. In IR 89-44 it was stated that TVA's corrective actions adequately addressed the original concern with the allowable limits used for the emergency and faulted loads combinations in the design criteria. However, the review of calculation CD-1067-892499, Revision 0, August 18, 1989, found that the faulted limit used in the calculation exceeded the allowable limit in the design criteria. Unresolved Item EMG-005 remained open pending the completion of TVA's review of all standard component support calculations to confirm that the allowable loads used in the calculations are consistent with the allowable loads in the design criteria.

TVA's March 16, 1990, response to IR 89-44 stated that only calculations performed by Bechtel have listed separate Level D allowables for the faulted condition and that all Bechtel calculations would be reviewed and revised as necessary to be consistent with the TVA criteria. This review was documented in calculation CD-Q0000-900398. The calculation identified 31 Bechtel calculations that required revision or further evaluation. The calculation also documented the results of the review of the TVA prepared pipe support calculations.

b. (CLOSED) Unresolved Item EMG-013, Code Consistency

IR 89-15 identified a concern with TVA's use of the equations from the 1971 edition of Section III of the ASME Code through the Summer 1973 Addenda for piping in combination with criteria from the USAS B31.1.0-1967 piping code. TVA was requested to provide an ASME Code NA-1140 evaluation of this combination to justify the acceptability of the criteria. IR 89-44 documented the review of this evaluation which was provided in TVA calculation CD-Q0999-894489. This review identified a concern with TVA's potential use of furnace-welded piping. TVA was requested to verify that furnace-welded pipe has not been used in safety-related applications or to address its qualification if it has. Unresolved Item EMG-013 remained open pending TVA's response to this request.

TVA's March 16, 1990, response to IR 89-44 identified four systems that contained furnace-welded piping. In addition, TVA identified that furnace-welded piping was used in pipe support stanchions. TVA determined that the use of furnace-welded pipe was a condition adverse to quality and wrote CAQR-BFP890758 to track the corrective actions. TVA identified that furnace-welded ASTM A53 Type F carbon steel pipe 4 inch and under had been used for construction in the diesel generator starting system, the fuel oil system, the carbon dioxide system and the reactor building closed cooling water system. TVA stated that the evaluation of the furnace-welded pipe in these systems found that they met the interim criteria of BFN-50-C-7303 and that some post restart modifications may be required to meet the long term criteria of BFN-50-C-7103.

CAQR-BFP890758 identified two issues with furnace-welded piping. The first issue was that the proper joint efficiency factor as required by USAS B31.1.0-1967 had not been used in the calculation of allowable stress values for the four piping systems identified above. The second issue involved the acceptability of the use of furnace-welded pipe in the fuel oil system based on the code requirement that furnace-welded pipe shall not be used for flammable fluids. In addition, the CAQR identified that the furnace-welded pipe was sometimes used as structural element in pipe supports.

In order to address the second issue, TVA presented an argument, based on Department of Transportation Hazardous Materials Definitions from 49 CFR 173.115, that the fuel oil for the diesel generators would not be considered a flammable liquid. In addition, TVA has submitted a proposed change to the Browns Ferry Technical Specifications, Change No. 277, which would implicitly change the classification of portions of the fuel oil transfer system from safety-related to non-safety-related. According to TVA, the acceptance of this change would eliminate the furnace-welded pipe from the safety-related portion of the fuel oil system. TVA also intends to submit a revision to FSAR Section 8.5.3.4 if the technical specification change is approved (TVA change Request to Licensing Document No. BFEP-SW-90001-R00). The proposed change in TS No. 277 is currently being reviewed by the Office of NRR. Pending NRR's acceptance of the proposed technical specification change, no further review of TVA's use of furnace-welded pipe in the fuel oil system is planned.

The use of furnace-welded pipe in the remaining systems was addressed in two contractor reports. These reports addressed the joint efficiency factor and the lower material allowable stresses for both piping and supports. The results of the large bore piping evaluation were transmitted to TVA by a Bechtel letter dated June 8, 1990, (RIMS B22 90 0607 201). The Bechtel letter stated that no piping or support modifications were required as a result of the evaluation. The results of the small bore piping evaluation were transmitted to

TVA by a Stone and Webster letter dated June 15, 1990 (RIMS B22 90 615 003). The Stone and Webster letter stated that 8 support modifications were required to meet the Browns Ferry long term design criteria. DCN W11720A was written to initiate the modifications.

TVA's proposed change in the TS involving the change of classification of portions of the fuel oil transfer system will be the subject of a separate licensing review.

c. (CLOSED) Unresolved Item EMG-016, Horizontal Slice Package NI-274-18R

IR 89-15 documented two concerns that were identified during the review of Stone and Webster's evaluation for Horizontal Slice Package N1-274-18R. Based on TVA's subsequent revision of the IE Bulletin 79-14/79-02 program, which eliminated the horizontal slice evaluation, one of the concerns was considered no longer applicable as documented in IR 89-44. The other concern involved an interference between a 4-inch-diameter branch line from the RHR system and a 2-inch-diameter strut. TVA's planned program for addressing potential interferences was discussed in IR 89-44. TVA had identified the specific interference from the RHR system branch line in RFI 12825. However, Unresolved Item EMG-016 remained open pending the receipt and review of a TVA submittal providing additional details describing the interference program and its implementation.

TVA's March 16, 1990, response to IR 89-44 stated that potential interferences were currently being dispositioned by Stone and Webster and Bechtel. TVA described the disposition of piping interferences in a memorandum from J. R. Rupert to J. D. Hutson (RIMS B22 90 0514 005). In the memorandum it was stated that thermal interferences and potential seismic interactions with fragile items would be evaluated in the program. The memorandum further stated that the remaining potential seismic interactions would be evaluated in a post-restart USI A-46 seismic interaction program. The preliminary evaluation of interferences for piping in the IE Bulletin 79-14/79-02 program is contained in a Bechtel report transmitted to TVA by letter dated March 9, 1990. The evaluation of interferences for piping in the small bore program is contained in Stone and Webster report, "Final Report -Piping Clearance Program -Task S095," April 13, 1990. As a result of the small bore program 5 modifications were identified. The large bore program including torus attached piping, attached piping had identified 4 modifications as of May 14, 1990. Based on follow-up discussions with TVA, a total of 8 modifications were required for large bore piping.

d. (CLOSED) Unresolved Item EMG-018, EA Review

IR 89-15 identified a concern that TVA had revised the design criteria documents for piping and pipe supports, BFN-50-C-7103 and BFN-50-C-7107, after TVA's EA had reviewed the previous revisions of these documents as part of its oversight function under the Design

Baseline Verification Program. TVA stated that EA would review the new revisions of the criteria documents. A subsequent TVA reorganization eliminated the EA group, and the organization responsible for performing technical audits was transferred to the technical audit section of the TVA nuclear quality assurance and evaluation group. IR 89-44 stated that this organization planned an audit in January 1990, that would include the piping and pipe support criteria documents. Unresolved Item EMG-018 remained open pending the completion and submittal of the results of this audit.

TVA's March 16, 1990, response to IR 89-44 stated that the review of the changes to the civil general design criteria was included in Audit BFA-90015, completed on January 30, 1990. Audit report BFA-90015 (approved February 28, 1990) did not identify any concerns with the changes to the design criteria.

e. (CLOSED) Unresolved Item EMG-023, Torus Criteria Revision

IR 89-15 identified a concern with changes that had been made to TVA's design criteria for torus-attached piping in Attachment E of design criteria document BFN-50-C-7103. The design criteria for torus-attached piping had previously been reviewed by the staff and its consultants in 1985. IR 89-44 noted that one of the changes had also been identified by the employee concerns program and documented in Employee Concerns Subcategory Report 21800. In the subcategory report, TVA had committed to revise the plant-unique analysis report to incorporate the design criteria change and submit it to the NRC staff. Unresolved Item EMG-023 remained open pending TVA's completion of the employee concerns program commitment and for further staff review of the criteria changes pertaining to torus-attached piping.

TVA stated, in the March 16, 1990, response to IR 89-44, that it would revise design criteria BFN-50-C-7103 to eliminate the provision that allowed stresses to exceed the code allowable limit by 5 percent. This change addressed the concern identified in the employee concerns subcategory report and eliminated the need to revise the plant-unique analysis report. TVA implemented the criteria change in a design input memorandum dated May 10, 1990. In addition, TVA revised the corrective action plan (CATD No. 21804-BFN-01) for the employee concerns subcategory report to require a review of all cases where the 5 percent overstress was used and revise the calculations, as necessary, to demonstrate that the code allowables have been met.

The review of the remaining changes to Attachment E of design criteria document BFN-50-C-7103 did not identify any additional concerns.

f. (CLOSED) Unresolved Item EMG-026, Uncontrolled Source Document Used For ZPA and SAM Loads

IR 89-15 identified a concern that SWEC was using an uncontrolled telecopy as a source of design input for seismic ZPA loads and SAM values. The review of piping stress problem N1-274-1R, Revision 3, found that the analysis had referenced the TVA reactor building seismic analysis for the data instead of the telecopy which was the actual data source. In response to the concern, TVA issued QIR CEBBFN 89027 which specified the ARS and methods of analysis and QIRCEBBFN 89045 which provided the ZPA and SAM data. TVA's corrective actions to provide controlled information for ZPA and SAM input were verified during the follow-up inspection and documented in IR 89-44. However, the review of the latest revisions of pipe stress problems N1-274-1R, N1-285-1R and N1-273-6R found that the correct ARS references were not included in the calculations. During the inspection, TVA verified that the correct ARS had been used in the analyses but the references had not been revised. TVA stated that these calculations had been performed by SWEC and that TVA planned to perform a complete review of all SWEC piping analyses that had not been superseded or reanalyzed. According to TVA, the review effort would involve 14 piping analyses performed by SWEC. Unresolved Item EMG-026 remained open pending receipt from TVA of information describing the analysis attributes to be reviewed, the procedures to be followed and the schedule for the completion of the review.

TVA stated in the March 16, 1990, response to IR 89-44 that a review of the calculations was being performed and that a checklist of items from various TVA audits and reviews had been developed to document the review. According to TVA, 13 of the piping analysis problems were reviewed by SWEC. The SWEC review included a checklist of items identified from past audits of the calculations which contained the ARS as an item to be reviewed. The remaining piping analysis problem was reviewed by the Browns Ferry Project.

g. (CLOSED) Unresolved Item EMG-028, USI A-46 Interface

In IR 89-36, it was stated that the review of small bore pipe stress calculation CD-Q2075-883002 identified that core spray pump 2C nozzle loads had exceeded the allowable limits specified in BFN-RAH-307. TVA had accepted these loads based on TVA calculation CD-Q0999-892719. This calculation justified higher allowable limits for valve accelerations and valve and equipment nozzle loads based, in part, or earthquake experience data used for the resolution of USI A-46. Although Browns Ferry is subject to the resolution of USI A-46, in accordance with NRC staff's Generic Letter 87-02, the appropriateness of the use of earthquake experience based data at the interface with rigorously analyzed piping was questioned. Unresolved Item EMG-028 remained open pending further NRC staff review of this issue.

Revision 1 of calculation CD-Q0999-892719 was approved April 26, 1990 and revision 2 was approved November 27, 1990. These revisions to the calculation clarified that the increased allowables for valve accelerations and equipment nozzle loads are only used on an interim basis until a case by case justification is completed. TVA stated

these items have been verified to meet all requirements in the GIP used for the resolution of USI A-46 in the case by case justifications. According to TVA, there were approximately 90 items justified by this method. TVA also stated that, when actual vendor data for allowables existed, the vendor data was used for the qualification. Revision 1 of the calculation also eliminated the requirements for checking valve nozzle loads. According to TVA, Browns Ferry did not have additional interface requirements to limit the stresses on valve nozzles except for torus-attached piping. Torus-attached piping is outside the scope of the IE Bulletin 79-14/ 79-02 and small bore programs which were covered by calculation CD-Q0999-892719. TVA's implementation of the procedures in calculation CD-Q0999-892719, Revision 2 to evaluate valve accelerations and equipment nozzle loads is considered accept-able as interim criteria for restart. TVA should include these items in the scope of the USI A-46 program implementation. The final verification of the valves and floor-mounted equipment using the GIP in conjunction with the implementation of the USI A-46 resolution is acceptable. However, TVA shall ensure that the final verification of the valves and floor-monitored equipment satisfies all requirements of the final staff-approved version of the GIP.

h. (CLOSED) Unresolved Item EMG-032, Use of Actual Material Properties in Piping Analysis

IR 89-44 identified a concern that TVA had used the measured wall thickness to compute the section modulus for a tee in stress problem N1-274-17R. The measured wall thickness was used at node point number R832 because the code equation 11 allowable stress had been exceeded when the nominal wall thickness was used in the calculation. However, Paragraph 119.7.3 of USAS B31.1-1967 requires that dimensional properties used in the flexibility calculations be based on nominal dimensions. Unresolved Item EMG-032 remained opened pending TVA's revision of the pipe stress calculation using the correct code computation, and TVA's review of additional piping analyses to determine if actual properties instead of nominal properties were used in any other calculations.

TVA stated, in the March 16, 1990 response, that stress problem N1-274-17R had used a conservative value for the thermal anchor movement at the 30 inch branch connection on the RHR suppression pool suction header. TVA stated that by using realistic thermal anchor movements at the 30 inch header connection, piping stress at data point R832 is within the code allowables using the nominal wall thickness of the tee. In addition, TVA stated that a review of 72 additional pipe stress problems found no additional cases where actual pipe properties were used.

i. (CLOSED) Unresolved Item EMG-034, Control of Desktop Procedures

IR 89-44 noted that several discrepancies had been identified between PEGs-001 and -002 and other controlling procedures for piping and support analyses. Although the PEGs contained design input

information and references to criteria documents and procedures, the PEGs were not QA-controlled documents. According to Bechtel, the PEGs were a reference source used by the engineers, that did not require formal QA controls and their use was in accordance with EDPI 1.1-38. TVA committed to review the use of the PEGs during an upcoming QA audit of Bechtel. Unresolved Item EMG-034 remained open pending the completion and submittal of the results of the QA audit .

TVA stated, in the March 16, 1990 response, that the site quality group found that the PEGs contained technical data, and that the use of the PEGs without formal technical review and document control was unacceptable. TVA also stated that a review of the PEGs determined that they did not contain guidance or criteria that deviated from established TVA criteria and that the PEGs were no longer in use. The results of the site quality group review were documented in a memorandum form G.G. Turner to Patrick P. Carrier dated February 5, 1990. The review of the PEGs was documented in a letter from Bechtel to TVA dated February 5, 1990.

j. (CLOSED) Unresolved Item EMG-035, Piping Analysis Deficiencies in Calculations Performed by TVA

IR 89-44 identified that TVA technical audit BFN-CEB-89-05 had found numerous deficiencies in piping stress problem N1-167-3RB. The calculation for this stress problem was performed at TVA's offices in Knoxville, Tennessee. As part of the corrective action for the technical audit finding, TVA committed to review all calculations performed at the Knoxville office. This review was still ongoing at the time of the inspection. Unresolved Item EMG-035 remained open pending TVA's submittal of a response that included (1) verification of TVA's completion of the review activities, (2) detailed information regarding the procedures governing the review, and (3) descriptions of any follow-up actions.

TVA stated, in the March 16, 1990 response, that a checklist review based on the technical audit findings was made on the Knoxville stress calculations and the checklist was incorporated in the subsequent revision to these design calculations. According to TVA two problems required reanalysis due to numerous support relocations outside the installation tolerances, and that one also had a boundary condition problem. TVA stated that the remaining stress problems required documentation changes. In addition, TVA stated that the geometry models for two stress analysis problems performed at the site were completely rechecked and found acceptable.

A summary of TVA's review of the Knoxville stress calculations is attached to a memorandum from C. P. Brillante to J. R. Rupert dated January 11, 1990. TVA's review was conducted using a nine page checklist which required a review of the stress problem for each finding from the technical audit.

k. (CLOSED) Unresolved Item EMG-036, Reducing Elbow SIF Valves

IR 89-44 identified a concern with the SIFs used for reducing elbows in pipe stress problem N1-274-IR. TVA's Rigorous Analysis Handbook, Section BFN-RAH-311 required the use of a stress intensification value based on a standard elbow with the dimensions of the larger end. However, reduced SIFs were used on two 6-inch by 4-inch and two 24-inch by 20-inch reducing elbows. These reduced SIFs were based on the results of a finite element analysis of an 8-inch short radius elbow. The reduced SIFs were used at the smaller end of the reducing elbow in order to meet code stress limits. Since TVA did not have calculations for the specific component sizes, the finite element results based on the 8-inch short radius elbow were not considered adequate. Unresolved Item EMG-036 remained open pending receipt, review and approval of TVA's proposed corrective action.

TVA stated, in the March 16, 1990 response, that it had developed a finite element analysis of a 24-inch by 20-inch reducing elbow and was comparing the results with that of the 8-inch short radius elbow. TVA stated that both finite element models showed that the maximum stress intensification decreases toward the ends of the elbows.

The results of the finite element analysis of the 24-inch by 20-inch reducing elbow were presented in a letter from Bechtel dated March 10, 1990. According to the Bechtel letter, the SIF for the small end of the elbow was derived from the finite element analysis by obtaining the ratio of the stress intensity at the small end to the highest stress intensity in the elbow and then multiplying this ratio by the stress intensification factor calculated based on the code equation for a standard long radius elbow. Since the equation for calculating the elbow SIF used in USAS B31.1.0-1967 does not calculate the theoretical maximum stress intensity in an elbow, Bechtel's ratio method was not considered appropriate. TVA was requested to provide a direct comparison of the stress intensity results from the finite element analysis with the results that used the ratioed SIF at the small end of the elbow. This comparison showed that the SIF at the small end of the elbow would be higher than the SIF derived by Bechtel's original method. However, TVA stated that based on a subsequent piping problem reanalysis, the 24-inch by 20-inch reducing elbow met code allowables without using the reduced SIF based on the finite element analysis results. TVA further stated that only the 6-inch by 4-inch reducing elbow required the use of a reduced SIF at the small end. A revised calculation of the SIF, based on a direct comparison with the finite element stress intensity, resulted in a SIF that was 36 percent higher than the value calculated by the original ratio method. However, using the code equation for stress with the moment multiplied by .75i, the results would not change significantly from TVA's original evaluation.

l. (CLOSED) Unresolved Item EMG-037, Gang Hanger Deflection Criteria

IR 89-44 identified that Section 1.4.2.13(e), of design criteria document BFN-50-C-7107, Revision 3, allowed the deflection criteria for gang hanger supports to be satisfied using each pipe load separately. TVA had used this criteria to qualify gang hanger supports. This criteria was not considered adequate because it allowed the use if a different load for checking deflection criteria than was used to calculate member stresses. Unresolved Item EMG-037 remained open pending review and approval of TVA's proposed corrective actions.

TVA stated, in the March 16, 1990 response, that it would revise Section 1.4.2.13(e) of BFN-50-C-7107 to ensure consistency with the section for calculating loads and moments on gang hangers. TVA further stated that 8 supports in the IE Bulletin 79-14/02 program scope required re-evaluation for the revised deflection criteria and that no modifications were required as a result of the evaluation. Supports in the small bore program were being evaluated at the time of response.

The design criteria was revised, by a dated April 6, 1990. The results of the evaluation of gang hangers in the IE Bulletin 79-14/79-02 program scope are contained in calculation CD-Q0000-900259 dated April 9, 1990. The results of the evaluation of gang hanger supports in the small bore program scope are contained in calculation CD-Q2999-900320 dated April 11, 1990. In the calculation for the small bore program scope it was identified that 40 gang hangers used deflection criteria and that 5 of these gang hangers used the criteria for checking deflection with one load at a time. These 5 gang hangers were re-evaluated and found to meet the new criteria provisions. In the calculation it was also noted that for large number of the gang hanger supports the actual stiffness had been used in the piping analyses and, therefore, these gang hangers did not have to meet the deflection criteria.

m. (CLOSED) Unresolved Item EMG-038, Recirculation Piping Time History Analysis

IR 89-44 identified that TVA had used time history analysis to qualify the reactor water recirculation piping. As discussed in the inspection report, the use of time history analysis for piping was subject to a case-by-case review and approval by the staff. This review of the reactor water recirculation piping identified two concerns. The first concern was with the failure to analyze a weld which attaches a pipe whip restraint to the elbow adjacent to node 290, for a specific loading. The second concern involved differences between the piping model and the piping as-built configuration. TVA agreed to address these concerns by correcting the model and reanalyzing the problem. Unresolved Item EMG-038 remained open pending review of TVA's submittal verifying that the reanalysis had been performed.

The staff had additional discussions with TVA with regard to the results of the recirculation piping time history after the inspection. TVA stated that it was preparing additional support data on ALARA considerations and that it was performing an independent review of the results of the recirculation piping time history analysis.

TVA stated in the March 16, 1990, response that the unanalyzed weld mentioned in the first concern had been qualified in calculation CD-Q2068-900042. TVA also stated that it thoroughly reviewed the recirculation model and incorporated the variances identified in the second concern and that the model had been reviewed by two outside consultants. Finally, TVA stated that the recirculation piping was now qualified by the method of analysis. Since TVA has now qualified the recirculation piping using ARS instead of time history, this piping system no longer requires a case-by-case review and approval or additional review of ALARA consideration to justify the use of time history for the qualification.

According to TVA, the new ARS qualification of the recirculation piping model required 12 snubber replacements, 12 support structural modifications and three additional supports to be added. One support was deleted.

16. Regulatory Performance Improvement Program Status

On September 18, 1990, E.G. Wallace, Manager of Nuclear Licensing and Regulatory Affairs wrote a letter requesting closure of Confirmatory Order EA 84-54. The order had been issued July 13, 1984 and required implementation of a Regulatory Performance Improvement Program, periodic status reports to the NRC of RPIP progress, and the management changes outlined by TVA to the NRC be carried out.

The NRC, in a 10 CFR 50.54f letter dated September 17, 1985, requested TVA to supply plans for correcting the root causes of problems at Browns Ferry. The letter indicated the RPIP had not been effective. TVA submitted the NPP in response to the letter. The NPP included in it a statement in Volume 3, Appendix A, Part 1.2, that TVA planned to close out the remaining open RPIP items. In addition, the Appendix stated that the management plan described in Volume 3 was the replacement for the get well program outlined in the RPIP. Appendix A discussed the status of the open RPIP items. NRC IR 88-31, performed in July and September of 1988, closed all the remaining short term RPIP items and eight long term items, leaving eight long term items open. This inspection reviews the status of those remaining eight long term items and closes them, as well well as two other open items not yet closed, but not listed in the NPP or IR 88-31.

a. Item II-1.1, Vendor Manual Control Program

RPIP Long Term Item 1.1 dealt with determination of root causes and implementation of corrective actions for identified open inspection and audit findings and commitments. The final portion left to be

closed under this item was implementation of a Vendor Manual Control Program initiated in part in response to VIO 84-23-02. IFI 84-50-02, Review Program for Revising Manuals and Instructions, also addressed this issue. VIO 85-45-08 closed the tracking of the original violation because the corrective action was to be tracked with closure of this repeat violation. IR 85-37 closed VIO 85-45-08 without detailing any inspection of the Vendor Manual Control Program.

The Maintenance Team Inspection, IR 89-56, closed IFI 84-50-02. The inspectors reviewed the Vendor Manual Control Program at the site. SDSP 10.1, Vendor Manual Control Program was reviewed and items were identified that the procedure did not include. The licensee agreed the items would be included in a future revision and the inspectors closed the IFI based upon the existing program in place at the time of the inspection.

A review of the status of the Vendor Manual Control Program was performed to determine if the RPIP items could be closed based upon the closure of the items described above. The inspector reviewed SDSP 10.1, Revision 5, dated September 27, 1990. Those items identified by the MTI were incorporated. The inspector also reviewed Nuclear Power Standard STD 9.1.5, Revision 0 dated September 28, 1990, Vendor Manual Control, SCNs number 1 and 2 incorporated, and BFEP PI 86-27, Revision 3, dated September 19, 1990, Vendor Manual Technical Review. The inspector interviewed the Manager of Vendor Manuals, Master Equipment List and Equipment Management System and selected members of his staff. The program in place resolves the concern of the open RPIP item. This closes 50-259/IFI 84-SC-58.

b. Item II-2.1, Assign management priority to the Browns Ferry As-Constructed Task Force for resolution of open/incomplete ECNs, workplans, as constructed drawings, and resolve the disapproved but implemented FCR backlog issue.

This long term issue concerned the disagreement between the plant's as built configuration and the drawing's in place in 1984. The problem was due in part to the large backlog of modifications closures preventing documenting closeout as addressed in RPIP II-2.0.

The RPIP item required a management priority to be established to, over the long term, correct the problem. In the period since the plant shutdown the backlog of ECN/DCN closeouts has been reduced dramatically. At the time of this inspection, of the 1639 ECN/DCN closeouts required for startup, only 172 remained. The paperwork closures holding up the drawing revisions have been worked off.

In addition, system walkdowns have been conducted to correct drawing deficiencies, and the final SPOC walkdowns verify the as-built condition matches the primary drawings. An interview with the

Manager of Engineering Drawing Services indicated that non-critical drawing updates should be complete by the end of FY 1992. Many secondary drawings are updated at the time of system SPOC because they are considered critical.

The management priority was assigned to the drawing and modifications backlogs. This RPIP item and 50-259/IFI 84-SC-60 are closed.

c. Items II-3.1, II-3.6, II-3.7

These three items are Action Items under objective 3.0, Ensure division and plant procedures are clear, correct, and complete.

Item 3.1, Rethink and clearly define objectives for issuing and implementing Nuclear Power Area Plan Program Procedures. The objective being to provide maximum support to the plant with minimum administrative burden, addresses the corporate level procedures.

Item 3.6, Require program managers to come onsite and observe activities, review plant instructions affecting their areas, and revise program procedures on the spot. Emphasis on this review must be directed to ensuring regulatory compliance and eliminating unnecessary and counterproductive requirements, addresses the revision of program procedures on site.

Item 3.7, Consolidate and develop integrated plant procedure upgrade effort including standard practices, addresses an integrated effort to attain a uniform result, versus piecemeal effects resulting in inconsistencies.

Since the initiation of RPIP, integrated procedure development efforts for both corporate and site procedures have been ongoing. The original action items of these RPIP items were accomplished, and have been superseded by Volume 3 efforts related to program improvements and procedure upgrades. The Nuclear Program Procedures effort has evolved into development of the Interim Corporate Standards, all of which have been issued. These Interim Standards are being replaced by permanent standards.

Interviews with the Site Procedure Manager indicated that a Standards Improvement Task Force has recently developed a new plan for the corporate standards and by March of 1991, the new standards should all be written. This effort goes beyond the original RPIP task and is a long term improvement in the corporate procedures effort. A new numbering system has been developed and many of the older standards are being combined. Some new standards are being issued for areas not yet covered, but the total number of corporate standards is still being reduced. The different TVA Nuclear Facilities are involved in the corporate standards development. The inspector reviewed documents summarizing their efforts, including a listing correlating the new standards with the old, and team guidelines for writing the new standards.

After Browns Ferry startup the site procedures will be upgraded to reflect the guidance of the corporate standards. The inspector verified procedure upgrades have occurred for all those areas identified in the RPIP. The current procedures in the various areas have been reviewed through programmatic inspection in those areas.

Both the onsite and corporate procedure upgrades are ongoing efforts and will not stop with RPIP or Volume 3 closure. The program review for the various areas is beyond the scope of this inspection.

RPIP items II-3.1, II-3.6, and II-3.7 are closed based upon past actions in these areas. Also closed are IFI 50-259/84-SC-61, IFI 50-259/84-SC-66, and IFI 50-259/84-SC-67 which relate to these RPIP items.

d. Item II-3.5, Revise modification control procedures to assure completion of work, post-modification testing, configuration control, instructions, and training documents before a modification is placed in service.

The inspector interviewed the Modifications Procedures and Maintenance Requests Supervisor. The current modifications process and its controlling documents were discussed. The inspector reviewed the following procedures to ensure they incorporated steps to address the concern in the RPIP:

SDSP-8.4, Modification Workplans, Rev. 21, 7/17/89.
SDSP-12.4, Return to Service and Closure of Modifications, Rev. 13, 10/22/90.
SDSP-17.2, Post Modification Test Program, Rev. 11, 10/2/90.
SDSP-8.10, Plant Review of DCNs, Rev. 15, 7/10/90.
BFEP PI 87-48, Revising and Controlling As-Constructed/Configuration Control Drawings, Rev. 3, 11/15/90.

The procedures required preplanning to control activities concerning work completion, post modification testing, impact evaluations on plant drawings and procedures, and a training review. The procedures also verified work completion, implemented and verified complete the required testing, provided a final as-built verification of the plant drawings, provided the drawing change requests to update them as required, and verified training requirements prior to plant acceptance of the modification. Procedures to cover partial turnovers, test exceptions, deferred testing and punchlist of incomplete items are also included.

Based upon the review of these procedures, RPIP Item II-3.5 for the revision of modification control procedures is complete. IFI 50-259/84-SC-65, RPIP Item II-3.5, is closed.

e. Item II-4.1, Provide for close and continuing interaction between Engineering Design and plant to resolve modification implementation questions and to establish priority and evaluation of nonconformance items and failure evaluations.

This item was not included in the NPP, Volume 3 RPIP items that were open, but is an open item for the NRC. This item was opened to help coordinate activities between the design organization and the onsite modifications group. These interaction problems were evidenced by the large number of open modifications that could not be closed because engineering closure work was outstanding. Coordination was improved by moving the modifications design effort to the site and organizing the modifications functions under a single responsible manager. At the time of this inspection, of the 4768 workplans required for restart, only 29 were requiring approval, and 321 were requiring closeout. The ECN/DCNs remaining for restart were 172 out of 1639. Coordination and prioritization of work allowed the backlog of open engineering items to be worked off.

The inspector, as part of closure of II-3.5, interviewed modifications personnel and reviewed the modifications work flow described in the procedures. The reviewed procedures provided for a responsible engineer for each modification. The procedures describe the actions necessary for changes to modifications, and review of partial turnovers of modifications. The design engineering review is included in this process.

Based upon the procedure review, the workoff of open modifications, and the reorganization, this RPIP item is closed. IFI 50-259/84-SC-68, RPIP II 4.1, is also closed.

f. Item II-6.2, Create joint task force consisting of Purchasing/Stores/Nuclear Power to investigate, study, and recommend corrective actions in procurement process, utilize findings of Ad Hoc Committee.

This item is part of objective number 6.0, Streamline the Procurement Process. Since the RPIP was implemented, BFNP has overhauled its procurement process. IR 90-36, performed November 5-9, 1990, details an inspection of the new warehouse and facilities. The inspector sampled procurement packages to evaluate implementation of the program. Based upon the results of that inspection, this RPIP item and IFI 50-259/84-SC-74, RPIP II-6.2, are closed.

g. Item-II-7.0, Establish an onsite training organization under the supervision of the Training Branch with responsibility for onsite training activities, work toward INPO accreditation of BFNP training program.

This item was not described in the RPIP portion of the NPP, Volume 3, but is not yet closed by the NRC. The TVA Training Organization has been established since the RPIP was issued. The inspector reviewed correspondence in which INPO accredited all ten training programs at Browns Ferry. This item and IFI 50-259/84-SC-75, RPIP II-7.0, are closed.

h. Item II-9.6, Utilize outside contractor (GE) to:

a. Evaluate NSSS operation. This includes normal operation surveillance procedures and shift activities.

b. Evaluate plant trip history and present test methods to assure causes of plant trips have been correctly identified and corrective action taken.

c. Evaluate outage scope and duration.

General Electric as part of their review issued 22 reports containing recommendations for improvements to systems. IR 89-16 closed the commitment made in the NPP concerning implementation of contractor recommendations. The GE recommendations were part of this closure. This RPIP item and IFI 50-259/84-SC-83 are closed based upon IR 89-16.

17. Exit Interview (30703)

The inspection scope and findings were summarized on December 18, 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 Number	Description and Reference
260/90-37-01	URI, Problems Encountered During SLC Surveillance, paragraph 2.

Licensee management was informed that 5 LERs, 1 IFI, 1 VIO, 2 Bulletins, 1 TI, 1 TMI Action Item, the remaining RPIP Items, and 13 of 14 open items for IR 89-44 were closed.

17. Acronyms

ARS	Amplified Response Spectra
ASME	American Society of Mechanical Engineers
ASOS	Assistant Shift Operations Supervisor
ASTM	American Society For Testing and Materials
AUO	Auxiliary Unit Operator
BFNP	Browns Ferry Nuclear Plant
CAQR	Condition Adverse to Quality

CCW	Condenser Circulating Water
CFR	Code of Federal Regulations
CO2	Carbon Dioxide
CRLD	Change Request to Licensing Document
DCN	Design Change Notice
DG	Diesel Generator
DNE	Division of Nuclear Engineering
EA	Enforcement Action
EA	Engineering Assurance
ECN	Engineering Change Notice
EDPI	Engineering Department Project Instruction
EECW	Emergency Equipment Cooling Water
EMI	Electrical Maintenance Instruction
EQ	Environmental Qualification
ESF	Engineered Safety Feature
FSAR	Final Safety Analysis Report
FY	Fiscal Year
GE	General Electric
GIP	Generic Implementation Procedure
GOI	General Operating Instruction
HPFP	High Pressure Fire Protection
HVAC	Heating, Ventilation, & Air Conditioning
IFI	Inspector Followup Item
INPO	Institute of Nuclear Power Operations
IR	Inspection Report
KV	Kilovolt
LCO	Limiting Condition for Operation
LER	Licensee Event Report
LOCA	Loss of Coolant Accident
LT	Level Transmitter
MG	Motor Generator
MIC	Microbiological Induced Corrosion
MOV	Motor Operated Valve
MTI	Maintenance Team Inspection
NE	Nuclear Engineering
NEP	Nuclear Engineering Procedure
NIC	Non Intent Change
NOV	Notice of Violation
NPP	Nuclear Performance Plan
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
NSSS	Nuclear Steam Supply System
OI	Operating Instruction
PCIS	Primary Containment Isolation System
PEG	Project Engineering Guidelines
PMT	Post Modification Testing
PSIG	Pounds Per Square Inch Gauge
PS	Pressure Switch
QA	Quality Assurance
QC	Quality Control
QDCN	Quality Design Change Notice

REV	Revision
RFI	Request For Information
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
RMOV	Reactor Motor Operated Valve
RPIP	Regulatory Performance Improvement Program
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RTP	Restart Test Program
SAM	Seismic Anchor Motion
SBGT	Standby Gas Treatment System
SC	Section Chief
SDSP	Site Directors Standard Practice
SI	Surveillance Instruction
SIF	Stress Intensi-Fication Factors
SLC	Standby Liquid Control
SOV	Solonoid Operated Valve
SPAE	System Plant Acceptance Evaluation
SPOC	System Pre-Operability Checklist
SWEC	Stone & Webster Engineering Corporation
TARS	Transient Analysis Recorder Systems
TD	Test Deficiency
TI	Temporary Instruction
TS	Technical Specification
TS	Temperature Switch
TVA	Tennessee Valley Authority
URI	Unresolved Item
USAS	USA Standards
USI	Unresolved Safety Issue
VIO	Violation
WO	Work Order
WP	Work Plan
WR	Work Request
ZPA	Zero Period Acceleration