UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
61 FORSYTH STREET, SW, SUITE 23T85
ATLANTA, GEORGIA 30303-8931

May 15, 2006

Tennessee Valley Authority
ATTN: Mr. K. W. Singer
Chief Nuclear Officer and
Executive Vice President
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: BROWNS FERRY NUCLEAR PLANT UNIT 1 RECOVERY - NRC INTEGRATED INSPECTION REPORT 05000259/2006006

Dear Mr. Singer:

On April 15, 2006, the U.S. Nuclear Regulatory Commission (NRC) completed a quarterly inspection period associated with recovery activities at your Browns Ferry 1 reactor facility. The enclosed integrated inspection report documents the inspection results, which were discussed on May 3, 2006, with Mr. Masoud Bajestani and other members of your staff.

We previously informed you, in a letter dated December 29, 2004, of the transition of four Reactor Oversight Process (ROP) Cornerstones (Occupational Radiation Safety, Public Radiation Safety, Emergency Preparedness, and Physical Protection) to be monitored under the ROP baseline inspection program. Consequently, as of January 2005, inspections for these cornerstones are integrated with Unit 2 and 3 ROP baseline inspections and Integrated Quarterly Reports. They will no longer be documented in the Unit 1 Recovery Quarterly Integrated Reports such as this one. Inspection Report 05000259,260,296/2006002, issued April 28, 2006, is the most recent Unit 2 and 3 Integrated Quarterly Report which contains Unit 1 ROP inspection of this type.

This inspection examined activities conducted under your Unit 1 license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license and also with fulfillment of Unit 1 Regulatory Framework Commitments. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel. Overall, we primarily found only minor discrepancies, indicating that your oversight of recovery activities was generally effective. However, we will continue to monitor implementation of your corrective actions to address implementation deficiencies associated with installation of Environmentally Qualified cable splices and the system return to service process. Additional inspections will be required to determine the adequacy of this Special Program and your process for system turnover to the operating organization is being implemented satisfactorily.

Based on current and previous inspections of Unit 1 Recovery activities associated with four of your Special Programs, the staff has concluded that your implementation of these Special Programs has been adequate and when fully implemented should satisfy NRC regulatory requirements and commitments in your regulatory framework letter dated December 13, 2002. These Special Programs include the areas of Miscellaneous Steel Frames, Platform Thermal Growth, Control Rod Drive Insert and Withdrawal Piping, and Flexible Conduit. We do not anticipate additional inspections for these areas.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Malcolm T. Widmann, Chief
Reactor Projects Branch 6
Division of Reactor Projects

Docket No. 50-259
License No. DPR-33

Enclosure: Inspection Report 05000259/2006006
w/Attachment: Supplemental Information

cc w/encl: (See page 3)

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Distribution w/encl: (Se page 4)

Report to Karl W. Singer from Malcolm T. Widmann dated May 15, 2006.

SUBJECT: BROWNS FERRY NUCLEAR PLANT UNIT 1 RECOVERY - NRC INTEGRATED INSPECTION REPORT 05000259/2006006

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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket No:	50-259
License No:	DPR-33
Report No:	05000259/2006006
Licensee:	Tennessee Valley Authority (TVA)
Facility:	Browns Ferry Nuclear Plant, Unit 1
Location:	Corner of Shaw and Nuclear Plant Roads Athens, AL 35611
Dates:	January 15, 2005 - April 15, 2006
Inspectors:	W. Bearden, Senior Resident Inspector, Unit 1 E. Christnot, Resident Inspector C. Stancil, Resident Inspector J. Lenahan, Senior Reactor Inspector (Sections E1.6, E1.7, E1.8, E1.9, E1.10, E1.11, E8.7) R. Hamilton, Health Physicist (Sections E8.3, E8.4) N. Staples, Reactor Inspector (Sections E1.4, E1.5, E8.12) R. Chou, Reactor Inspector (Section E1.12) S. Vias, Senior Reactor Inspector (Section E8.6) J. Fuller, Reactor Inspector (Section E8.5)
Approved by:	Malcolm T. Widmann, Chief Reactor Project Branch 6 Division of Reactor Projects

Enclosure

EXECUTIVE SUMMARY

Browns Ferry Nuclear Plant, Unit 1
NRC Inspection Report 05000259/2006006

This integrated inspection included aspects of licensee engineering and modification activities associated with the Unit 1 recovery project. This report covered a three month period of resident inspector inspection. In addition, NRC staff inspectors from the regional office conducted inspections of Unit 1 Recovery Special Programs in the areas of electrical cable installation/separation; flexible conduit; control rod drive insert and withdrawal piping; large bore pipe and supports; long term torus integrity; platform thermal growth; miscellaneous steel frames; containment coatings; small bore piping and instrument tubing; and open inspection items. The inspection program for the Unit 1 Restart Program is described in NRC Inspection Manual Chapter 2509. Information regarding the Browns Ferry Unit 1 Recovery and NRC Inspections can be found at http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/bf1-recovery.html. Per the Partial Cornerstone Transition letter from the NRC to TVA dated December 29, 2004, four Reactor Oversight Process (ROP) Cornerstones (Occupational Radiation Safety, Public Radiation Safety, Emergency Preparedness, and Physical Protection) are monitored under the ROP baseline inspection program as of January 2005. Consequently, inspections for these cornerstones are integrated with Unit 2 and 3 ROP baseline inspections and are no longer documented in the Unit 1 recovery quarterly integrated reports such as this one, but in the Unit 2 and 3 Integrated Quarterly Reports.

Inspection Results - Engineering

- The inspector's review of four planned modification design change packages concluded that the design changes were appropriately developed, reviewed, and approved for implementation per procedural requirements. The designs adequately addressed the changes needed to restore Unit 1 to current requirements. (Section E1.1)

- Modification installation activities associated with four permanent plant design changes were observed and found to be performed in accordance with the documented requirements. (Section E1.1)

- Three temporary alterations which affected secondary containment boundary integrity, secondary containment isolation logic, and temporary vendor panels did not cause any significant impacts on the operability of equipment required to support operations of Units 2 and 3. (Section E1.2)

- A significant weakness was identified in the process for turnover of systems to the operating organization. Although only a limited number of risk significant systems have completed the turnover process numerous deficiencies were identified with the recently completed Residual Heat Removal Service Water system turnover which indicated that the ongoing process was not functioning as originally intended. There has been very little involvement by operating plant personnel in these ongoing activities. Additional observation of future system turnover activities will be required to determine adequacy of the licensee's program in this area. (Section E1.3)

- Implementation of restart testing activities continued to be acceptable. Minor deficiencies were identified during performance of testing which did not effect the results of the testing. Licensee processes were effective at identifying problems before components were placed in service. (Section E1.4)

- Based on current and previous reviews, the inspectors determined that implementation of four sub-programs for the Cable Installation Special Program were proceeding in accordance with licensee commitments and regulatory requirements. These sub-programs include use of condulets as pull points for large 600 volt cable, missing conduit bushings, pulling cable through a 90E condulet or mid-run flexible conduit, and brand rex cable installation. Completed actions to address these issues for Unit 1 are consistent with those previously committed to and performed for Units 2 and 3. The inspectors concluded that no issues related to these sub-programs that would negatively impact the restart of Unit 1 were identified as the result of the above reviews. No further inspections are anticipated for these four sub-programs. However, implementation activities associated with the remaining cable installation sub-programs will need further inspections by the NRC to verify corrective actions are in accordance with licensee commitments. (Section E1.5)

- Flexible Conduit special program activities continued to be performed in accordance with documented requirements and licensee commitments. No further inspections of this Special Program are anticipated (Section E1.6)

- The inspectors determined that corrective actions to resolve deficiencies identified in design and construction of miscellaneous structural steel for Unit 1 are adequate and consistent with those previously performed for Units 2 and 3. No further inspections of the Miscellaneous Steel Frames Special Program are anticipated. (Section E1.7)

- The inspectors determined that the licensee had considered thermal loads in their analysis of Unit 1 structural steel platforms, that those platforms had been re-designed to meet current thermal design criteria, and that modifications had been implemented which incorporated the thermal design. No further inspections of the Platform Thermal Growth Special Program are anticipated. (Section E1.8)

- The inspectors determined that modifications for control rod drive hydraulic drive piping supports were implemented in accordance with design requirements. No further inspections of the Control Rod Drive Insert and Withdrawal Piping Special Program are anticipated. (Section E1.9)

- Small Bore Piping and Instrument Tubing activities were performed in accordance with documented requirements. The inspectors determined that the licensee’s program for correction of deficiencies identified in support of small bore piping and instrument tubing complies with the design criteria, commitments to NRC, and NRC requirements. Inspection of small bore piping and instrument tubing installed inside the drywell has been completed. No further inspections of small bore piping and instrument tubing installed inside the drywell are anticipated. However, additional samples of small bore piping and instrument tubing installed in the reactor building (outside the drywell) will need to be inspected prior to closure of this Special Program. (Section E1.10)

- Based on previous reviews the inspectors determined that the licensee's program for repair and inspection of coatings in the torus were consistent with their commitments to the NRC. No further inspections of the Containment Coatings Special Program associated with the torus are anticipated. (Section E1.11)

- The inspectors determined that the licensee's program for inspection of protective coatings in the drywell, and identification and documentation of deficiencies are consistent with their commitments to the NRC. Work orders are being prepared to specify corrective actions. Additional inspections of the Containment Coatings Special Program will be performed to examine work orders and implementation of corrective actions associated with protective coatings in the drywell. (Section E1.11)

- Based on review of pipe stress analysis and pipe support design calculations for piping in the Long Term Torus Integrity Program the inspectors concluded that ongoing activities for this Special Program continued to be acceptable. However, additional review of completed activities will need to occur prior to closure of this Special Program. (Section E1.12)

- Based on a review of pipe support design calculations for piping in the Large Bore Piping and Support Special Program the inspectors concluded that ongoing activities for this Special Program continued to be acceptable. However, additional samples of completed supports will need to be inspected prior to closure of this Special Program. (Section E1.12)

- The licensee had adequately resolved previous concerns about the qualification of the vendor ultrasonic examination process for the jet pump hold down beams and associated bolting. The inspectors determined that the licensee's in-vessel inspection program had satisfied all Boiling Water Reactor Vessel Internals Project requirements, applicable code requirements and licensing commitments. (Section E1.13)

- The licensee's evaluation of a potentially damaged section of High Pressure Coolant Injection piping was acceptable. Non-destructive examination of the affected section of piping did not identify any problems and the piping satisfied ASME code requirements. (Section E7.1)

- Increased management focus and oversight has resulted in fewer documentation errors and an improvement in craft performance associated with installation of electrical cable splices. However, inspectors will continue to evaluate the effectiveness of the licensee's long term corrective actions in this area. (Section E7.2)

Inspection Results - Maintenance

- The maintenance organization continued to provide appropriate and comprehensive repairs to Unit 1 components which did not require design changes to support Unit 1 Restart. Work order packages included sufficient technical guidance to allow personnel to adequately perform the associated work activity. Maintenance personnel and foreman were knowledgeable of applicable requirements and appropriately documented work actually performed, as required by plant procedures (Section M1.1)

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REPORT DETAILS

Summary of Plant Status

Unit 1 has been shut down since March 19, 1985, and has remained in a long-term lay-up condition with the reactor defueled. The licensee initiated Unit 1 recovery activities to return the unit to operational condition following the TVA Board of Directors decision on May 16, 2002. During the current inspection period, reinstallation of plant equipment and structures continued. Recovery activities include ongoing replacement of small bore piping and instrument tubing in the drywell and reactor building; reinstallation of balance-of-plant piping and turbine auxiliary components; installation of small and large bore pipe supports; and installation of new electrical cables, conduits, and conduit supports. The amount of restart testing and system return to service activities increased during this reporting period as the Unit 1 recovery effort continued to transition away from bulk construction work.

II. Engineering

E1 Conduct of Engineering

E1.1 Permanent Plant Modifications (71111.17, 37550, 37551)

a. Inspection Scope

In order to have some oversight of licensee recovery activities not directly limited to specific Unit Restart List Items, the inspectors reviewed planned Design Change Notice (DCN) packages associated with modifications to the Core Spray (CS) System, Residual Heat Removal (RHR) System, and various instrumentation and controls equipment in the drywell and Unit 1 Auxiliary Instrument Room. The inspectors reviewed criteria in licensee procedures Standard Program and Process (SPP)-9.3, Plant Modifications and Engineering Change Control; SPP-7.1, Work Control Process; SPP-8.3, Post-Modification Testing; and SPP-8.1, Conduct of Testing, to verify that risk-significant plant modifications were developed, reviewed, and approved per the licensee's procedure requirements.

The inspectors reviewed and observed ongoing modification activities to various electrical components in the RHR System, 120 VAC distribution, 480 VAC distribution and various electrical switchyard upgrade program activities. The inspectors evaluated the adequacy of the modifications and observed field work to verify that the design basis, licensing basis, and Technical Specification (TS) requirements for the systems had not been degraded as a result of the modifications.

b. Observations and Findings

b.1 DCN Package Review

The inspectors reviewed the following DCNs associated with planned modifications on Unit 1 to verify that the packages contained adequate design information and supporting analyses to allow modifications personnel to properly implement the desired change, update plant documentation, and resolve the identified condition. In addition, the inspectors verified that the planned modifications would not adversely affect the design basis of the system or interfacing systems. Also, the inspectors verified that the planned modifications would not place either of the operating units in an unsafe condition.

DCN 51081

The inspectors reviewed permanent plant modification DCN 51081, Instrumentation and Control (I & C) Equipment - Control Bay, for Main Steam, System 1; Standby Liquid Control, System 63; and Primary Containment, System 64. The intent of this DCN was to implement the modifications recommended for the listed systems in the control bay. The DCN consisted of five stages and included various work activities involving instrumentation and controls equipment. Planned changes included removal and replacement of various cables between control panels 1-9-5, 1-9-15, and 1-9-17; installation of new cables between control panels 1-9-3A, 1-9-18, 1-9-33, 1-9-42, 1-9-81,and 1-9-82; determinatation, reterminatation, labeling, and abandonment of various cables; installation of conduits and conduit supports for the various cables in areas including the Unit 1 Auxiliary Instrument Room; relocation, rewiring, addition of resistors and add zener diodes to various power supplies; and replacement or removal of various relays and instruments such as flow modifiers.

DCN 51166

The inspectors reviewed permanent plant modification DCN 51166, I & C Equipment - Drywell, for Primary Containment, System 64. The intent of this DCN was to implement the modifications recommended for the Primary Containment system instrumentation and control equipment in the Drywell. The DCN was not subdivided into stages, but consisted of eight Work Orders (WO). Planned changes included installation of conduits, junction boxes, and terminal blocks; installation of conduit and junction box supports; installation and termination of new cables; and replacement of limit switches and temperature detecting elements.

DCN 51200

The inspectors reviewed the Unit 1 permanent plant modification DCN 51200, Core Spray (CS) Mechanical - Reactor Building, System 75. The intent of this DCN was to implement the mechanical and electrical modifications recommended for the CS system in the reactor building. The DCN consisted of four stages. Planned changes included replacement of various check valves including 1-CKV-75-606, 1-CKV-75-607, 1-CKV-75-609, and 1-CKV-75-610; installation of upgraded packing for various valves including 1-FCV-75-02, CS Pump 1A suction Division (DIV) I, 1-FCV-75-30 CS Pump 1B suction DIV II, 1-FCV-75-22 CS flow test DIV I, and 1-FCV-75-50 CS flow test DIV II; replacement of 12 inch piping between valves 1-FCV-75-51 and 1-CKV-75-53, reactor vessel injection inboard and outboard isolation valves; removal of existing valve operators and installation of new operators for various valves including pump suction, minimum flow, flow test, and reactor vessel injection; performance of static installation testing for piping and components installed in the reactor building; performance of dynamic testing of selected CS system valves; and performance of modifications on the Primary System Charging (PSC) Head Tank, including addition of drop leg to the CS piping, installation of sediment trap, and installation of small bore pipe supports.

DCN 51222

The inspectors reviewed the Unit 1 permanent plant modification DCN 51222, Residual Heat Removal (RHR) Electrical - Reactor Building, System 74. The intent of this DCN was to implement the electrical modifications recommended for the RHR system in the reactor building. The DCN consisted of three stages. Planned changes included removal, replacement, termination, labeling, and abandonment of various cables; installation of new conduits, reworking old conduits, installation new conduit supports, and reworking old conduit supports; replacement of cable and electrical components associated with various valves including 1-FCV-74-02, RHR Pump 1A shutdown cooling suction DIV I, 1-FCV-74-12 RHR Pump 1A Torus suction DIV I, 1-FCV-74-13 RHR shutdown cooling Pump 1D suction DIV I, 1-FCV-74-57 RHR flow test DIV I, 1-FCV-74-24, RHR Pump 1B Torus suction DIV II, 1-FCV-74-35 RHR Pump 1D Torus suction DIV II, 1-FCV-74-36, RHR Pump 1D shutdown cooling suction DIV II, 1-FCV-74-47, RHR shutdown cooling suction from recirc loop 1A, and 1-FCV-74-66, RHR outboard injection into reirc loop DIV II; and replacement and rewiring various control panel switches.

b.2 Implementation of Permanent Plant Modifications

The inspectors reviewed selected portions of the following ongoing modifications on Unit 1 to verify adequacy of the modifications and observed field work to verify that the design basis, licensing basis, and TS requirements for the systems had not been degraded as a result of the modifications.

DCN 51084

The inspectors reviewed and observed portions of the permanent plant modification activities associated with DCN 51084, 500 KV Switchyard and Main Generator - Control Bay, System 242, Stage 2. The stage involved the trip logic for the main generator. The activities were controlled by modifications WO 03-016900-02. Work activities observed included selected portions of the installation of wiring and relays for a second channel to the main generator trip backup logic in the Unit 1 Auxiliary Instrument Room.

DCN 51090

The inspectors reviewed and observed portions of the permanent plant modification activities associated with DCN 51090, 480V AC Distribution - Control Bay, System 57-4, Stage 80, Stage 82, and Stage 83. This DCN involved the 480V AC distribution for the standby diesel generator Auxiliary Board A, shutdown board battery SB-C, and standby diesel generator Auxiliary Board B. The activities were controlled by WO's 04-720414-51, 04-720414-52, and 04-720414-58.

Work activities observed included selected portions of the installation of qualified control logic cables for the battery and the auxiliary boards.

DCN 51222

The inspectors reviewed and observed portions of the permanent plant modification activities associated with DCN 51222, RHR Electrical - Reactor Building, System 74, Stage 2. The DCN stage also impacted the 4KV Electrical Distribution - Reactor Building, System 57-5, in that the activities were in the shutdown boards. The activities were controlled by WO's 03-000997-33. Work activities observed included selected portions of ongoing work associated with cables for the RHR motor heaters on the 1B and 1D RHR pumps, replacement of contact blocks for hand switch 1-HS-74-28B, and re-tagging of cables with identification labels.

DCN 51214

The inspectors reviewed and observed portions of the permanent plant modification activities associated with DCN 51214, 120V AC Distribution - Control Bay, System 57-2 Stage 2. This DCN involved the cables for the new inverter, 1-INVT-256-02, to be installed in the Unit Preferred power distribution system for Unit 1. The activities were controlled by WOs 03-004725-10 and 03-004725-11. Work activities observed included selected portions of the installation of conduit supports, conduit, new cables, and tagging the new cables with identification labels. The inverter will replace the DC motor-AC motor-fly wheel-AC generator system currently used for the Unit Preferred power distribution system for Unit 1.

c. Conclusions

The inspector's review of modification design packages associated with four DCNs concluded that the design changes were appropriately developed, reviewed, and approved for implementation per procedural requirements. The DCNs adequately addressed the changes needed to restore Unit 1 to current requirements.

Modification activities associated with four ongoing permanent plant modifications were performed in accordance with the documented requirements.

E1.2 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed licensee procedure SPP-9.5, Temporary Alterations. The inspectors also reviewed and observed temporary alterations involved temporarily disabling secondary containment logic, removal of the temporary extended secondary containment boundaries, and verification of prior removal of temporary vendor startup panel in the Main Control Room (MCR). The inspectors reviewed the associated 10 CFR 50.59 screening against the system design bases documentation and reviewed selected completed work activities of the system to verify that installation was consistent with the modification documents and the Temporary Alteration Control Form (TACF). In addition, special emphasis was placed on the potential impact of these temporary modifications on operability of equipment required to support operations of Units 2 and 3.

b. Observations and Findings

The inspectors reviewed and observed selected portions of ongoing activities associated with temporary alterations for Secondary Containment and a temporary MCR panel. The temporary alterations potentially impacted Secondary Containment and involved verification of a previously removed temporary vendor panel and the temporary removal of power to the secondary containment logic to support ongoing outage activities. The inspectors verified that the ongoing temporary modification activities were consistent with the applicable documentation, configuration control of the temporary modification was adequate, post-installation testing confirmed actual impact of the modification on permanent systems and interfacing systems. In addition, the inspectors verified that the activities did not cause an adverse impact on operability of structures, systems, and components (SSCs) required to support operations of Unit 2 and 3. The temporary alterations reviewed and observed were as follows:

- WO 06-710437-00, Technical Evaluation (TE), Rev 1, was initiated to temporarily disable portions of the logic for Secondary Containment Isolation, System 64C. The temporary alteration was required to support the installation of DCN 51081, Primary and Secondary Containment - Electrical, System 64; DCN 51102, CRDR Panel 1-9-25; and DCN 51190, Containment Purge and Reactor Building Ventilation - Electrical, System 64B. The TE specifically justified the short term impact on the Unit 1 Reactor Zone and the Units 1, 2 and 3 Plant Refueling Zone. The work activities were performed in the Unit 1 Main Control Room Panel 1-9-25, and the Unit 1 Auxiliary Instrument Room Panels 1-9-42 and 1-9-43. The inspectors reviewed the WO and TE which allowed for the installation of jumpers affecting two relays, 0-RLY-64-R2A and 0-RLY-64-R2B, Plant Refueling Zone Isolation Logic. These relays were located in the auxiliary instrument room panels and controlled the plant refueling zone ventilation inboard isolation dampers and the outboard isolation dampers, respectively. The inspectors noted that Steps 22 and 23 of the WO momentarily affected Secondary Containment Isolation logic while power was secured, wires to hand switches 0-HS-64-119 and 120, Plant Refueling Zone Isolation Test, were de-terminated, jumpers installed, and power restored. When power was restored the jumpers allowed for the operation of relays 0-RLY-64-R2A and R2B. This restored the Plant Refueling Zone Isolation Logic from Units 2 and 3. The inspectors verified that Step 23 of the WO was performed to re-terminate wiring to the hand switches and restore the isolation logic to the final design configuration.

- Engineering Work Request (EWR) 05CEB001101, Main Steam Extended Secondary Containment Boundary Integrity, Rev 1, was previously issued by the Civil Engineering Branch to provide a method and justification for a temporary modification to the secondary containment by extending the containment boundary beyond the Outboard Main Steam Isolation Valves (MSIV). EWR 05CEB001101 also contained the evaluation of the static and seismic adequacy for the extension of secondary containment to support ongoing refurbishment of the Unit 1 outboard MSIVs. Previous review of this EWR was documented in Inspection Report 259/2005-09. The approved method required the installation of expandable piping plugs in each of the four main steam pipes downstream of the MSIVs. The instructions also included directions that the air pressure to the expandable piping plugs be monitored periodically while installed. The inspectors reviewed and observed ongoing activities associated with WO 05-719158-00 for removal of the expandable main steam piping plugs upon completion of the work activities on the outboard MSIVs. Additionally, the inspectors verified that the outboard MSIVs were closed prior to removal of the plugs.

- TACF 1-86-001-057, Provide Instructions to Install a Seismically Qualified Panel Support, Rev 0, had previously installed seismic supports on panels located in both Main Control Rooms, one for each unit. The panels were originally installed to provide temporary equipment connections to support reactor startup testing. During the Unit 2 recovery and restart it was determined that the panels were no longer required and the Units 1 and 3 panels were to be removed. WO 03-002058-00 was issued to independently verify in the field that the Unit 1 supports and temporary panel had been removed. The inspectors walked down the panel location in the Unit 1 MCR and observed that the supports and the panel had been removed.

c. Conclusions

The inspectors determined that activities associated with three temporary alterations which affected secondary containment boundary integrity, secondary containment isolation logic, and temporary vendor panels did not cause any significant impacts on the operability of equipment required to support operations of Units 2 and 3. No violations or deviations were identified.

E1.3 System Return to Service Activities (37550, 37551)

a. Inspection Scope

The inspectors continued to review and observe portions of the licensee’s ongoing System Return to Service (SRTS) activities. The SRTS activities were performed in accordance with Technical Instruction 1-TI-437, System Return to Service Turnover Process for Unit 1 Restart. The level of SRTS activities continued to increase during this reporting period as the Unit 1 recovery effort started to transition away from bulk construction work. However, only a limited number of important risk significant systems have completed SRTS activities.

Additionally, the inspectors focused on a review of licensee activities associated with the recently completed SRTS process for the Residual Heat Removal Service Water System (RHRSW).

b. Observations and Findings

The SRTS process consisted of three parts: System Plant Acceptance Evaluation (SPAE), which consists of verification of design changes, engineering programs analysis, drawings, calculations, corrective action items, and licensing issues; System Pre-Operability Checklist (SPOC) I, which consists of the completion of items required for system testing; and SPOC II, which consists of the completion of system testing and the completion of items that affect operational readiness.

Activities observed included periodic meetings to discuss the SRTS status, which included the status of the SPAE process, the status of the SPOC I checklists, status of the SPOC II process, status of outstanding work items and identified deficiencies. Additionally, the inspectors focused on SPOC II activities associated with System 23, RHRSW. Documents and activities reviewed included of RHRSW System SPOC exceptions, deferrals, and special operating conditions; system testing requirements; temporary alterations, completed work orders (WOs); engineering calculations; SRTS open items punchlist (OIP); and various PERs associated with the SRTS process. The inspector also held discussions with engineering and operations personnel responsible for SRTS activities and performed walkdowns of selected portions of the RHRSW System.

Insufficient procedure guidance

- BP-323, Browns Ferry Nuclear Plant Organizational Structure, Roles and Responsibilities for Unit 1 Restart does not require early operating plant organizational involvement in the SPOC turnover process. Operating plant system engineer (SE) and operations personnel were not specifically required to attend SPOC meetings, testing, and walkdowns.

- Written guidance is lacking concerning who (restart or plant organizations) and how to document new open operability items following the SPOC II approval date and prior to plant operations declaration of system operability.

- The SRTS Open Item Punchlist (OIP) was being loaded with non-impact open items. Approximately 47 items on the SRTS-OIP were coded 90 for project completion and not required for system operation, operability, testing, or turnover. 1-TI-437 currently defines the SRTS-OIP as “an Integration Task Equipment List (ITEL) generated report specifying open items that affect the ability to test systems/components or to make systems/components operable.”

- Once an open item is coded as 00 by the restart system engineer (RE-SE), 1-TI-437 does not require further peer or cross-disciplinary review, especially by operating plant personnel with integrated plant operability experience (e.g., licensed individuals, plant SE personnel). The current process restricts operating plant licensed operators from seeing open items coded as 00. Although, plant operations is required to review the SRTS-OIP for adverse impact to system operability prior to SPOC II acceptance, items coded 00 do not apply to the SPOC system and will not show on the SRTS-OIP.

- There is a lack of procedural guidance for assuring tracking of Special Operating Conditions (SOCs) as coded open items on the SRTS-OIP, operability impact or not.

Operating Plant Staff Involvement

During the inspectors attendance at periodic SRTS meetings and the review of open items related to RHRSW several deficiencies were identified which were related to poor ownership of SRTS activities by the operating plant staff. These included:

- System RHRSW open items coded as 00 are not getting reviewed by operating plant personnel with integrated plant operability experience. There were approximately 300 for RHRSW. The RE-SE was the only person determining coding of open items assigned during the SPOC process. Subsequently items coded as 00 would not show on the SRTS-OIP for later review. Unit 1 operations support personnel, who previously had held Senior Reactor Operator (SRO) licenses relied heavily on the RE-SE to properly scope open items. The Plant System Engineer (SE) was not involved in reviewing open items coded as 00.

- Plant operating representatives were absent at all SPOC meetings attended by the inspectors. Specifically SE personnel, licensed operators, and maintenance personnel who will receive responsibility to maintain and operate SPOC II systems were not present.

- Discussions with key restart personnel in the SPOC process indicate poor operating plant system engineer involvement in the SPOC turnover process, specifically SPOC meetings, testing, and walkdowns.

Failure to Document Operability Items or Exceptions

Examples of open items that should be exceptions but were not listed as exceptions were identified. These included:

- WOs for installation of double counterweights on RHRSW supply header check valves impacted operability and were not captured as an exception but required followup flow testing.

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- RHRSW header flow orifice replacements and associated instrument loop calibrations impacted operability and were not captured as an exception, but were appropriately system coded 50 (operability impact) on the SRTS-OIP for followup instrument loop calibrations and flow testing.

Inappropriately Coded Open Items

- WOs for installation of double counterweights on RHRSW supply header check valves were identified on the SRTS-OIP but were not coded 50 (for operability impact). One had been coded 90 (does not impact operation, operability, testing, or turnover) and the other coded 40 (programmatic, does not affect operability). These two work orders required followup flow testing.

- WOs for replacement of RHRHX floating head gaskets with a new design, were listed on Exception PL-05-1600 but not on the SRTS-OIP. The gaskets require an ASME Section XI pressure test. The WOs implemented DCN 51199-01 (RHR system 74 DCN) which was on the system 23 SRTS-OIP, however, the DCN was coded 90-MX for no impact to the RHRSW system.

- SOC PL-05-1111 is an operability impact to the RHRSW system and was listed as an exception. However, it was not listed and coded 50 on the SRTS-OIP. The SOC was for interim fire protection Appendix R measures to keep Unit 1 RHRSW flow control valves closed and breakers open.

Documentation Deficiencies

- Although, the licensee originally intended one exception for each A and B RHR Heat Exchangers (HXs) for ease of closure, most written descriptions and listed document identifiers for both exceptions were for both HXs. Therefore, both HXs must be returned to operations (RTO) to close either exception.

- Two surveillances listed on Exception PL-05-1600, 0-SR-3.3.3.2.1(23) and 2-SI-3.2.10.B, should not have been listed as exceptions. The tests were in periodicity and being performed under the operating plant testing program.

- Technical Instruction TI-63, intended for monitoring and trending for flow blockage, did not impact RHRSW system operability, but was on both exceptions.

- Three documents listed on Exception PL-05-1600 had typographical errors resulting in nonexistent procedures.

- The Residual Heat Removal Heat Exchanger (RHRHX) outlet flow control valve (FCV) numbers were transposed for the A and C RHRHXs.

- The Post-Modification Testing Instruction Results Package cover sheet for PMTII-51177-STG05(SYS023) transposed the procedure title and test number with PMTII-51177-STG06(SYS023).

Training

- Unit 1 restart operations support had previously conducted SPOC training to plant operators approximately two years ago with no followup refresher training.

- The RHRSW RE-SE had not received specific SPOC process training, other than reading process procedures and on-the-job performance.

- Operating plant SE personnel were not trained on the SPOC process or use of ITEL.

The licensee was conducting an ongoing self assessment of the SPOC process during this inspection period. The final assessment report had not been issued at the end of the reporting period. However, based on discussions with licensee management and members of the assessment team the inspectors determined that many of the above deficiencies noted by the inspectors were also identified by members of the assessment team. Concerns associated with compliance with documented licensee procedures were not resolved at the end of the report period. Additional NRC review of these deficiencies will need to occur after these deficiencies are documented in the corrective action program and designated as required to be addressed as part of the SRTS process. An Unresolved Item (URI) 50-259/2006-06-01, Adequacy of SRTS Activities, was identified.

c. Conclusions

A significant weakness was identified in the licensee's SRTS process. There has been very little involvement by operating plant personnel in ongoing SRTS activities. Several deficiencies were identified with the recently completed RHRSW SPOC II process which indicated that the ongoing SRTS process is not functioning as originally intended. An unresolved item was identified pending the inspectors review of licensee resolution of these deficiencies. Additionally, observation of future SRTS activities will be required to determine adequacy of the licensee's SRTS process.

E1.4 System Restart Testing Program Activities (37551)

a. Inspection Scope

The inspectors reviewed and observed the ongoing Restart Test Program (RTP) activities associated with post modification testing for four risk significant systems and one Appendix R system to ensure activities were in compliance with design basis requirements. Additionally, the inspectors reviewed RTP activities associated with Base Line Test Requirements Documents (BTRD) testing for two risk significant systems and one supporting system to ensure activities were in compliance with design basis requirements.

b. Observations and Findings

b.1 Post Modification Testing Activities

Post modification testing activities reviewed and observed consisted of post modification testing performed on System 24, Raw Cooling Water (RCW); System 247, Emergency Lighting; System 65, Standby Gas Treatment (SBGT); System 82, Standby Diesel Generators (EDG); and System 57-4, 480V Electrical System - Control Bay.

Test procedures were either existing surveillance instructions or Post Modification Test Instructions (PMTIs) developed, written, approved, and issued to test portions of applicable DCNs. The inspectors verified that pretest briefings were held, assignments made, and communications were established. Specific post modification testing activities reviewed and observed included the following:

1-PMTI-51102-STG08 & 10

This testing satisfied the post modifications test requirements for portions of stages 8 and 10 of DCN 51102, Main Control Room (MCR) Panel 1-9-25. This DCN is part of the Control Room Design Review (CRDR) program for the Reactor Building Ventilation, System 64B, and SBGT, System 65, Train B. These stages consisted of modifications to relocate various hand switches and replace escutcheons for Human Engineering Deficiency (HED) purposes. The DCN also added barriers to provide for electrical separation. The objective of this testing was to demonstrate that the switches such as 0-HS-65-25A, Train B Inlet Damper, and 0-HS-65-40A, Standby Gas Treatment Fan B Blower, performed their design functions. The inspectors observed portions of the test, reviewed the completed test package, and verified acceptance criteria for the test were satisfied. The inspectors determined that the testing successfully fulfilled the testing requirements for portions of work performed under DCN 51102, Stage 8 and Stage 10. There were no test deficiencies.

1-PMTI-51102-STG8

This testing satisfied the post modifications test requirements for portions of stage 8 of DCN 51102, MCR Panel 1-9-25. This DCN is part of the CRDR program for the SBGT System, Train A. This stage consisted of modifications to relocate hand switch 0-HS-65-4A, Standby Gas Treatment Train A Decay Heat Damper, and replace associated escutcheon for HED purposes. This post modification test instruction PMTI could not be performed until 1-PMTI-51102-STG08 & 10 was completed and SBGT Train B was declared operable. This was done in order to prevent two trains of SBGT from being out of service at the same time. The objective of this test was to demonstrate that the switch performed its design function. The inspectors observed portions of the test, reviewed the completed test package, and verified acceptance criteria for the test were satisfied. The inspectors determined that the testing successfully fulfilled the testing requirements for portions of work performed under DCN 51102, Stage 8. There were no test deficiencies.

0-SR-3.8.1.1 (B)

This surveillance test instruction, Diesel Generator B Monthly Operability Test, satisfied the post modifications test requirements for DCN 51217, 4KV Electrical Distribution - Reactor Building, System 57-5, Stage 2. The stage consisted of modifications to the Units 1 and 2 EDG B electrical control network. The objective of this surveillance test was to demonstrate that the EDG B performed its design function affected by the DCN stage. The specific test requirement was to verify that the EDG B started from the standby condition and achieved a steady state voltage of between 3940 V and 4400 V with a frequency of between 58.8 Hz and 61.2 Hz. The inspectors observed portions of the test, reviewed the completed test package, and verified acceptance criteria for the test were satisfied. There were no test deficiencies.

3-SR-3.8.1.1 (3B)

This surveillance test instruction, Diesel Generator 3B Monthly Operability Test, satisfied the post modifications test requirements for DCN 51217, 4KV Electrical Distribution - Reactor Building, System 57-5, Stage 9. The stage consisted of modifications to the Unit 3 EDG 3B electrical control network. The objective of this surveillance test was to demonstrate that the EDG 3B performed its design function affected by the DCN stage. The specific test requirement was to verify that the EDG 3B started from the standby condition and achieved a steady state voltage of between 3940 V and 4400 V with a frequency of between 58.8 Hz and 61.2 Hz. The inspectors observed portions of the test, reviewed the completed test package, and verified acceptance criteria for the test were satisfied. There were no test deficiencies.

1-PMTI-BF-51229-STG09

This testing satisfied the post modifications test requirements for tested Stage 9 of DCN 51229, Appendix R Reactor Building - Electrical. This DCN is part of the emergency lighting program for the fire protection system. The stage consisted of modifications to the 240V emergency lighting, System 247, and fire protection battery pack lighting, System 999-2, located in the Reactor Building. The objective of this test was to demonstrate that adequate emergency lighting was installed for manual operation of RHRSW valves 1-FCV-23-34 and 40. The valves are the control valves for the RHRSW discharge for RHR heat exchangers 1A and 1C respectively. The test also was to verify that adequate emergency lighting was installed for ingress and egress for the valves. The inspectors noted that one acceptance criterion for the test had initially been considered as not satisfied. The luminance in one area of the valves was less the 1 foot-candle of light. PER 93858 was initiated and it was later determined by operations that the luminance was adequate for the ingress and egress. The inspectors observed portions of the test, reviewed the completed test package, and verified acceptance criteria for the test were satisfied. The inspectors determined that the testing fulfilled the testing requirements for work performed under DCN 51229, Stage 9. There were no test deficiencies.

1-PMTI-BF-51118-STG06

This testing satisfied the post modifications test requirements for tested Stage 6 of DCN 51118, RCW, Turbine Building - Mechanical, System 24. The stage consisted of modifications to the automatic start and stop logic for the RCW pumps located in the Turbine Building. The logic is initiated from pressure switches 1-PS-24-7A and 7B. The objective of this test was to demonstrate that the logic performed its design function by initiating a start signal from the switches to the pumps on low RCW header pressure. The inspectors observed portions of the test, reviewed the completed test package, and verified acceptance criteria for the test were satisfied. The inspectors determined that the testing successfully fulfilled the testing requirements for work performed under DCN 51118, Stage 6. There were no test deficiencies.

1-PMTI-BF-51090-STG32

This testing satisfied the post modifications test requirements for tested Stage 32 of DCN 51090, 480V Electrical System - Control Bay, System 57-4. This DCN is part of the load shed program for the emergency diesel generator system. The stage consisted of modifications to Drywell Fan 1B-3 load shed reset time delay relay. The objective of this test was to demonstrate that the reset time delay relay performed its design function. The test requirement was to verify that after actuation the time delay relay would not reset for equal to or greater then 10 seconds. The inspectors observed portions of the test, reviewed the completed test package, and verified acceptance criteria for the test were satisfied. The inspectors determined that the testing successfully fulfilled the testing requirements for work performed under DCN 51090, Stage 32. There were no test deficiencies.

1-PMTI-BF-51118-STG05

This testing satisfied the post modifications test requirements for tested Stage 5 of DCN 51118, Raw Cooling Water (RCW) Turbine Building - Mechanical, System 24. The stage consisted of modifications to pressure differential switch 1-PDSI-24-2, located across the system strainer. The objective of this test was to demonstrate that when the switch activated the associated annunciator 1-XA-55-20A, Window 25, labeled RCW STRAINER DP HIGH, would also illuminate. The inspectors observed portions of the test, reviewed the completed test package, and verified acceptance criteria for the test were satisfied. The inspectors determined that the testing successfully fulfilled the testing requirements for work performed under DCN 51118, Stage 5. There were no test exceptions.

b.2 Restart Testing Activities for Baseline Test Requirements Documents (BTRDs)

The inspectors reviewed and observed the activities associated with the completed BTRD testing involving the Essential Equipment Cooling Water (EECW), System 67, Residual Heat Removal Service Water (RHRSW), System 23, and Sodium Hypochlorite, System 50. The Sodium Hypochlorite System provides chemical treatment for these two risk significant service water systems. Additionally, testing requirements associated with System 50 are included within BTRD testing for System 67. The BTRD tests were performed to demonstrate that the systems were capable of fulfilling the safe shutdown functional requirements for three unit operations as identified in the Safe Shutdown Analysis (SSA).

The BTRD testing was performed in accordance with applicable procedures to demonstrate system configurations associated with the various EECW system and RHRSW system modes would satisfy the SSA. One Post Modification Test Instruction (PMTI), one Technical Instruction (TI), and four surveillances (SI/SR) were performed to satisfy these RTP test requirements. The inspectors observed selected ongoing testing activities and reviewed completed test results to verify that system modes were satisfactorily tested during the ongoing testing activities. Additionally, the inspectors attended and participated in routine restart testing meetings where initial test results and overall testing plans were discussed. System modes tested included the following:

1-BFN-BTRD-067/50 System Mode 67-01

This mode required the EECW System to provide cooling water to various safety-related System 31 air-conditioning chillers, RHR equipment room coolers, Emergency Diesel Generators, RHR pump seal coolers, containment inerting H2 and O2 analyzers, Core Spray equipment room coolers, and emergency water to the fuel pool; provide EECW valve interlock signals for auto-start of the RHRSW pumps; verify that each of the automatic Reactor Building Closed Cooling Water (RBCCW) heat exchanger supply valves from the EECW system, 1-FCV-67-50 and 1-FCV-67-51, operate and fail closed; and verify that the applicable EECW check valves close at the various interfaces between the North and South EECW headers and also at each end of the North and South headers. Testing conducted by the licensee to address this system mode included 1-TI-496, EECW Flow Test; 1-SI-3.2.3, Testing ASME Section XI Check Valves; 3-SI-4.5.C.1(2), EECW Pump Operation; and PMTI-51192-STG4, Functional Test of Unit 1 EECW Flow control Valves 1-FCV-67-50 and 1-FCV-67-51. The inspectors observed selected portions of the ongoing testing. Additionally, the inspectors reviewed completed test results and verified that this system mode was satisfactorily tested during the ongoing testing activities.

1-BFN-BTRD-067/50 System Mode 67-02

This mode required the EECW System to provide a secondary containment boundary. The secondary containment boundary is common to all three units. The inspectors verified that this system mode required no special testing and had been adequately addressed by previous restart testing on Units 2 and 3 along with periodic operating plant surveillance testing to verify secondary containment.

1-BFN-BTRD-067/50 System Mode 67-03

This mode required the EECW System to provide for safe shutdown from outside the Main Control Room (MCR) and provide cooling water to various safety-related System 31 air conditioning chillers, RHR equipment room coolers, Emergency Diesel Generators, RHR pump seal coolers, Core Spray equipment room coolers, and emergency water to the fuel pool; provide EECW valve interlock signals for start of the RHRSW pumps; and verify that the applicable EECW check valves close at the various interfaces between the North and South EECW headers and also at each end of the North and South headers. Testing conducted by the licensee to address this system mode included 1-TI-496, 1-SI-3.2.3, and 3-SI-4.5.C.1(2). The inspectors reviewed completed test results and verified that this system mode was satisfactorily tested during the ongoing testing activities.

1-BFN-BTRD-067/50 System Mode 67-05

This mode required that a fire hose be attached to the EECW system to maintain water level in the fuel pool as described by procedure 1-TI-496. The inspectors verified that this system mode required no special testing and had been adequately addressed by existing fire hoses which serve that purpose for Units 2 and 3.

1-BFN-BTRD-067/50 System Mode 50-01

This mode required pressure boundary integrity for the EECW and RHRSW systems. The inspectors verified that this system mode required no special testing as these systems are shared with the operating units and had been adequately addressed by previous restart testing on Units 2 and 3.

1-BFN-BTRD-023 System Mode 23-03

This mode required for the operation of the RHRSW pumps, dedicated to the EECW system, from the MCR to supply cooling water to the EECW system based on given EECW switch position interlock signals. Testing conducted by the licensee to address this system mode included 0-SR-3.3.3.2.1 (67), Back Up Control Panel Testing, Rev 4; and 1-SI-4.5.C.1 (2), EECW Pump Operation, Rev 4. The inspectors reviewed completed test results and verified that this system mode was satisfactorily tested during the ongoing testing activities.

1-BFN-BTRD-023 System Mode 23-10

This mode required for the operation of the RHRSW pumps, dedicated to the EECW system, from outside the MCR to supply cooling water to the EECW system given EECW switch position interlock signals. Testing conducted by the licensee to address this system mode included 0-SR-3.3.3.2.1 (67). The inspectors reviewed completed test results and verified that this system mode was satisfactorily tested during the ongoing testing activities.

c. Conclusions

Implementation of restart testing activities was acceptable. Only minor deficiencies were identified during performance of testing which did not effect the results of the testing. Licensee processes were effective at identifying problems before components were placed in service.

Based on the above review and observations, the inspectors determined that testing was conducted according to applicable licensee procedures and emergent issues during the testing were adequately addressed by the licensee.

E1.5 Special Program Activities - Cable Installation and Cable Separation (37550, 37551)

a. Inspection Scope

The programs for investigating and resolving the issues of cable installation and cable separation are described in TVA's letter to the NRC dated May 10, 1991. This letter describes programs as essentially the same as described in the Browns Ferry Nuclear Performance Plan which outlined the corrective actions to be implemented before restart of Unit 2, and repeated for restart of Unit 3. NRC Inspection Manual MC2509, Browns Ferry Unit 1 Restart Project Inspection Program, endorses the special programs utilized on Units 2 and 3 as sufficient to address corresponding issues on Unit 1 if implemented in the same manner.

This inspection focused on several sub-programs within the Cable Installation Special Program. These sub-programs included use of condulets as pull points for large 600 volt cable, missing conduit bushings, pulling cable through a 90E condulet or mid-run flexible conduit, and brand rex cable installation.

Additionally, the inspectors continued to observe and/or review ongoing licensee activities associated with the installation of electrical cables. The installation activities were controlled by modification work orders (WOs) and licensee procedures. Among the procedures were the following: Modification and Addition Instruction (MAI) 1.3, General Requirements for Modification; MAI-3.2, Cable Pulling for Insulated Cables Rated Up to 15,000 Volts Units 1, 2, and 3; MAI-3.3, Cable Terminating and Splicing for Cables Rated Up to 15,000 Volts Units 1, 2, and 3; and MAI-3.7, Cable Pull Force Monitoring Breaklink Fabrication, Verification, and Control.

b. Observations and Findings

b.1 Observation of Cable Installation Activities

The licensee continued to perform bulk cable installation activities during this report period. In addition to the bulk activities other cable installation activities were also performed. These additional activities included DCN 51090, Electrical 480V Distribution - Control Bay, System 57-4; DCN 51214, Electrical 120V Distribution - Reactor Building, System 57-2; DCN 51094, CRDR - Main Control Room, Panel 1-9-3; DCN 51223, Core Spray - Electrical, System 75; and DCN 51217, Electrical 4KV Distribution - Reactor Building, System 57-5. Cable installation WO activities observed or reviewed included the following:

- 04-720414-51, replace control circuit cable for the normal feeder breaker on the A Diesel Auxiliary Board, System 82, and associated internal wiring per DCN 51090

- 04-720414-52, replace control circuit cable for the normal feeder breaker on the B Diesel Auxiliary Board, System 82, and associated internal wiring per DCN 51090

- 04-720414-58 determinate existing cable, pull back, reroute from transfer 0-XSW-248-00C1 switch in compartment 2A to the transfer switch in compartment 20A of 480V Shutdown Board 2A, modify compartment 20A to accept new breaker, 0-BKR-248-00C1, and install new cable to provide alternate feed to Battery Charger SB-C, per DCN 51090

- 03-002713-08, determinate existing cables, relabel as abandoned, terminate new cables, and label new cables for the B Emergency Diesel Generator, System 82, per DCN 51217

- 03-004725-06, determinate existing cables, re-label as abandoned, terminate new cables, and label new cables to Panel 1-LPNL-925-32 for the Unit Preferred 120V AC, System 252, per DCN 51214

- 02-011686-09, complete final cable terminations to Primary Containment Isolation System (PCIS), in MCR Panel 1-LPNL-9-3A, Channel A, System 64D, per DCN 51094

- 02-016202-53, pull, install, and terminate cables in Panels 25-45A, 45B, and 45C for Core Spray, System 75, per DCN 51223

During the above cable installation activities, the inspectors verified that the ongoing cable installation activities were performed in accordance with the documented requirements including monitoring pulling tension. The inspectors attended pre-job briefings and concluded that craft personnel were knowledgeable of cable installation requirements. Additionally, the inspectors observed that quality control (QC) inspection personnel and craft supervisors monitored cable installation.

b.2 Review of Special Program Activities

Missing Conduit Bushings

During a previous inspection, as documented in Inspection Report 50-259/2005-009, the inspectors reviewed the details of the issue, the licensee's proposed corrective actions, and the implementation of modifications. The modifications were confirmed by reviewing the appropriate documentation. The inspectors verified that Unit 1 did not have any of this type cable that was susceptible to this problem installed. The inspectors also performed a complete walkdown of junction boxes that were identified with missing bushings. The inspectors concluded that the licensee performed an aggressive search for conduits with missing bushings that were beyond the scope of the program on Unit 1. During this inspection, the inspector followed up on proposed corrective actions for an outlier identified by the licensee on the previous inspection. This outlier is discussed in more detail in Section E8.12 of this report. The inspector also evaluated the use of cable jacket material and tie-wraps around the effected cables to prevent damage to cables for existing plant configuration. Also during this inspection, the inspector reviewed additional work orders to verify that new installation of conduits had included the use of bushings. In addition, the inspector verified that G-40, (General Engineering Specification) also addressed the use of bushings for conduits to prevent any type of cable from being damaged by conduit edges. Based on current and previous reviews the inspectors concluded that implementation of this sub-program was adequate and no further inspections in this area were required.

Pulling Cable Through a 90E Condulet or Mid-run Flexible Conduit

During previous inspections, as documented in Inspection Reports 50-259/2004-009 and 50-259/2005-009, the inspectors performed a detailed review of the licensee's methodology and a walkdown inspection of the Unit 1 control complex to look for examples of this issue. During those inspections the inspectors reviewed a sample of flexible conduits and verified quality control type checks of calculations, walkdown data and dispositions. During isometric-based walkdown of cables installed, the inspectors verified that cable issues were appropriately dispositioned per calculations and that current installation practices were adequate. Based on those previous reviews the inspectors concluded that implementation of this sub-program was adequate and no further inspections in this area were required.

Brand Rex Cable

During a previous inspection, as documented in Inspection Report 50-259/2005-009, the inspectors reviewed the details of this issue and the relevant design criteria. The inspectors reviewed calculation, Evaluation for Use of Brand Rex Cable and Contract 80K6-825419, in Unit 1. The calculation concludes that no cables from the problem batch were installed on Unit 1. The inspectors reviewed records to verify date of delivery of this batch versus time frame of Unit 1 construction and dates that batches were installed to verify that Unit 1 did not have Brand Rex cables installed. The inspectors also reviewed work orders and the EQ program cable list for installed cable types. The inspectors concluded that the calculation was adequate and no cables from the problem batch from Brand Rex is installed on Unit 1. Based on this previous review the inspectors concluded that implementation of this sub-program was adequate and no further inspections in this area were required.

Use of Condulets as Pull Points for Large 600 Volt Cable

The sub-program associated with the use of condulets as pull points is designed to ensure that possible damage to 300 MCM and larger 600V cable would not result from inserting large, stiff, single conductor cables in standard form condulets at the completion of the pull. Specifics for this program are defined in Supplemental Safety Evaluation Reports transmitted on March 19, 1993, and October 3, 1995, which applied to all three units at Browns Ferry. The essential elements of this sub-program were development of installation criteria, documented walkdown inspections of installed conduits.

During a previous inspection the inspectors verified that calculation EDQ1 999 2002 0074, Analysis of Unit 1 Large 600 V cables in Standard Condulets, which contains a list of 40 safety-related large 600 V cables installed in standard condulets, was acceptable. This review was documented in Inspection Report 50-259/2004-07. During that review the inspectors also reviewed two separate lists of all Unit 1 and common V3/V4 300 KCMIL or larger safety-related cables and verified these 40 cables were acceptable.

During the current inspection period the inspectors reviewed targeted conduits, calculations, drawings, and physical arrangement. The licensee informed the inspectors that all 40 potentially affected cables are either being replaced or were previously evaluated and dispositioned under U2 or U3 restart programs. The inspectors performed a review of PIC 61525 and also a walkdown related to reinstating Transformer TS1E, System 231. The licensee originally planned to replace the existing 4160V to 480V PCB oil filled 750KVA transformer with a Class 1E, dry type, 1000/1333KVA transformer to support increased capacity for Unit 1 restart. However, further evaluation determined that the existing transformer was sufficient for supplying alternate power to the safety related 480V Shutdown Board 1A and 1B. The inspectors performed a walkdown related to the replacement of Transformer TS1E, System 231. The licensee performed walkdowns to assess if existing conduit for the reinstated transformer should be evaluated for the use of condulets as pull points for large 600 volt cables. The inspector performed an independent inspection of accessible portions of conduits at the termination points of the four installed cables to verify that the concerned cables, which could be susceptible to damage from the use of condulets as pull points, were not damaged in any standard condulets. It was determined that the conduits did not have condulets by visual inspection at the termination points. Activities observed by the inspectors included observing termination points of 4KV cables: PP627-1B, PP628-1B, PP630-1B, and PP631-1B. For the areas not accessible for inspection, the inspectors used isometric drawings to verify that drawings and physical arrangement of cables were consistent. In addition, the inspector reviewed DS-E13.1.4 Maximum Cable Diameter for Various Rigid Steel Conduits for cables that were reinstated for use to verify that cables installed met accepted industry standards. The inspectors concluded that all cables which could be affected were identified during the licensee walkdowns and all issues have been resolved. Based on current and previous reviews the inspectors concluded that implementation of this sub-program was adequate and no further inspections in this area were required.

c. Conclusions

Cable installation activities continued to be performed in accordance with documented requirements. Additionally, based on current and previous reviews, the inspectors determined that implementation of four sub-programs for the Cable Installation Special Program were proceeding in accordance with licensee commitments and regulatory requirements. These sub-programs include use of condulets as pull points for large 600 volt cable, missing conduit bushings, pulling cable through a 90E condulet or mid-run flexible conduit, and brand rex cable installation. Completed actions to address these issues for Unit 1 are consistent with those previously committed to and performed for Units 2 and 3. The inspectors concluded that no issues related to these sub-programs that would negatively impact the restart of Unit 1 were identified as the result of the above reviews. No further inspections are anticipated for these four sub-programs. However, implementation activities associated with cable separation and the five remaining cable installation sub-programs (sidewall pressure, pullybys, jamming, vertical supports in raceways, and bend radius of medium voltage cables) will need further inspections by the NRC to verify corrective actions are in accordance with licensee commitments.

E1.6 Restart Special Program Activities - Flexible Conduit (37550)

a. Inspection Scope

The special program associated with the installation of flexible conduit is designed to ensure that flexible conduits are installed on equipment and devices that rotate, vibrate, are subject to thermal movement, or from seismic events without damage to the conduit or installed cables. Specifics for this program are defined in Supplemental Safety Evaluation Reports transmitted on March 19, 1993, and October 3, 1995, which applied to all three units at Browns Ferry. The essential elements of the special program for flexible conduits are development of installation criteria, documented walkdown inspections of installed flexible conduit and correction of any conduits not meeting the criteria.

b. Observations and Findings

Previous NRC reviews of flexible conduit activities were documented in Inspection Reports 50-259/2004-07, 50-259/2004-09, and 50-259/2005-09. During this inspection period the inspectors conducted walkdowns of all areas in the Unit 1 Control Bay focusing on flexible conduit installation. Conduits were inspected against the design drawings for configuration, bend radius, length, connection details, and other construction requirements. The inspectors performed walkdowns of various modifications and verified that the modifications were implemented in accordance with the applicable design requirements. Following various walkdowns, any deficiencies noted by the inspectors, were subsequently addressed by the licensee and documented in the licensee's corrective action program as PERs to disposition the discrepancies. The issues identified were of a low safety significance, and would not have had any significant consequences on the ability of the flexible conduit to perform the intended function.

Additionally, the inspectors reviewed calculation EDQ1 999 2003 0014, Analysis of Flex Conduit to Devices for Unit 1, Rev. 1 for flexible conduit installations for completeness, accuracy and adherence to design criteria and procedural requirements specified in engineering specification G-40. The modifications were implemented in accordance with DCN 51222. No deficiencies were identified.

c. Conclusions

Implementation of the Flexible Conduit Special Program continued to be acceptable. Based on current and previous observations, document reviews, and discussions with engineering personnel, the inspectors determined that this program is proceeding in accordance with licensee commitments and regulatory requirements. Corrective actions to address issues for flexible conduit have been performed or are being performed by the licensee. Completed or planned actions to address these issues for Unit 1 are consistent with those previously performed for Units 2 and 3. No issues related to flexible conduit that would negatively impact the restart of Unit 1 were identified as the result of the above review. Based on this and previously documented NRC inspections, the inspectors concluded that at this time, no further inspections are anticipated for this Special Program.

No violations or deviations were identified during this review of the licensee's Flexible Conduit Special Program for Unit 1.

E1.7 Special Program Activities - Miscellaneous Steel Frames (37551)

a. Inspection Scope

During investigations performed in the 1980's by the licensee and the NRC related to restart of Unit 2, numerous deficiencies were identified in design and construction of safety-related structural steel platforms. These included deficient structural connections, failure to construct the platforms in accordance with design documents, deficiencies in welding (primarily undersized fillet welds), seismic design issues, and configuration management issues (i.e., failure to control addition of more loads to platforms). Miscellaneous steel frames are various structural steel platforms in the reactor building which support pumps, coolers, piping and other equipment.

The licensee's commitments for resolution of issues associated with the structural steel platforms are stated in TVA letter dated December 13, 2002, Subject: Browns Ferry Nuclear Plant - Unit 1 - Regulatory Framework for the Restart of Unit 1. The letter references previous commitments for restart of Units 1 and 3 stated in a letter dated July 10, 1991, Subject: Regulatory Framework for the Restart of Units 1 and 3, and NRC approval of the licensee's plans in a letter dated April 1, 1992. Design criteria for design and seismic qualification of the miscellaneous structural steel platforms were submitted to NRC in TVA letters dated June 12, 1991, June 13, 1991, and February 6, 1992. Acceptance of the licensee's design criteria for the structural steel platforms by NRC is documented in a Safety Evaluation Report dated July 13, 1992, Subject: Design Criteria for Lower Drywell Steel Platforms and Miscellaneous Steel. Structural steel platforms in the drywell were previously inspected under the Drywell Steel Platforms Special Program. The NRC has completed inspections for that Special Program as was documented in Inspection Report number 259/2005-008.

b. Observation and Findings

The Unit 1 miscellaneous structural steel platforms have been redesigned to meet current design criteria. Design changes were issued to modify the structural steel platforms to correct the deficiencies. The modifications primarily involved reinforcing connections on the platforms by replacing bolts, addition of clip angles and stiffeners, addition of cover plates to some beams, and reinforcement of existing welds. The modified structural platforms were designed to meet current design criteria and have a design margin for addition of future loads, if necessary.

In order to perform the design verification for the as-built miscellaneous structural steel platforms in Unit 1, walkdown inspections were performed by engineering personnel to document the condition, configuration, and existing loadings on the miscellaneous platforms. The inspectors reviewed TVA procedure WI-BFN-0-CEB-02, Walkdown Instructions for Seismic Issues (Civil), which specified the requirements for performance of the walkdown. The inspectors independently verified selected as-built details on miscellaneous structural steel Elevation 541' 6" platforms in the southwest and northeast quad rooms. The details included weld size and length, dimensions of members, connection details, and identification of type and condition of structural steel bolts. The inspectors reviewed walkdown package numbers WDP-BFN-1-CEB-303-02-PLT-01, -03, and -04 which documented results of the walkdowns. The inspectors reviewed Calculation numbers CDQ1-303-2003-1027 and -1029 which evaluated the Elevation 541' 6" platforms in the reactor building southwest and northeast quad rooms. The inspectors verified the existing as-built details identified during the walkdowns were evaluated in the design calculations.

During review of the design calculations, the inspectors determined that two beams with bent flanges in the southwest quad had not been evaluated in design calculation number CDQ1-303-2003-1029. The licensee issued Problem Evaluation Report (PER) number 100727 to document this issue. The calculation was reissued as Revision 4 to address the beams with the bent flanges. The bent flanges had no significant effect on the capacity of the beams.

The inspectors reviewed design drawings, work control instructions, quality control inspection procedures and walked down the platforms to examine the completed modifications. During the walkdowns, the inspectors verified the following attributes complied with the requirements shown on the design drawings: member sizes, configuration, installation of cover plates on radial beams, weld sizes, type, and length, connection details, and installation of correct type bolts in existing connections. Platforms examined were those in the northeast and southwest quadrants of the reactor building, elevation 541' - 6", the core spray platform at Elevation 604' - 8", and the core spray pump cooler platform at Elevation 555' - 1".

During the walkdowns to determine the as-built conditions of the platforms, the licensee’s walkdown inspection teams were unable to ascertain the type of bolts in some connections. Bolts for which the type could not be identified were either evaluated in the design calculations as lower strength bolts, or were designated for replacement on the modification drawings with high strength structural steel bolts conforming to ASTM A325. After review of the licensee’s walkdown documents, design calculations, and the DCN documents, the inspectors identified several connections where bolts which could not be identified during the licensee’s walkdown inspections were not addressed in either the calculations or specified for replacement on the drawings. The licensee issued PER 96471 to document and disposition this issue. The licensee reinspected the connections which were shown in the walkdown records to have unidentified bolts. The existing bolts were found to conform to ASTM A325 in all but a few locations where access to the bolts were restricted. For connections where the existing bolt type was indeterminate, the existing bolts were evaluated in Revision 3 of Calculation CDQ1-303-2003-2202 as lower strength Type A307 bolts. The inspectors reviewed the calculation which showed the connections were adequate with the lower strength bolts. The inspectors also examined a sample of the remaining bolts identified by the licensee’s as-built walkdown teams as unknown type, and verified the bolts were Type A325. The inspectors concluded that this issue was an isolated example of a documentation discrepancy which had minor safety significance.

The inspectors also reviewed a self-assessment, number RES-REN-05-010 conducted by TVA engineering personnel to assess the adequacy of the design methodology, calculations, and design output documents for the Unit 1 drywell structural steel modifications. Two PERs were initiated as a result of the assessment.

c. Conclusion

Based on observations, document reviews, and discussions with engineering personnel, the inspectors determined that completed actions to address concerns with the Unit 1 miscellaneous structural steel platforms complied with the licensee's commitments to NRC. Based on this inspection, no further inspections are anticipated for the Unit 1 Miscellaneous Steel Frames Special Program. No findings of significance were identified.

E1.8 Special Program Activities - Platform Thermal Growth (37551)

a. Inspection Scope

Platform thermal growth involved the effect of thermal loads on structural steel platforms. The issue was initially identified during reviews performed in the 1980s by the licensee and the NRC related to restart of Unit 2. Additionally, the NRC Office of Nuclear Reactor Regulation had questioned the licensee regarding their use of nonlinear analysis which predicted plastic deformation of structures due to thermal loads. The licensee committed to revise the criteria to require steel members to remain within the elastic limit for all loading combinations. These issues were addressed by TVA for the Unit 1 recovery as part of the Drywell Platforms and Miscellaneous Steel Frames Special Programs.

b. Observation and Findings

During previous NRC reviews of Unit 1 Special Program activities related to the drywell structural steel platforms, the inspectors verified that the licensee had considered thermal loads in their analysis of Unit 1drywell structural steel platforms. Modifications required by thermal loads included installation of bolts in slotted holes in beam connections, replacement of welds with bolts in some connections to change the connection from fixed to pinned, and strengthening of beams to carry higher stresses with addition of beam stiffeners, and/or cover plates. These were implemented by design change packages previously inspected by NRC. The drywell structural steel platform program was closed in NRC Inspection Report number 50-259/2005-008.

During the current inspection, the inspectors reviewed Calculation number CD-Q2303-930315, Resolution of NRC Comments on Thermal Issues for Miscellaneous Steel. This calculation addressed specific NRC review comments on the evaluation of the platform thermal growth program applicable to miscellaneous steel platforms, which was summarized in an NRC letter dated March 22, 1993, Subject Summary of the March 2, 1993 Meeting regarding Thermal Growth of Steel Structures at the Browns Ferry Nuclear Plant. The inspectors verified that modifications specified in the calculation to permit the platforms to grow under thermal conditions were included in the DCN documents. The inspectors examined the completed modifications which involved removal of welds at fixed connections and replacement of the welds by installation of bolts, tightened snug tight, in slotted holes. Specific details examined were those shown on Drawing DCA-51375-001, Sections C2 - C2 on the Core Spray Pump Cooler

Platform, NW Corner at Elev. 555' - 1"; Sections V2 - V2 on the Core Spray Pump Cooler Platform, NE Corner at Elev. 557' - 31/2 "; and Sections D2 - D & G2 - G2 on the RHR Pump Cooler Platform, SW Corner at Elev. 554' - 6". The inspectors also reviewed quality records documenting weld removal, and installation of snug tight bolts in slotted holes at the connections, in the work plan packages.

c. Conclusion

Based on observations, document reviews, and discussions with engineering personnel, the inspectors determined that the licensee has completed actions to address concerns with the Unit 1 structural steel platforms and complied with commitments to the NRC regarding platform thermal growth. Based on this inspection, no further inspections are anticipated for this Special Program. No findings of significance were identified.

E1.9 Special Program Activities - Control Rod Drive (CRD) Insert and Withdrawal Piping (37551)

a. Inspection Scope

During inspection of cable tray supports in the Unit 2 reactor building the licensee identified an issue regarding attachment of control rod drive hydraulic drive (CRDH) system piping to the cable tray support structure. The licensee performed an extensive design evaluation of the Unit 2 CRDH piping system which identified concerns regarding the adequacy of the CRDH supports to carry the design basis seismic loads. The Unit 2 CRDH frames, which were fabricated from unistrut members required extensive modifications prior to Unit 2 restart. Walkdown inspections by licensee engineering personnel of the Unit 3 CRDH piping and support frames showed that the Unit 3 frames, were identical to the Unit 2 CRDH frames. Due to cost and schedule considerations, the licensee decided to replace the Unit 3 CRDH frames by installing 32 new CRDH pipe support frames fabricated from structural tube steel. On Unit 1, the licensee also decided to remove the existing 32 CRDH frames which had been fabricated from unistrut and replace them with new structural steel frames. The criteria for seismic qualification of the CRDH frames is summarized in an NRC letter to TVA, dated December 11, 1989, Browns Ferry Nuclear Plant - Unit 2 - Revised Program Plan - Seismic Qualification of the Control Rod Drive Hydraulic (CRDH) Piping System.

b. Observations and Findings

During the current inspection, the inspectors examined three of the new Unit 1 CRDH support frames. The frames examined were numbers 114, 116, and 128 installed in the reactor building. The new frames were inspected against the design drawings for configuration; member size; weld size, type and length; connection details; attachment of the CRDH piping to the new support frame; and concrete expansion anchor and baseplate installation details stipulated by the drawings and installation procedures. The inspectors reviewed quality records documenting fabrication and inspection of CRDH frame numbers 106, 112, 120, and 128. The quality records included concrete expansion anchor installation data sheets, weld data sheets documenting weld filler material, welding process, identification of welder, and weld inspection results, pipe clamp installation records, and documentation of materials/hardware used to construct supports.

The inspectors also reviewed PERs issued to document and disposition discrepancies identified by quality control inspectors during final as-built walkdown of CRDH frame numbers 114, 115, 116, 117, 124, and 132. The inspectors examined calculation number CDQ1-085-2002-1258, Revision 4, which evaluated adequacy of CRDH Frame 115 to use-as-is, some of the changes to the frame identified in the final as-built walkdown and incorporated field changes to the original design documents. The inspectors also examined WO 06-712186-000, issued to repair CRDH supports 1-47E468-113, 1-47E468-114, 1-47E468-115, and 1-47E468-116 to resolve discrepancies identified during final as-built walkdowns, as referenced in PER numbers 93329, 95942, 96114, and 95538.

c. Conclusions

The inspectors concluded that the modifications to the CRDH support frames were implemented in accordance with design requirements, except for the discrepancies identified by the licensee's quality control inspectors. These discrepancies have been evaluated in PERs, and corrective actions have been planned in accordance with the licensee's quality assurance program. Based on this inspection and a previous NRC inspections documented in Inspection Report number 50-259/2005-009, no further inspections are anticipated for this Special Program. No findings of significance were identified.

E1.10 Special Program Activities - Small Bore Piping and Instrument Tubing (37550)

a. Inspection Scope

The small bore piping (less than 2.5 inch diameter) program was developed by the licensee to address concerns identified with application of design criteria, incomplete support details, questions regarding seismic qualification, and lack of design calculations. The small bore piping includes instrument tubing, but does not include piping which had been rigorously analyzed, such as the control rod drive (CRD) piping. The licensee's program to resolve the concerns involve identification of the small bore piping and instrument tubing systems; performance of walkdown inspections to identify inadequately supported piping and tubing, missing supports, and missing hardware from existing supports; preparation of as-built drawings; completion of design calculations to qualify the small bore piping and tubing; issuing DCNs to correct discrepancies; and implementation of the DCNs.

The licensee's commitments for resolution of issues associated with the small bore piping and instrument tubing are documented in TVA letter dated December 13, 2002, Subject: Browns Ferry Nuclear Plant - Unit 1 - Regulatory Framework for the Restart of Unit 1. The letter references previous commitments for restart of Units 1 and 3 stated in a letter dated July 10, 1991, Subject: Regulatory Framework for the Restart of Units 1 and 3, and NRC approval of the licensee's plans in a letter dated April 1, 1992. Design criteria for design and seismic qualification were submitted to NRC in TVA letters dated February 27, 1991, Subject: Action Plan to Disposition Concerns Related to Units 1 and 3 Small Bore Piping; February 27, 1991, Subject: Action Plan to Disposition Concerns Related to Units 1 and 3 Instrument Tubing; and December 12, 1991. February 27, 1991, Subject: Small Bore Piping, Tubing, and Conduit Support Plan for Units 1 and 3 - Additional information. Acceptance of the licensee's program for resolution of the small bore piping and instrument tubing concerns by NRC is documented in a Safety Evaluation Reports dated October 24, 1989, and January 23, 1991.

b. Observations and Findings

The inspectors reviewed walkdown procedures and design criteria, reviewed results of walkdown inspections performed by licensee engineering personnel, reviewed design calculations and DCNs, walked down selected small bore piping and instrument tubing systems, and examined completed modifications. The inspectors also examined new instrument tubing and supports which were not included under the small bore piping program, but were designed and installed using the same criteria. These systems included the Reactor Vessel Level Indicating System (RVLIS) installed under DCN 51163 and new jet pump instrumentation tubing for the Reactor Water Recirculation (RWR) system installed under DCN 51045.

The inspectors reviewed results of walkdown inspections, design calculations, design change documents and completed modifications. The inspectors walked down the small bore piping/instrument tubing on portions of the Emergency Equipment Cooling Water (EECW) System (System 67), High Pressure Coolant Injection (HPCI) System (System 73) and portions of the Main Steam System (System 01) instrumentation in the drywell to verify the system walkdowns were completed in accordance with the licensee's walkdown procedures and deficiencies were identified and documented. The deficiencies included excessive span lengths between supports, missing hardware on supports, overloaded supports, and inadequately constructed supports. The inspectors reviewed calculations which evaluated the deficiencies and the design output documents (DCNs) which specified the required field work to correct the deficiencies.

The inspectors walked down portions of small bore piping and instrumentation tubing for the Main Steam (MS), HPCI, EECW, RVLIS and RWR to verify that the design changes were implemented in accordance with the design documents. Attributes examined were support location, configuration, including member size and type, weld size, and hardware for attachment of piping/tubing to supports, and support attachment to building structure. The inspectors also examined supports which were identified with missing or incorrect hardware to verify the correct type hardware was installed as specified in the DCN design drawings.

Supports examined were as follows:

- Main Steam Instrumentation support numbers 1-47B600-250, 1-47B600-251, 1-47B600-252, 1-47B600- 255, 1-47B600-256, and 1-47B600-257.

- High Pressure Coolant Injection Instrumentation support numbers 1-47B455-3, 1-47B455-4, 1-47B455-5, and 1-47B455-6.

- Portions of EECW system support numbers 1-47B451-3057, 1-47B451-3058, 1-47B451-3059, and 1-47B451-3061 (Note: Due to insulation on piping, details of attachment of piping to supports were not inspected since they were covered by insulation).

- RWR system support numbers 1-47B600-2673, 1-47B600-2674, 1-47B600-2675, 1-47B600-2676, 1-47B600-2677, 1-47B600-2678, 1-47B600-2679, and 1-47B600-2681.

- RVLIS support numbers 1-47B600-2658 , 1-47B600-2659, 1-47B600-2662, and 1-47B600-2664.

Note: All above listed supports were located in the drywell except for the EECW supports which were in the reactor building.

During examination of support number 1-47B600-2659 on the RVLIS system, the inspectors noted that the clearance between the sides of the supporting box frame and instrument tubing was less than specified on the design drawing. This issue was documented in PER 100686, Minimum Clearance Between RVLIS Tubing and Support 1-47B600-2658 Not Met. Design engineering analyzed the as-constructed clearance and determined it was acceptable. The inspectors concluded that this issue was an isolated example of a construction discrepancy which had minor safety significance.

The inspectors also identified several discrepancies between completed supports and the design drawings. The licensee provided copies of engineering change control documents, Post Issue Changes (PICs), which authorized the changes. Final review of the PICs are being conducted by engineering. The drawings will be revised to incorporate the approved PICs. In the case when a PIC is not approved, a WO will be issued to perform additional field work to make any changes necessary so the affected supports meet design criteria. These discrepancies between the original design drawings and as-constructed work resulted from construction changes to the design drawings which were controlled in accordance with the licensee's design control procedures approved in accordance with the TVA Quality Assurance Program.

c. Conclusions

During the walkdown inspection, the inspectors verified the following attributes complied with the requirements shown on the design drawings: support locations, support member sizes and configuration, weld sizes, type, and length, connection details, and verification of correct type of hardware for attachment of small bore piping/ tubing to supports. The inspectors determined that the licensee's program for correction of deficiencies identified in support of small bore piping and instrument tubing complies with the design criteria, commitments to NRC, and NRC requirements. Inspection of small bore piping and instrument tubing installed inside the drywell has been completed. However, additional samples of small bore piping and instrument tubing installed in the reactor building (outside the drywell) will be inspected prior to closure of this Special Program. No findings of significance were identified.

E1.11 Special Program Activities - Containment Coatings (37500)

a. Inspection Scope

The licensee committed to evaluate the drywell coatings under Section 14.3 of their Nuclear Performance Plan. The program includes inspections to identify unqualified coatings, calculation of the allowable quantity of uncontrolled coatings, and removal of uncontrolled coatings if required. Acceptance of the licensee’s program for containment coatings by NRC is documented in NUREG-1232, Vol.3, Supp. 2, Section 3.7, Containment Coatings.

b. Observations and Findings

Based on previous reviews the inspectors determined that the licensee’s program for repair and inspection of coatings in the torus were consistent with their commitments to the NRC. No further inspections of the Containment Coatings Special Program associated with the torus are anticipated.

The licensee performed a walkdown inspection of the drywell in accordance with procedure numbers 0-TI-417, Inspection of Service Level I, II, III Protective Coatings, Rev. 4, and WI-BFN-1-MEB-03, BFN Unit 1 Primary Containment Coatings Inspection Plan, Rev. 1. The results of the inspections were documented on Notification of Indication forms and drywell coating inspection records which list components with unqualified coating and include the area and average dry film thickness of the existing coating. WOs are being prepared to address the deficiencies. Corrective actions may include removal of the component with an unqualified coating, removal of the unqualified coating and repaint the area affected, or reducing the thickness of the unqualified coating to less than three mils. All unqualified coatings remaining in the drywell will be listed in the Unqualified Coating Log.

During the current inspection period the inspectors performed a walkdown of the drywell from Elevation 550 to 628 and examined the coatings on the interior of the drywell vessel and on hardware installed inside the drywell. The inspectors verified the accuracy of the licensee’s coatings walkdown inspection records. The inspectors reviewed Notification of Indications issued by the licensee’s Inservice Inspection staff which documented defective coatings on the drywell vessel and other defects such as degraded moisture barrier between the elevation 550 concrete floor and steel drywell vessel, areas of corrosion on the vessel, and presence of other defects such as grind marks or gouges in the vessel base metal. The inspectors also witnessed adhesion tests performed at a location selected by the inspectors at Elevation 616, Azimuth 330, to determine the pull-off strength of the coatings, and reviewed results of other adhesion tests performed by the licensee. The test results showed the coatings met acceptance criteria for adhesion to the steel drywell vessel.

In May of 1987 the licensee initiated a long term corrosion study for the steel drywell vessel. The coatings were removed from six areas measuring 5" by 7" at approximately even spaced areas around the circumference of the vessel. The test areas are at elevation 550, located just above the moisture barrier. Thickness measurements have been performed on the steel vessel in 20 locations in each test area using ultrasonic techniques approximately every three years since 5/87. The most recent measurements were performed in 2004. Review of the data shows that corrosion has been negligible during the test period. The thickness of the vessel wall meets or exceeds design requirements.

c. Conclusions

Based on previous reviews the inspectors determined that the licensee’s program for repair and inspection of coatings in the torus were consistent with their commitments to the NRC. No further inspections of the Containment Coatings Special Program associated with the torus are anticipated. Additionally, the inspectors determined that the licensee’s program for inspection of protective coatings in the drywell, and identification and documentation of deficiencies are consistent with their commitments to NRC. Work orders are being prepared to specify corrective actions. However, additional inspections of the Containment Coatings Special Program will be performed to examine implementation of corrective actions associated with protective coatings in the drywell.

E1.12 Special Program Activities - Long Term Torus Integrity Program & Large Bore Piping and Supports Program (IP 50090)

a. Inspection Scope

The inspectors reviewed Design Criteria BFN-50-C-7103, Structural Analysis and Qualification of Mechanical and Electrical Systems - Piping and Instrument Tubing, Rev. 5 (Including Attachment A - Rigorous Piping Analysis and Attachment E - Analysis of Torus Attached Piping), BFN-50-C-7100, Design of Civil Structures, Attachment A - General Design Criteria for the Torus Integrity Long Term Program, Rev. 16, and BFN-50-C-7107, Design of Class I Seismic Pipe and Tubing Supports, Rev. 7. The inspectors selected one piping stress analysis calculation and three pipe support calculations for Torus Attached Core Spray Piping System and two support calculations for Residual Heat Removal (RHR) Service Water System for Large Bore Piping for review. The calculations were reviewed for adequacy and compliance with the design criteria, drawings, IE Bulletin 79-02, Pipe Support Base Plate Designs Using Concrete Expansion Anchors, and IE Bulletin 79-14, Seismic Analysis for As-Built Safety-Related Piping Systems.

b. Observations and Findings

The inspectors reviewed the torus attached piping stress analysis, which was revised to change the thermal analysis, due to the torus water temperature increase associated with the anticipated power uprate. The inspectors reviewed support design calculations for the Torus Attached Core Spray System and RHR Service Water System for Large Bore Piping. The calculations reviewed are shown below.

Torus Attached Core Spray Piping System

Stress Calculation CDQ1-075-2003-1024, Summary of Piping Analysis Problem N1-175-1R, Rev. 13

Support Calculation No.	Rev. No.	Support No.
CDQ1-075-2003-1643	003	1-47B458-954
CDQ1-075-2003-1656	008	1-47B458-967
CDQ1-075-2003-1659	008	1-47B458-970

Large Bore RHR Service Water Piping System

Support Calculation No.	Rev. No.	Support No.
CDQ1-023-2003-0238	006	1-47B450-262
CDQ1-023-2003-0252	003	1-47B450-267

The elements in the stress calculation were partially reviewed and included isometric drawings, assumptions, geometry and coordinates, loads and combinations, computer model, input and output, computation, and summary of the analysis. The loads included deadweight, thermal, earthquake (Operating Basis Earthquake - OBE and Safe Shutdown Earthquake - SSE), Safety-Relief valve discharge (SRV), and Loss of Coolant (LOCA) loading conditions.

The elements in the support calculations reviewed included assumptions, design methodology, special requirements or limitations, computer model, computer design input data, computer output data, computations and analyses, summary of results, conclusions, and attachments. The computer input data included node numbers and coordination, member numbers, end nodes, and properties, joint fix, member releases, seismic coefficient, loads and load combinations, weld sizes and configurations, base plates, anchor bolts, pipe support load transmittal from the stress group, structural attachment loading schedule to Civil Engineering Group, and allowable stresses for the members.

c. Conclusions

Based on a review of one piping stress analysis calculation and two pipe support design calculations for Core Spray piping in the Long Term Torus Integrity Program, and two pipe support design calculations for RHR Service Water in the Large Bore Piping and Support Special Program the inspectors concluded that ongoing activities for these Special Programs continued to be acceptable. However, additional samples of completed supports will need to be inspected prior to closure of these Special Programs.

E1.13 Boiling Water Reactor Vessel Internals Program (BWRVIP) Activities (71111.08, 37551)

a. Inspection Scope

The inspectors reviewed the licensee's program for completion of BWRVIP activities to determine status of completion of BWRVIP requirements. Surveillance Instruction, 1-SI-4.6.G, Inservice Inspection Program - Unit 1, defines the scope of ASME Code required Inservice Inspection (ISI) NDE examinations for Reactor Pressure Vessel (RPV) components. Technical Instruction, 0-TI-365, Reactor Pressure Vessel Internals Inspections (RPVII) Units 1, 2, and 3, defines the scope of in-vessel components subject to augmented examination requirements. Numerous BWRVIP Project documents provide the basis for including various in-vessel components within the scope listed in 1-TI-365.

b. Observations and Findings

The licensee planned to perform 100% new baseline examinations of RPV internals during the Unit 1 Recovery Project. General Electric (GE) was contracted by the licensee to perform the Phase I (contracted work scope) in-vessel inspections during the period of June 29 through October 7, 2005, and the Phase II (maintenance work scope) during the period October 21 through December 15, 2005. These in-vessel inspections were integrated with other scheduled in-vessel and refueling floor activities. During the ongoing inspections in the RPV the inspectors observed selected portions of ongoing NDE examinations of in-vessel components. Those previous observations and review of the Final Unit 1 ISI Report, BFN-1C06R-KCZKG, were documented in Inspection Report 50-259/2005-09. Activities associated with the Phase II examinations were to be detailed in a separate GE report which had not been issued at the close of that inspection period.

The inspectors reviewed the Final Unit 1 Phase II ISI Report, BFN-1C06R-MJJYJ, which was issued in February 2006. This report detailed the completed Phase II IVVI NDE examinations along with various examinations which had been deferred from Phase I. Components covered by this report included ultrasonic examination (UT), visual (VT-3) or enhanced visual (EVT) examination of the baffle plate welds (H8 and H9) at 0 degrees and 180 degrees; CRD housings, guide tubes, stub tubes, and stub tube to vessel welds; core plate bolting; feedwater nozzle N4A; core shroud head bolts; core shroud support legs and attachments; steam dryer and internals; top guide rim pins and rim welds; and vessel general area including below core plate region. Additionally, the inspectors reviewed selected UT examination procedures and qualification records for the GE UT and visual examination personnel.

During the previous review, as discussed in Inspection Report 50-259/2005-09, the inspectors had verified that all UT examinations required by the BWRVIP Program were satisfied except for the 20 jet pump hold down beams (tapered area BB-3) and associated bolting as required by BWRVIP-138. Those items had been examined but qualification of that vendor UT examination process was being questioned by the industry as a generic issue. During this inspection period the inspectors verified that the licensee had resolved this issue. The licensee had subsequently contracted for additional UT examination of those 20 jet pump hold down beams and associated bolting with a qualified process by a separate vendor that had also performed the recent RPV internals examinations during the spring 2006 Unit 3 refueling outage. Based on this review the inspectors concluded that the licensee had adequately resolved previous concerns about the qualification of the vendor UT process for the jet pump hold down beams and associated bolting. Additionally, the inspectors determined that the licensee’s in-vessel inspection program had satisfied all BWRVIP requirements, applicable code requirements and licensing commitments.

The inspectors noted that it will be necessary for General Electric personnel to return to the site for the purpose of performing several in-vessel repairs based on the results of these inspections. These will include installation of a Core Spray sparger clamp and Jet Pump aux wedges along with dryer repairs. This work is scheduled to be performed during August 2006.

c. Conclusions

The licensee had adequately resolved previous concerns about the qualification of the vendor UT process for the jet pump hold down beams and associated bolting. The inspectors determined that the licensee’s in-vessel inspection program had satisfied all BWRVIP requirements, applicable code requirements and licensing commitments.

E7 Quality Assurance in Engineering Activities

E7.1 Potential Damage to HPCI Discharge Piping (37550, 71152)

a. Inspection Scope

The inspectors evaluated the adequacy of licensee activities associated with potentially damaged sections of the Unit 1 HPCI discharge piping. Also, the inspectors’ review was to assess whether any issues were processed in accordance with licensee Procedure SPP-3.1, Corrective Action Program.

b. Observations and Findings

The inspectors reviewed the licensee activities associated with PER 97951. This PER documented the licensee’s identification of hammer marks on sections of the 14 inch HPCI carbon steel discharge piping located near support 1-47B455-2260 in the Unit 1 Reactor Building. The damaged areas consisted of multiple hammer marks which most likely resulted when craftsman accidently hit the piping during installation of a temporary pipe support installed on the HPCI line associated with modification activities performed on pipe support, 1-47B455-2260. The temporary support had been installed under WO 03-013930-006 during February 2006. The inspectors reviewed WO 03-013930-006 and Base Metal Evaluation (BME), 1-03013930-006-S22-BME-001, which documented special non-destructive examinations (NDE) and subsequent engineering evaluation of the damaged area of HPCI piping. The inspectors also reviewed completed Liquid Penetrant (PT) and Ultrasonic (UT) Thickness Examination reports performed on February 27, 2006. These NDE activities were performed by the licensee to verify minimum pipe wall thickness was still satisfied and to determine if any recordable indications, as required by ASME Section III requirements, existed in the damage area. No recordable indications were identified during the PT exam. Additionally, the UT thickness measurements showed that the lowest pipe wall thickness measurements in the area of damage was .934 inches which was well above the minimum allowable value of .821 inches. Both NDE reports were incorporated into the BME report. The inspectors concluded that the licensee's evaluation of the damaged piping was satisfactory and no further actions were required.

c. Conclusions

The licensee's evaluation of a damaged section of HPCI piping was acceptable. NDE examination of the affected section of piping did not identify any problems and the piping satisfied ASME code requirements. No violations or deviations were identified.

E7.2 Deficiencies with Electrical Cable Splices and Terminations in EQ Applications (37550, 71152)

a. Inspection Scope

The inspectors evaluated the adequacy of the licensee corrective actions to address various documented licensee identified deficiencies associated with installation of electrical cable splices. These discrepancies included completed work activities on splices installed on environmentally qualified (EQ) applications. The inspectors reviewed selected PERs and observed field activities. In addition, the inspectors held discussions with TVA and Stone & Webster Engineering Corporation (SWEC) management personnel, Nuclear Assurance (NA) personnel, and craft personnel. The inspectors evaluated of the effectiveness of self-assessments and audits in this area, and corrective actions associated with documented deficiencies. Also, the inspectors' review was to assess whether any issues were processed in accordance with licensee Procedure SPP-3.1, Corrective Action Program.

b. Observations and Findings

The nuclear industry, including Browns Ferry, has used heat shrink material manufactured by Raychem Corporation for many years. In 2001, Raychem Corporation was purchased by Tyco Corporation. Tyco Corporation continued suppling heat shrink material and proper certification and qualification paperwork for the available stock of heat shrink material. During 2003, due to changes in base material availability, Tyco Corporation changed the manufacturing process and provided new qualification and EQ test reports to TVA for the new formulation of heat shrink material.

The licensee had recently identified concerns associated with newer formulated Raychem heat shrink material introduced during 2003 versus the original formulated material and their application. Weaknesses associated with the installation and documentation of the new formulated heat shrink material were identified by the licensee. Additionally, during NA Assessment, NA-BF-06-001, documentation discrepancies were identified on completed QA records. These discrepancies were associated with completed Raychem/Tyco heat shrink installation WOs and were documented in B level PER 98178. Examples of discrepancies included MAI 3.3 Data Sheet 6 with installed material different from specified material, Raychem/Tyco material part number/lot number not properly documented, responsible engineer and second party failure to properly sign data sheet, and cable jacket diameter not properly entered. Similar conditions had been previously identified and documented in C level PER 98052. To ensure proper tracking of the new material and its usage TVA had developed new sets of EQ binders to contain records for the new formulation heat shrink material. The licensee determined that the change process was not adequate to ensure proper roll out of requirements for the new material. As the result procurement data sheets for various Raychem heat shrink material Catalog IDs (CAT IDs) were revised to include the new Raychem heat shrink material CA IDs as equivalent replacement without a DCN revision. This condition was documented in PER 98171.

As the result of these problems the Unit 1 Modifications manager placed a management hold on all Raychem/Tyco heat shrink installation activities and developed an action plan for resuming heat shrink installation which required SWEC management, NA management, and Unit 1 Vice President concurrence prior to resuming cable splice activities.

The inspectors reviewed PERs 98052, 98171, and 98178 and the licensee’s root cause analysis for PER 98178. The licensee concluded that the problems had occurred due to lack of craft training on requirements and expectations and inadequate staffing of personnel responsible for work package closure review. These had resulted in the use of non-qualified craft and craft supervision personnel for cable splice installation activities, failure to identify craft errors by craft supervision and during QC verification activities, and failure to identify work package documentation errors during first party and second party closure reviews. The licensee decided to select a limited number of electrical craft personnel for all future installation of cable splices and establish specific training and qualification requirements prior to resumption of cable splice activities. Corrective actions including completion of classroom training to address previously identified deficiencies and practical sessions for cable splice installation for craft, craft supervision, work package closure personnel, and field engineers; training on attention to detail and performance expectations for QC inspection personnel; and assigning additional personnel for 2nd party closure. The inspectors noted that corrective actions included a licensee review of all closed and in process safety-related and EQ related work orders that installed Raychem/Tyco cable splices for procedural adherence to MAI-3.3, Cable Terminating and Splicing for Cables Rated Up to 15000 Volts. The licensee reinspected a sample of completed splices and determined that the physical condition of the installed heat shrink material was found to be acceptable. Additionally, the associated EQ binders were properly updated. As the result of this review the licensee identified several additional examples of documentation errors similar to those identified in PERs 98052 and 98178.

The inspectors attended at least one class for each of the separate ongoing training and practical sessions for craft and field engineering personnel conducted prior to resumption of cable splice activities. Additionally, the inspectors attended the first pre-job briefing conducted after work activities were resumed. Cable splice activities resumed shortly before the end of the reporting period. The inspectors concluded that the licensee's corrective actions should be adequate to preclude future recurrence of similar problems in this area. Additionally, the inspectors noted that fewer documentation errors and improvement in craft performance had resulted from the increased management focus and oversight in this area. However, as of the end of the reporting period only non-safety related cable splicing activities had been performed.

c. Conclusions

Increased management focus and oversight of electrical cable splicing activities has initially resulted in fewer documentation errors and improvement in craft performance. However, the inspectors will continue to evaluate the effectiveness of the licensee's long term corrective actions in the area. No violations or deviations of significance were identified.

E8 Miscellaneous Engineering Issues (92701)

E8.1 (Closed) Bulletin 95-02, Unexpected Clogging of RHR Pump Strainer While Operating in Suppression Pool Cooling Mode

The inspectors reviewed Bulletin 95-02. This Bulletin requested licensees address the potential for debris plugging of RHR suction strainers while operating in the suppression pool cooling mode. TVA letters dated November 15, 1995, November 17, 1995, and March 5, 1996, provided the licensee's response to Bulletin 95-02. NRC review of the licensee's response was documented in a letter dated March 14, 1996. During that review the NRC determined that the response was acceptable for all three units. In addition, closure of this item prior to restart of Unit 3 was documented in NRC Inspection Report 50-259,260,296/95-60 based on inspectors review of the licensee's programs for maintaining cleanliness of the suppression pool. As was performed during Unit 3 recovery the inspectors have closely monitored the licensee's actions associated with maintenance and cleanliness of the Unit 1 torus. This included detailed inspections of torus prior to filling, inspection of coatings, cleanliness of torus, and installation of replacement Emergency Core Cooling System (ECCS) suction strainers. The inspectors determined that the licensee's actions associated with Bulletin 95-02 were adequate. The inspectors determined that this issue was effectively addressed for Unit 1. No further NRC inspection in this area is required. This item is closed.

E8.2 (Closed) Generic Letter (GL) 95-07, Pressure Locking and Thermal Binding of Safety-Related and Power-Operated Gate Valves

On August 17, 1995, the NRC issued GL 95-07, Pressure Locking and Thermal Binding of Safety-Related and Power-Operated Gate Valves. The GL requested licensees to take actions to ensure that safety-related power-operated gate valves that were susceptible to pressure locking or thermal binding are capable of performing their safety functions.

Inspectors reviewed licensee actions to address this issue. This issue was resolved on Units 2 and 3 in a SER dated, June 23, 1999. The issue was resolved on Units 1 in a safety evaluation report (SER) dated, January 28, 2005. Additionally, licensee actions to resolve this issue had been previously inspected for Unit 3 as documented in Inspection Report 259, 260, 296/95-53. Based on this review the inspectors concluded that Unit 1 activities are consistent with Units 2 and 3 actions.

The inspectors reviewed licensee letters to the NRC dated, May 11 and July 29, 2004, which provided the licensee’s response to this GL for Browns Ferry Unit 1. That response identified five safety-related valves that the licensee had determined were susceptible to pressure locking and one safety-related gate valve which was susceptible to thermal binding. This response also provided a description of modifications planned to resolve the issue. The inspectors reviewed Unit 1 DCNs which described the details of modifications planned to address the concern associated with pressure locking of twin disk gate valves. These modifications drilled a 1/4 inch hole on the high pressure side valve disk to preclude pressure locking in a manner similar as had previously been performed on Units 2 and 3. Specifically, the inspectors determined that DCN 51196 modified 1-FCV-71-39, RCIC Injection Valve; DCN 51199 modified 1-FCV-74-53, RHR Loop I Low Pressure Coolant Injection (LPCI) Injection Valve, and 1-FCV-74-67, RHR Loop II LPCI Injection Valve; and DCN 51200 modified 1-FCV-75-25, Core Spray Loop I Injection Valve, and 1-FCV-75-53, Core Spray Loop II Injection Valve. Additionally, the inspectors reviewed DCN 51198 which described the details for the modification planned to address the concern associated with thermal binding of 1-FCV-73-16, HPCI Steam Admission Valve. 1-FCV-73-16 is to be replaced with a new Anchor Darling gate valve similar to as previously installed on Units 2 and 3. The inspectors determined that the planned modifications will meet the licencee’s commitment to be resolve this issue. The inspectors determined that no further actions were required for Unit 1. Therefore, because these modifications are being tracked under the facility modification process and any deficiencies would likely be detected by the licensee’s oversight programs, this item meets the closure criteria established for Unit 1 recovery issues. This issue is closed for Unit 1

E8.3 (Closed) TMI Action Item II.B.3, Post Accident Sampling System NUREG 0737

The inspectors reviewed TMI Action Item II.B.3, Post Accident Sampling System, to determine the status of the licensee's efforts in this area. The inspectors determined that license amendments 245, 282, and 240 to licence numbers DPR-33, DPR-52 and DPR-68, issued May 9, 2003, had eliminated the requirement to establish and maintain dedicated equipment to obtain radioisotope information in a prompt manner. In the NRC safety evaluation it was noted that Unit 1 had been shutdown before a post accident sampling system was installed and that the amendment superceded prior commitments to install the system and the licensee may not need to complete activities to comply with those regulatory commitments or related regulatory requirements such as TS 5.5.3 which was deleted by the amendment. The licencee does, however, need to maintain and, if necessary, install the equipment to fulfill its regulatory commitment to have a contingency plan for obtaining highly radioactive samples of reactor coolant, suppression pool and containment atmosphere. To meet this requirement the licensee has opted to install a scaled down version of the sampling system that was installed in Units 2 and 3 that is capable of sampling reactor coolant, suppression pool water and containment atmosphere. The inspectors verified that a similar installation for Unit 1 is described in DCN 51185. Additionally, Calculation NDQ0043900029, "Post Accident Sampling Doses" was reviewed to determine the expected mission doses to personnel using the designed Unit 1 sampling system and the ability to meet the 5 rem whole body dose requirement. The inspectors determined that the licensee's planned sampling system installation will meet the licencee's commitment to be able to sample the reactor coolant, suppression pool water and containment atmosphere within a reasonable period of time for less than or equal to the doses expected from the installations on Units 2 and 3. The inspectors determined that no further actions were required for Unit 1. Therefore, because this modification is being tracked under the facility modification process and any deficiencies would likely be detected by the licensee's oversight programs, this item meets the closure criteria established for Unit 1 recovery issues. This issue is closed for Unit 1

E8.4 (Closed) TMI Action Item II.F.1.2.C, High Range Containment Radiation Monitors (Generic Letter 83-36)

The inspectors reviewed TMI Action Item II.F.1.2.C, High Range Containment Radiation Monitors, to determine the status of the licensee's efforts in this area. GL 83-36 identified several technical specifications that would be required by boiling water reactors to implement the post TMI action items contained in NUREG 0737. The licensee had previously submitted an amendment request on October 7, 1993, that incorporated the high range containment radiation monitor and recorder into the Technical Specifications for Units 1 and 3 and clarified a prior amendment to those for Unit 2 to include the recorders. The amendments were approved by the NRC on December 21, 1994. The inspectors reviewed the modifications to upgrade the Unit 1 high range containment radiation monitors for consistency with the requirements in NUREG 0737 and the amended Technical Specifications. The specific modifications associated with the unit 1 high range containment radiation monitors were described in DCNs 51171 and 51241. Additionally, the inspectors reviewed these DCNs and compared the planned modifications to DCN W-17556 which was used to install and upgrade the high range containment monitors in Unit 3.

During the review of DCN W-17556 an inconsistency with the requirements was identified in that the connectors for the detectors did not appear to meet the environmental qualification requirements for radiation exposure. Additional documentation, an environmental qualification package BFN0EQ- ECON 001 was reviewed which indicated that the connector did meet the environmental qualification requirements for radiation exposure. A review of BFN0EQ-CABL-050 established a similar qualification for the instrumentations cabling.

The inspectors determined that the licensee's planned upgrades were intended to bring Unit 1 instrumentation up-to-date, remain comparable to Units 2 and 3, and comply with NUREG 0737 requirements. The inspectors determined that no further actions were required for Unit 1. Therefore, because this item is effectively being tracked in the licensee's corrective action program, is being corrected similarly to the Unit 2 and 3 solutions with the same process, and because any implementation deficiencies would likely be detected by the licensee's oversight programs, this item meets the closure criteria established for Unit 1 recovery issues. This issue is closed for Unit 1.

E8.5 (Discussed) GL 88-01, NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping

GL 88-01 requested licensees to provide their plans for replacement, inspection, repair, and leakage detection of piping susceptible to IGSCC, and state whether they intend to follow the NRC staff positions or propose alternatives.

This issue was resolved on Unit 3 in SER dated, December 3, 1993, and Unit 1 activities are consistent with Unit 3 actions. Additionally, the inspectors had previously reviewed licensee actions for Unit 1 required by GL 88-01 to ensure compliance with guidance described in NUREG 0313, Rev 2. The most recent review in this area was documented in Inspection Report 50-259/2005-08. During that inspection the inspectors determined that the licensee had developed a thorough program to mitigate the long term effects of IGSCC on Unit 1.

The inspectors reviewed the licensee's letter to the NRC dated, July 21, 2004, BFN Unit 1 – Supplemental response to GL 88-01, NRC position on IGSCC in BWR austenitic stainless steel piping. This letter provided a comparison between the welds and categorization proposed for Unit 1 and those approved by NRC for Units 2 and 3. Licensee actions were determined to be consistent with details provided in this letter. No further NRC inspection in this area is anticipated. However, final closure of this item will be deferred until the Office of Nuclear Reactor Regulation (NRR) completes their review in this area and any SERs, if required, are issued.

E8.6 <u>(Discussed) GL 87-02, Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors and Unresolved Safety Issue (USI) A-46, Seismic Qualification of Equipment in Operating Plants</u>

On February 19, 1987, the NRC issued GL 87-02, “Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46.” The GL encouraged licensees to participate in a generic program to resolve the seismic verification issues associated with USI A-46. As a result, the Seismic Qualification Utility Group (SQUG) developed the “Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment,” Revision 2.

On May 22, 1992, the NRC issued Supplement 1 to GL 87-02 including the staff’s Supplemental Safety Evaluation Report No. 2 (SSER-2), pursuant to the provisions of Title 10, Code of Federal Regulations (10 CFR) 50.54(f), which required that all addressees provide either (1) a commitment to use both the SQUG commitments and the implementation guidance described in GIP-2 as supplemented by the staff’s SSER-2, or (2) an alternative method for responding to GL 87-02. The supplement also required that those addressees committing to implement GIP-2 provide an implementation schedule, as well as detailed information including the procedures and criteria used to generate the in-structure response spectra (IRS) to be used for USI A-46.

By letter dated March 21, 2000, the NRC issued the safety evaluation for the USI A-46 implementation program at Browns Ferry, Units 2 and 3. The USI A-46 program at BFN was established in response to Supplement 1 to Generic Letter 87-02 through a 10 CFR 50.54(f) letter. The staff has concluded that the Browns Ferry USI A-46 implementation program meets the purpose and intent of the criteria in the Generic Implementation Procedure, Revision 2 (GIP-2), and the staff’s Supplemental Safety Evaluation Report No. 2 on GIP-2 for the resolution of USI A-46. The corrective actions and completed physical modifications for resolution of outliers will result in sufficient basis to close the USI A-46 review for the Browns Ferry Unit 1 facility. The NRC staff has also concluded that its findings regarding the USI A-46 program do not warrant any further regulatory action under the provisions of 10 CFR 50.54(f). The above letter stated that activities related to the Unit 1 USI A-46 implementation will be subject to future NRC inspection.

By letter dated October 7, 2004, TVA submitted their response to GL 87-02, Supplement 1 stating that TVA has completed its seismic verification of BFN Unit 1 in accordance with GL 87-02. TVA stated that they used the methodology previously utilized for resolution of USI A-46 for BFN Units 2 & 3.

The inspectors reviewed the documents discussed below to evaluate the licensee’s evaluation and resolution of USI A-46 for BFN Unit 1. To accomplish this review the inspectors reviewed: Browns Ferry Nuclear Plant Unit 1 USI A-46 Seismic Evaluation Report ” Facility Risk Consultants, Inc., Report No. TVA/BFN-01-R-004, Revision 0, September 2004. This report provided supporting documentation for the resolution of USI A-46 for BFN-1. The injectors verified that the evaluations were performed in accordance with the SQUG GIP Revision 2A. The evaluations were performed consistent with the USI A-46 resolution programs for BFN-2 and BFN-3, and with the TVA and NRC agreed upon resolution process for USI A-46 at BFN.

The inspectors reviewed the requirements of USI A-46, as detailed in the GIP, which details the verification of issues into 4 areas: 1) Mechanical and Electrical Equipment, 2) Tanks and Heat Exchangers, 3) Essential Relays, and 4) Cable and Conduit Raceways. The inspectors held discussions with the engineering staff and reviewed the methodology and documentation of the issues including the Screening Evaluation Work Sheets (SEWS) and determined that the following areas have satisfied the commitments with respect to USI A-46.

The inspectors reviewed the program evaluations and walkdown data results for the following sub-parts of the A-46 program:

- ‘Tanks and Heat Exchangers’ had 16 tanks and heat exchangers that were identified on the Safe Equipment Shutdown Equipment List (SSEL) that fell within the scope of this program. Six outliers were identified and all have been resolved by further evaluation. No modifications were necessary.

- ‘Essential Relays’ there were no outliers identified.

- ‘Cable and Conduit Raceways’ which is discussed in DCN 51521, identified 14 outliers of all Reactor Building raceways. Of this group, 6 outliers were determined to be acceptable by further evaluation and 3 required plant modifications which have been completed.

- ‘Mechanical and Electrical Equipment’ which is discussed in DCNs 51521, 51085, and 51202, identified 523 items of equipment with 84 outliers of which 21 were resolved by further evaluation, 30 resolved by plant modification of which 17 have been implemented, six resolved by maintenance activities and 27 resolved by procedural changes. No outliers remain unresolved.

A second report “USI A-46 / Seismic IPEEE Relay Evaluation Browns Ferry Nuclear Plant Unit 1,” Facility Risk Consultants, Inc., Report No. TVA/BFN-01-R-001, Revision 0, January 23, 2004, provided information for the BFN Unit 1 relay evaluation for USI A-46 and the seismic portion of the Individual Plant Examination for External Events (IPEEE.) This work was performed in accordance with the appropriate industry guidance documents developed by the SQUG and the Electric Power Research Institute (EPRI), and approved by the NRC. The relay evaluation also utilized results of the similar relay evaluations for Units 2 and 3.

In summary, the relay evaluation findings are as follows:

- Inherent ruggedness of contact devices, chatter acceptability and seismic adequacy were sufficient to satisfactorily resolve the seismic acceptability of contact devices affecting the USI A-46 Safe Shutdown Equipment List (SSEL) components.

- No outliers were identified in the evaluation.
- No low ruggedness (bad actor) relays were found to be essential relays.
- No operator actions were identified in the evaluation as necessary to correct relay-chatter-caused malfunctions.
- Essential relays and the cabinets housing those essential relays were identified for the seismic capability engineers performing the seismic verification walkdowns and evaluations.

This report documented the combined USI A-46 and Seismic IPEEE relay evaluations for BFN-1. The evaluation was performed to assess the seismic adequacy of relays and other contact devices in the control circuits of equipment selected to bring the plant to safe shutdown in the event of an earthquake.

For the modifications work not yet completed, the inspectors verified that DCNs were being properly defined and tracked in a site tracking system that would assure proper completion. The inspectors also found that the utility had developed a punch list for a final walkdown verification for all the required work performed to implement all the commitments for the USI A-46 program. This walkdown just prior to start-up, would verify that all modifications remain intact as required and designed.

Based on this and previously documented NRC inspections, the inspectors concluded that no further actions were required for Unit 1. No further NRC inspection in this area is anticipated. However, final closure of these items will be deferred until NRR completes their review in this area and any SERs, if required, are issued.

E8.7 <u>(Discussed) GL 98-04, Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss- of- Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment</u>

The inspectors reviewed GL 98-04, and the licensee’s response to GL 98-04 for Unit 1 summarized in a TVA letter to NRC dated May 11, 2004, Subject: Browns Ferry Nuclear Plant Unit 1 - Response to Generic Letter 98-04 Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss- of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment. GL 98-04 requested information on Service Level I coatings related to controls on procurement, testing application and maintenance of the coatings. TVA Specification G-55, Technical and Programmatic Requirements for the Protective Coatings for TVA Nuclear Plants and TVA procedure MAI-5.3, Protective Coatings specify the licensee’s program for Service Level I coating. The inspectors reviewed the licensee’s specifications and procedures for control of coatings, examined procurement and storage of coating materials, surface preparation activities, application, and inspection of Service Level I coatings in the Unit 1 torus during inspections documented in NRC Inspection Report numbers 50-259/2004 - 007, 50-259/2004-009, and 50-259/2005-006. The inspectors also reviewed the results of preliminary inspections performed by the licensee to identify defective coatings in the drywell. Additional inspections are planned by the licensee following completion of major work activities and removal of scaffolding and construction equipment. The drywell coatings will be repaired in accordance with the licensee's procedures, TVA Specification G-55 and Procedure MAI-5.3.

Coatings which are determined to be unqualified and can't be removed or repaired are documented in an uncontrolled coatings log. The log is maintained in accordance with MAI 5.3, Attachment 4. The inspectors examined a preliminary copy of the Unit 1 coatings log. A calculation will be prepared to evaluate the uncontrolled coatings. Licensee engineers provided a copy of Calculation number MD-Q3303-940038, Revision 7, dated April 8, 2002, Primary Containment Uncontrolled Coatings Log, as an example of the methodology which will be used to evaluate the Unit 1 uncontrolled coatings. NRC acceptance of the licensee's evaluation criteria for uncontrolled coatings are documented in a Safety Evaluation Report on Browns Ferry Unit 2 Restart, NUREG-1232, Volume 3, Supplement 2, dated January, 1991.

No further inspection associated with this open item is anticipated. However, final closure of this item will be deferred until NRR completes their review in this area.

E8.8 <u>(Discussed) Generic Letter (GL) 97-04, Assurance of Sufficient Net Positive Suction for Emergency Core Cooling and Containment Heat Removal Pumps</u>

The inspectors reviewed licensee actions in response to GL 97-04. The purpose of this GL was to request licensees provide information necessary to confirm the adequacy of available net positive suction head (NPSH) for emergency core cooling (ECCS) and containment heat removal pumps. In response to GL 97-04, the licensee provided the requested information for the Browns Ferry units in a letter dated March 24, 1998. NRC review and closure of GL 97-04 for Units 2 and 3 was documented in a letter dated June 11, 1998. The inspectors reviewed the licensee's response for GL 97-04 for Unit 1 dated May 6, 2004. The inspectors identified several differences between the more recent Unit 1 response and the previous response dated March 24, 1998. The inspectors discussed these differences with applicable licensee engineers and determined that the analysis methodology used for Unit 1 was essentially the same as used on Units 2 and 3. However, the Unit 1 analysis had been performed for an extended power uprate condition which accounted for the difference. The extended power uprate is being handled under a separate licensee submittal requiring NRC approval. The inspectors determined that the licensee had provided adequate assurance of sufficient NPSH for ECCS and containment heat removal pumps. No further NRC inspection in this area is anticipated. However, final closure of this item will be deferred until NRR completes their review in this area and any SERs, if required, are issued.

E8.9 (Discussed) Bulletin 93-02 (and Supplement 1), Debris Plugging of ECCS Strainers

The inspectors reviewed licensee actions associated with Bulletin 93-02 dated May 11, 1993. The purpose of this bulletin was to notify licensees of a previously unrecognized contributor to the potential loss of NPSH for ECCS systems during a loss of coolant accident (LOCA). Subsequent to this the NRC issued Bulletin 93-02, Supplement 1, dated February 18, 1994 to inform licensees of additional information related to the vulnerability of ECCS suction strainers at boiling water reactors. Additionally, licensees were required to provide a response describing actions taken associated with the bulletin. The inspectors reviewed the licensee's letters dated May 23, 1993, May 24, 1993, April 18, 1994, and July 29, 1994, which provided requested information concerning fibrous material in the containment along with immediate compensatory measures required to assure the functional capability of the ECCS and actions to remove any such material. Additionally, the inspectors reviewed TVA letter dated May 6, 2004, which provided an updated response to Bulletin 93-02 for Unit 1. During the more recent response the licensee reported that all actions necessary to resolve this bulletin were taken for Unit 1 based on similar actions taken for Units 2 and 3. The NRC staff reviewed these responses and concluded that all requested information had been provided. In addition, closure of this item prior to restart of Unit 3 was documented in NRC Inspection Report 50-259,260,296/95-60 based on a detailed walkdown of the Unit 3 drywell and review of the licensee's programs for control of fibrous material. Replacement ECCS suction strainers which account for fibrous debris loading have been installed on Unit 1. Licensee activities associated with installation of these strainers was previously documented in Inspection Report 50-259/2005-07. The inspectors determined that actions taken by the licensee for Unit 1 recovery were equivalent to those previously performed on Units 2 and 3. The inspectors determined that this issue was effectively addressed for Unit 1. No further NRC inspection in this area is anticipated. However, final closure of this item will be deferred until NRR completes their review in this area and any SERs, if required, are issued.

E8.10 (Discussed) Bulletin 96-03, Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling Water Reactors

The inspectors reviewed Bulletin 96-03, Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling Water Reactor. Licensees were requested to implement appropriate procedural measures and plant modifications to minimize the potential for clogging of ECCS suction strainers by debris generated during a LOCA. Additionally, this bulletin required licensees report to the NRC the extent of requested actions taken. The inspectors reviewed TVA letters dated November 4, 1996, July 25, 1997, and October 28, 1999, which provided the licensee's response to Bulletin 96-03. Additionally, the inspectors reviewed NRC letter dated November 15, 1999, which closed Bulletin 96-03 for Units 2 and 3 based on installation of high capacity suction strainers, augmented operator training, and the licensee's Foreign Material Exclusion (FME) Program. The inspector reviewed DCN 51200 which provided details associated with the replacement of the Unit 1 suction strainers. Additionally, the inspectors reviewed a video of the strainer replacement. Additional details associated with installation of these strainers in contained in Inspection Report 50-259/2005-07. The inspectors determined that actions taken by the licensee for Unit 1 recovery were equivalent to those previously performed on Units 2 and 3. The inspectors determined that this issue was effectively addressed for Unit 1. No further NRC inspection in this area is anticipated. However, final closure of this item will be deferred until NRR completes their review in this area and any SERs, if required, are issued.

E8.11 <u>(Closed) IFI 50-259/05-09-01, Testing of Cable Damage in Junction Boxes with Missing Conduit Bushings</u>

During a previous review of cable installation activities the inspectors observed that cables in some junction boxes may have been damaged by exposure to conduit edges due to missing bushings. The inspectors were informed by the licensee that the affected cables had been in place since initial installation and limited visual inspections verified that cable jackets were not damaged when pulled. This item was identified to follow resolution of the licensee's resolution of that issue. During the current inspection period the inspectors performed additional reviews of the licensee's sub-program for missing conduit bushings. The inspectors noted that licensee inspections to identify and resolve missing conduit bushings were included in three PERs (04-001435-000,04-000811-000, and 04-000817-000). The inspectors concluded that implementation of this sub-program was adequate and no further action was required. Therefore, because this item is effectively being tracked in the licensee's corrective action program, and because any implementation deficiencies would likely be detected by the licensee's oversight programs, and have only minor consequences, this item meets the closure criteria established for the Unit 1 recovery issues. This issue is closed for Unit 1.

III. Maintenance

M1 Conduct of Maintenance

M1.1 <u>Maintenance Program</u>

a. <u>Inspection Scope</u>

The inspectors continued to observe and/or review ongoing licensee maintenance program activities. Maintenance work activities were controlled by approved procedures and work orders. Specific maintenance activities reviewed and observed included selected portions of ongoing activities associated with return to service of the outboard MSIVs and RHRSW system.

b. <u>Observations and Findings</u>

Licensee maintenance activities reviewed or observed by the inspectors during this report period were associated with the return to service of the outboard Main Steam Isolation Valves (MSIVs) and the Residual Heat Removal Service Water (RHRSW) System. Specific maintenance activities reviewed or observed included the following:

- WO 04-723636-00, 04-723637-00, 04-723638-00, and 04-723639-00 for the satisfactorily completion of leak rate testing of outboard MSIVs, 1-FCV-01-15, 1-FCV-01-27, 1-FCV-01-38, and 1-FCV-01-52 using 0-TI-106, General Leak Rate Test Procedure. The inspectors observed selected work activities and reviewed the leak rate testing results. The inspectors noted that leak rate testing indicated no detectable leakage through all four of the outboard MSIVs which allowed for the removal of the expandable plugs in the main steam piping which provided the temporary extension of the secondary containment boundary. The secondary containment boundary was re-established at the outboard MSIVs.

- WO 05-719158-00, removal of the expandable plugs in the main steam piping, which were installed to extend the secondary containment boundary to support work activities on the outboard MSIVs. Removal of these temporary steam line plugs is discussed in more detail in Section E1.2 of this inspection report.

- WO 04-724879-12, removal and replacement of the 1C RHR Heat Exchanger floating head. During the previous inspection period the inspectors had observed and reviewed ongoing testing associated with the four Unit 1 RHR Heat Exchangers. That NRC review was documented in Inspection Report 50-259/2005-009. During that testing it was discovered that the replacement lower head, referred to as the Floating Head, on the 1C Heat Exchanger leaked. This leakage prevented the completion of the restart testing program for that heat exchanger. During this inspection period the inspectors observed selected work activities and reviewed post maintenance test results associated with repairs for the 1C RHR Heat Exchanger. Work activities observed included removal of the floating heads, installation of the new floating heads, and torquing of the bolting. Following the replacement of the floating head on the 1C heat exchanger additional leakage was detected from the heat exchanger. This required the removal of the new floating head and Eddy Current (ET) examination of tubing. The inspectors reviewed the ET examination results which showed that a significant number of tubes were degraded. At the end of this report period the licensee was considering repair options.

The inspectors reviewed the applicable WO packages and observed selected portions of the ongoing maintenance activities. The inspectors determined that WO packages included sufficient guidance to allow maintenance personnel to adequately perform the associated work activity. Maintenance personnel and foreman were knowledgeable of applicable requirements and appropriately documented work actually performed, as required by plant procedures.

c. Conclusions

No deficiencies were identified during the review of the ongoing maintenance activities. The Maintenance organization continued to provide appropriate and comprehensive repairs to Unit 1 components which do not require design changes to support Unit 1 Restart. Maintenance WO packages included sufficient technical guidance to allow maintenance personnel to adequately perform the associated work activity.

Maintenance personnel and foreman were knowledgeable of applicable requirements and appropriately documented work actually performed, as required by plant procedures.

V. Management Meetings

X1 Exit Meeting Summary

On May 3, 2006, the resident inspectors presented the inspection results to Mr. Masuod Bajestani and other members of his staff, who acknowledged the findings. Although some proprietary information may have been reviewed during the inspection, no proprietary information will be identified in the final inspection report.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

M. Bajestani, Vice President, Unit 1 Restart
R. Baron, Nuclear Assurance Manager, Unit 1
M. Bennett, QC Manager, Unit 1
D. Burrell, Electrical Engineer, Unit 1
P. Byron, Licensing Engineer
J. Corey, Radiological and Chemistry Control Manager, Unit 1
W. Crouch, Nuclear Site Licensing & Industry Affairs Manager
R. Cutsinger, Civil/Structural Engineering Manager, Unit 1
B. Hargrove, Radcon Manager, Unit 1
K. Hess, SWEC Project Director
E. Hollins, Maintenance and Modifications Manager, Unit 1
R. Jackson, Bechtel
R. Jones, General Manager of Site Operations
S. Kane, Licensing Engineer
D. Kehoe, Nuclear Assurance, Unit 1
J. Lewis, Integration Manager
G. Little, Restart Manager, Unit 1
J. McCarthy, Licensing Supervisor, Unit 1
R. Moll, Mechanical Engineering and Systems Engineering Manager, Unit 1
J. Ownby, Project Support Manager, Unit 1
J. Schlessel, Maintenance Manager, Unit 1
J. Symonds, Modifications Manager, Unit 1
J. Valente, Engineering Manager, Unit 1

INSPECTION PROCEDURES USED

IP 37550	Onsite Engineering
IP 37551	Engineering
IP 71111.08	Inservice Inspection Activities
IP 71111.17	Permanent Plant Modifications
IP 71111.23	Temporary Plant Modifications
IP 71152	Identification and Resolution of Problems
IP 92701	Follow-up
IP 50090	Pipe Support and Restraint Systems

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-259/2006-06-01	URI	Adequacy of RHRSW SRTS Activities (Section E1.3)

Closed

95-02	BUL	Unexpected Clogging of RHR Pump Strainer While Operating in Suppression Pool Cooling Mode (Section E8.1)
95-07	GL	Pressure Locking and Thermal Binding of Safety-Related and Power-Operated Gate valves (Section E8.2)
II.B.3	TMI	Post Accident Sampling System (Section E8.3)
II.F.1.2.C	TMI	Accident Monitoring - Containment High Range Radiation (Section E8.4)
05-09-01	IFI	Testing of Cable Damage in Junction Boxes with Missing Conduit Bushings (Section E8.11)

Discussed

88-01	GL	NRC Position on IGSCC in BWR Austenitic Stainless Steel (Section E8.5)
A-46	USI	Seismic Qualification of Equipment in Operating Plants (Section E8.6)
87-02	GL	Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors (Section E8.6)
98-04	GL	Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System after a Loss of Coolant Accident because of Construction and Protective Coating Deficiencies and Foreign Material in Containment (Section E8.7)
97-04	GL	Assurance of Sufficient Net Positive Suction for Emergency Core Cooling and Containment Heat Removal Pumps (Section E8.8)
93-02	BU	Debris Plugging of Emergency Core Cooling Suction Strainers (Section E8.9)
96-03	BU	Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling Water Reactors (Section E8.10)

LIST OF DOCUMENTS REVIEWED

Section E1.1: Plant Modifications

Procedures and Standards

SPP-9.3, Plant Modifications and Engineering Change Control, Revision 9
MAI-4.2B, Piping, Revision 20
G-94, Piping Installation, Modification, and Maintenance, Revision 2
1-POI-64-2, MSIV Secondary Containment System, Rev 0

DCNs

51081, Instrumentation and Control (I & C) Equipment - Control Bay
51084, 500 KV Switchyard and Main Generator, System 242
51166, I & C Equipment - Drywell, for Primary Containment, System 64
51090, 480V AC Distribution - Control Bay, System 57-4
51200, Core Spray Mechanical - Reactor Building, System 74
51214, Electrical 120 VAC Distribution - Control Bay, System 57-2
51222, RHR Electrical - Reactor Building, System 74

Section E1.2: Temporary Modifications

Procedures, Guidance Documents, and Manuals

0-TI-405, Plant Modifications and Design Change Control, Revision 0
0-TI-410, Design Change Control, Revision 1
SPP-9.5, Temporary Alterations, Revision 6

Misc Documents

TACF 1-86-001-057, Provide Instructions to Install a Seismically Qualified Panel Support, Rev 0
EWR 05CEB001101, Main Steam Extended Secondary Containment Boundary Integrity, Rev 1
WO 06-710437-00, Technical Evaluation (TE), Rev 1, temporarily disable portions of the logic for Secondary Containment Isolation

Section E1.3: System Return to Service Activities

Procedures, Guidance Documents, and Manuals

0-SR-3.3.3.2.1(23) Backup Control Panel Test Revision 7
0-SR-3.7.1.1 Valve Position Indication
0-OI-23 Residual Heat Removal Service Water System Revision 60
1-SR-3.3.13 ASME Section XI System Pressure Test of the RHRSW System Revision 0
1-SI-4.5.C.1(3) RHRSW Pump and Header Operability and Flow Test Revision 29
1-TI-437 System Return to Service (SRTS) Turnover Process for U1 Restart Revisions 10/14
1-TI-439 ITEL (Integration Task Equipment List) Revision 10
1-TI-452 Unit 1 Restart Test Program Revision 0
2-SI-3.2.10.B Remote Position Indication

O-TI-63 RHRSW Flow Blockage Monitoring Revision 22
O-TI-517 Simultaneous Operation of RHRSW Pumps D1 and D2 Revision 0
BP-323 Organizational Structure, Roles and Responsibilities for Unit 1 Restart Revision 2 (and drafts)
BP-325 Closure of Open Items Unit 1 Restart Project Revision 3
PMTI-51177-STG-05 Logic Functional Testing for RHRHX 1B RHRSW Valve 1-FCV-23-46 and Standby Coolant Valve from RHRSW 1-FCV-23-57 Revision 0
PMTI-51177-STG-06 Logic Functional Testing for RHRHX 1D RHRSW Outlet Valve 1-FCV-23-52 Revision 0

Calculations
EDQ0-999-2003-0048 Appendix R Manual Actions
MDQ0023880122 RHRSW Design Pressure and Temperature
MDQ110020050013 Thermo-Log Fire Endurance Qualification
MDQ19992003006 Appendix R Fire Suppression Damage Evaluation
ND-Q0999-91033 Safe Shutdown Analysis Revision 20

Drawings
1-47E858-1 Flow Diagram RHR Service Water System Revision 55
0-47E610-23-3 Mechanical Control Diagram RHR Service Water System Revision 25
1-47E859-1 Flow Diagram Emergency Equipment Cooling Water Revision 66

Problem Evaluation Reports (PERs)

90324 RHRSW Pumps D1/2, C2, and B2 <4500 gpm through Unit 2 Heat Exchangers
95286 RHRSW Loops A and C Evaluation
96970 Extent of Condition System 999 Multiple System Calculations
87497 RHRSW Flow Calculation Accuracies
75059 RHRSW Pump New Impeller 25 kw Load Addition
64466 Structural Weld Splices
87688 Seismic Conduit Supports
89023 Missing Large Bore Supports
96516 NSRB Recommendation 6 for Unit 1 Management Oversight of SPOC

Design Change Notices (DCNs)

51177 Stages 4 and 18, System 23 RHRSW Mechanical
51229 Stage 9, System 247 Appendix R Emergency Lighting

Temporary Alteration Control Forms (TACFs)

0-04-004-023 Two Unit Operation Closure of Two Sluice Gates on Same Supply Line Rev 0
1-04-003-023 Piping Jumper to Flush Supply Header Rev 0

Work Orders

02-015487-013 In-service leak test on RHR C Heat Exchanger
03-005591-023 and -078 A and C RHRHX floating head gasket replacements
03-019416-050 RHRSW flow instrument calibrations
04-714532-001 Spring supports
05-721801-000 MOVATS
05-711472-000 Limit switch termination
05-716215-002 In-service leak test on RHR C Heat Exchanger
05-724430-000 A RHRHX floating head replacement

License Issues

NCO90006004 MOVATS
NCO900040015 Generic Letter 89-13 RHRSW Recommended Actions
NCO890016006 Start/Stop and Open/Close for Pumps and Valves
NCO880116007 RHRSW Radiation Monitor
NCO890113023 Q-List
NCO900040008 RHRSW/EECW Design Criteria

Miscellaneous Documents

System Plant Acceptance Evaluation Package (SPAE) System 23 RHRSW Revision 4
System Return to Service-Open Item Punchlist (SRTS-OIP) System 23 RHRSW, Mar 29, 2006
System Pre-Operability Checklist (SPOC) II System 23 RHRSW, Dec 8,2005
SPOC Item Exception PL-05-1600 System 23 RHRSW
SPOC Item Exception PL-05-1635 System 23 RHRSW
SPOC Walkdown System 23 RHRSW, Dec 1, 2005
Special Operating Condition PL-05-1111 System 23 RHRSW
Punchlist Open Item PL-05-1598
1-BFN-BTRD-023 Baseline Test Requirement Document for System 23 RHRSW Revision 3
1-BFN-BTRD-023 Test Summary Report
1-STS-023 System Test Specifications 1-BFN-BTRD-023 Revision 2
Design Criteria Document BFN-50-7023 RHRSW
Final Safety Analysis Report Section 10.9 RHR Service Water System Revision BFN-17
Unit 1 Technical Specifications 3.7.1 RHRSW and Bases B3.7.1
Unit 1 Technical Specifications 3.4.7 RHR SDC Hot, 3.4.8 RHR SDC Cold, 3.5.1 ECCS Operating, 3.5.2 ECCS Shutdown, 3.6.2.3 RHR Suppression Pool Cooling, 3.6.2.4 RHR Suppression Pool Spray, 3.6.2.5 RHR Drywell Spray, 3.9.7 RHR High Water Level, 3.9.8 RHR Low Water Level
Unit 1 (draft) and Units 2/3 Fire Protection Reports Volume I Section 4.0 Part 3.0 Revision 34
Engineering Design Change 65294 Appendix R note addition to Drawing 1-45E751-1
RTPRG-026 Restart Test Program Review Group Meeting Minutes dated Dec 7, 2005
NA Oversight Issues-System Return to Service Corrective Action Status: ITEL Deficiencies, SPAE/SPOC Documentation Issues and Boundaries
NA-BF-06-005 Unit 1 Restart NA-Oversight Analysis Report for the Period of July1 through December 31, 2005
Browns Ferry Excellence Plan, Unit 1 Restart Readiness Focus Area
BFR-REN-006-008 SPAE/SPOC Self Assessment, April 3-21, 2006

Section E1.4: Restart Test Program

Procedures and Standards

Technical Instruction 1-TI-469, Baseline Test Requirements, Rev 1
SSP-3.1, Corrective Action Program, Rev 9
SPP-8.1, Conduct of Testing, Rev 3
SPP-8.3, Post Modification Testing, Rev 6
SSP-9.5, Temporary Alterations, Rev 7
SSP-10.3, Verification Program, Rev 1.

Restart Test Procedures

1-PMTI-51102-STG08 & 10, Stages 8 and 10 of DCN 51102, MCR Panel 1-9-25
1-PMTI-51102-STG8, Stage 8 of DCN 51102, MCR Panel 1-9-25
1-PMTI-BF-51229-STG09, Stage 9 of DCN 51229, Appendix R Reactor Building - Electrical
1-PMTI-BF-51118-STG06, Stage 6 of DCN 51118, RCW Turbine Building - Mechanical, System 24
1-PMTI-BF-51090-STG32, Stage 32 of DCN 51090, 480V Electrical System - Control Bay, System 57-4
1-PMTI-BF-51118-STG05, Stage 5 of DCN 51118, Raw Cooling Water (RCW) Turbine Building - Mechanical, System 24
1-PMTI-BF-51192-STG4, Functional Test of Unit 1 EECW Flow control Valves 1-FCV-67-50 and 1-FCV-67-51
0-SR-3.8.1.1 (B), Diesel Generator B Monthly Operability Test, Rev 32
3-SR-3.8.1.1 (3B), Diesel Generator 3B Monthly Operability Test, Rev 26
0-TI-517, RHRSW Testing
1-TI-496, EECW Flow Test, Rev 04
1-SI-3.2.3, Testing ASME Section XI Check Valves, Rev 1
3-SI-4.5.C.1(2), EECW Pump Operation, Rev 87
0-SR-3.3.3.2.1 (67), Back Up Control Panel Testing, Rev 4
1-SI-4.5.C.1 (2), EECW Pump Operation, Rev 4

Section E1.5: Special Program Activities - Cable Installation and Cable Separation

Procedures and Standards

G-40, Installation, Modification and Maintenance of Electrical Conduit, Cable Trays, Boxes, Containment Electrical Penetrations, Electric Conductor Seal Assemblies, Lighting and Miscellaneous Systems, Rev.15
MAI-3.2, Cable Pulling for Insulated Cables Rated up to 15KV Units 1, 2, and 3, Rev. 41
DS-E13.1.4 Maximum Cable Diameter for Various Rigid Steel Conduits, Rev. 3
MAI-1.3, General Requirements for Modification, Rev 21
MAI-3.2, Cable Pulling for Insulated Cables Rated Up to 15,000 Volts Units 1, 2, and 3, Rev 44
MAI-3.3, Cable Terminating and Splicing for Cables Rated Up to 15,000 Volts Units 1, 2, and 3, Rev 49
MAI-3.7, Cable Pull Force Monitoring Breaklink Fabrication, Verification, and Control, Rev 6

Work Order Packages

04-720414-51, replace control circuit cable for the normal feeder breaker on the A Diesel Auxiliary Board, System 82, and associated internal wiring
04-720414-52, replace control circuit cable for the normal feeder breaker on the B Diesel Auxiliary Board, System 82, and associated internal wiring
04-720414-58 determinate existing cable, pull back, reroute from transfer 0-XSW-248-00C1 switch in compartment 2A to transfer switch in compartment 20A of 480V Shutdown Board 2A, modify compartment 20A to accept new breaker, 0-BKR-248-00C1, and install new cable to provide alternate feed to Battery Charger SB-C
03-002713-08, determinate existing cables, relabel as abandoned, terminate new cables, and label new cables for the B Emergency Diesel Generator, System 82
03-004725-06, determinate existing cables, re-label as abandoned, terminate new cables, and label new cables to Panel 1-LPNL-925-32 for the Unit Preferred 120V AC, System 252
02-011686-09, complete final cable terminations to Primary Containment Isolation System (PCIS), in MCR Panel 1-LPNL-9-3A, Channel A, System 64D
02-016202-53, pull, install, and terminate cables in Panels 25-45A, 45B, and 45C for Core Spray, System 75

Calculations

EDQ1 999 2002 0074, Analysis of Unit 1 Large 600 V cables in Standard Condulets, Rev. 2
EDQ199920030015, Cable Pulling Through 90 Degree Condulets and Mid-Run Flexible Conduits

DCNs

51216, Replacement of TS1E Transformer
51090, Electrical 480V Distribution - Control Bay, System 57-4
51214, Electrical 120V Distribution - Reactor Building, System 57-2
51094, CRDR - Main Control Room, Panel 1-9-3
51223, Core Spray - Electrical, System 75
51217, Electrical 4KV Distribution - Reactor Building, System 57-5

Miscellaneous Documents

WDP-BFN-1-EEB-231-TS1E-VCD-01, TS1E Transformer Cables Configuration and Walk-down Data Rev.0
PIC 61525

Problem Evaluation Reports

04-001435-000, Missing Bushings Damage Four Cables
04-000811-000, Conduit with Missing Bushing Discovered During Walkdown
04-000817-000, EQ verification of walkdown packages

Section E1.7: Special Program Activities - Miscellaneous Steel Frames

Specifications & Procedures

TVA General Engineering Specification G-29A, PS 0.C.1.2, Specification for Welding of Structures Fabricated in Accordance with AISC Requirements for Buildings and Inspected to the Criteria of NCIG-01
TVA General Engineering Specification G-29-S01, PS 4.M.4.4, ASME Section III and Non-ASME (Including AISC, ANSI B31.1 and ANSI B31.5)
TVA General Engineering Specification G-32, Bolt Anchors set in Hardened Concrete, Rev. 21
Procedure No. N-VT-6, Visual Examination of Structural Welds Using the Criteria of NCIG-01, Rev 6
MMDP-10, Controlling Welding, Brazing, and Soldering Processes, Rev. 4, dated 1/15/03
WI-BFN-1-GEN-01, General Requirements for Walkdowns, Rev. 4

Drawings

Drawing numbers 1-48E435-2, through -6 , Structural Steel Framing at El. 541' - 6", Plan, Sections, & Details
Drawing number 1-47E926-2 & -3, Miscellaneous Steel, Heat Exchanger Supports and Access Platforms
Drawing number 1-47E928-2, Miscellaneous Steel, Core Spray Platform, Plan, Sections, & Details
Drawing numbers 1-48E1019-2, Miscellaneous Steel Support Framing below El. 541' - 6", Plan
Drawing numbers 1-48E1019-3 & -4, Miscellaneous Steel Framing at El. 541' - 6", Sections & Details
Drawing number 1-48E991-2, Miscellaneous Steel Pump Support Cooler, Sections & Details
Drawing number 1-48E991-3, Miscellaneous Steel Pump Support Cooler, Sections & Details
Drawing number 1-47N9991/DCA-51375-001, Modifications to Miscellaneous Steel Support Platforms, Plan
As-Built Walkdown Drawing Numbers 541NE, 541NEHGR, 541NW, 541NWHGR, 541SE, 541SEHGR, 541SWHGR, Structural Steel Plan at 541' - 6"

Calculations

Calculation number CDQ1-303-2003-2179, Rev. 0, Evaluation of Unit 1 Core Spray Valve Access Platform at Elevation 604' 8"
Calculation number CDQ1-303-2003-1027, Rev. 0, Reactor Building Qualification, Northeast Platform Steel Framing at Elevation 541.5'
Calculation number CDQ1-303-2003-1029, Revisions 3 and 4, Reactor Building Qualification, Southwest Platform Steel Framing at Elevation 541.5'
Calculation number CDQ1-303-2003-2202, Revisions 2 and 3, Connections for Reactor Building Structural and Miscellaneous Platforms
Calculation number CDQ1-303-2003-2343, Revision 1, Qualification of Anchorages for the NE, NW, SW, & SW Corner Platforms @ Elevation 541' - 6", Reactor Building

Problem Evaluation Reports (PER)

96340, Deficient Welds on Torus Access Platform
96471, Evaluation of Unidentified Bolts on Torus Access Platform.
100727, Bent Flanges on Southwest Quad Elev. 541' 6" Platform

Miscellaneous Documents

TVA Nuclear Engineering Civil Design Standard DS-C1.7.1, General Anchorage to Concrete, Rev 9, dated 8/25/99
Walkdown Package WDP-BFN-1-CEB-303-02-PLT-03, As-Built CS NE Corner Room Structural Steel Access Platform (EL 541' 6")
Walkdown Package WDP-BFN-1-CEB-303-02-PLT-04, As-Built RHR SE Corner Room Structural Steel Access Platform (EL 541' 6")
Walkdown Package WDP-BFN-1-CEB-303-02-PLT-02, As-Built CS NW Corner Room Structural Steel Access Platform (EL 541' 6")
Walkdown Package WDP-BFN-1-CEB-303-02-PLT-01, As-Built CS SW Corner Room Structural Steel Access Platform (EL 541' 6")
Assessment Report BFN-REN-05-010, Miscellaneous Steel Frames
Design Change Notice 51520A Modifications to Miscellaneous Structural steel Platforms in BFN Unit 1, Core Spray Valve Access, RWCU Cleanup Valve Access, and CRD Relief Valve Access Platforms
Design Change Notice 51375 Modifications to Miscellaneous Structural steel Platforms in BFN Unit 1, NW, NE, SW, & SE Quad Rooms, and Support Platforms in the Quad Rooms
Safety Evaluation Report dated July 13, 1992, Subject: Design Criteria for Lower Drywell Steel Platforms and Miscellaneous Steel

Section E1.8: Special Program Activities - Platform Thermal Growth

Specifications & Procedures

TVA General Engineering Specification G-29A, PS 0.C.1.2, Specification for Welding of Structures Fabricated in Accordance with AISC Requirements for Buildings and Inspected to the Criteria of NCIG-01
TVA General Engineering Specification G-29-S01, PS 4.M.4.4, ASME Section III and Non-ASME (Including AISC, ANSI B31.1 and ANSI B31.5)
Procedure No. N-VT-6, Visual Examination of Structural Welds Using the Criteria of NCIG-01, Rev 6
MMDP-10, Controlling Welding, Brazing, and Soldering Processes, Rev. 4, dated 1/15/03

Drawings

Drawing number 1-47E926-2 & -3, Miscellaneous Steel, Heat Exchanger Supports and Access Platforms
Drawing number 1-47E928-2, Miscellaneous Steel, Core Spray Platform, Plan, Sections, & Details
Drawing number 1-48E991-2, Miscellaneous Steel Pump Support Cooler, Sections & Details
Drawing number 1-48E991-3, Miscellaneous Steel Pump Support Cooler, Sections & Details

Drawing number 1-47N9991/DCA-51375-001, Modifications to Miscellaneous Steel Support Platforms, Plan

Calculations

Calculation number CDQ2-303-930315, Rev. 2, Resolution of NRC Comments on Thermal Issues for Miscellaneous Steel

Miscellaneous Documents

Walkdown Package WDP-BFN-1-CEB-303-02-PLT-31, As-Built Misc Steel Supports in the SE Quad as Shown on Drawing 48N1020 (EL 565' 0")
Walkdown Package WDP-BFN-1-CEB-303-02-PLT-18, As-Built Core Spray Pump Cooler Support Corner Room Platform EL 555' 1"
Work Order numbers 03-006335-05, 03-006335-06, and 03-006335-07, and quality control inspection records in these work orders documenting implementing modifications to various pump cooler platforms to address platform thermal growth
Design Change Notice 51375 Modifications to Miscellaneous Structural Steel Platforms in BFN Unit 1, NW, NE, SW, & SE Quad Rooms, and Support Platforms in the Quad Rooms

Section E1.9: Special Program Activities - Control Rod Drive (CRD) Insert and Withdrawal Piping

Specifications & Procedures

TVA General Engineering Specification G-43, Installation, Modification, and Maintenance of Pipe Supports and Pipe Rupture Mitigative Devices, Rev. 13
TVA General Engineering Specification G-32, Bolt Anchors set in Hardened Concrete, Rev. 21
TVA General Engineering Specification G-29A, PS 0.C.1.2, Specification for Welding of Structures Fabricated in Accordance with AISC Requirements for Buildings and Inspected to the Criteria of NCIG-01
TVA General Engineering Specification G-29-S01, PS 4.M.4.4, ASME Section III and Non-ASME (Including AISC, ANSI B31.1 and ANSI B31.5)
Procedure No. N-VT-6, Visual Examination of Structural Welds Using the Criteria of NCIG-01, Rev 6
MAI-4.2A, TVA-BFNP Piping/Tubing Supports, Rev. 33
MMDP-10, Controlling Welding, Brazing, and Soldering Processes, Rev. 4, dated 1/15/03

Drawings

Drawing number 0-47B435-1 through -21, Mechanical General Notes, Pipe Supports
Drawing numbers 1-47E468-114, -115, 116-1, 116-2, and -128, Mechanical Control Rod Drive System Pipe Supports - Floor El 563

Calculations

Calculation number CDQ1-085-2002-1258, Rev. 4, Qualification of Pipe Support No. 1-47B468-115, which included resolution of discrepancies identified in PER 95538

Problem Evaluation Reports (PER)

95096, Discrepancies identified during final as-built verification of CRDH Frame 117
95538, Discrepancies identified during final as-built verification of CRDH Frame 115
95626, Discrepancies identified during final as-built verification of CRDH Frame 124
95942, Discrepancies identified during final as-built verification of CRDH Frame 114
96114, Discrepancies identified during final as-built verification of CRDH Frame 116
96319, Discrepancies identified during final as-built verification of CRDH Frame 132

Miscellaneous Documents

TVA Nuclear Engineering Civil Design Standard DS-C1.7.1, General Anchorage to Concrete, Rev 9, dated 8/25/99
General Design Criteria Document BFN-50-C-7103, Structural Analysis and Qualification of Mechanical and Electrical Systems (Piping and Instrument Tubing), Rev. 5, dated 9/9/91
General Design Criteria Document BFN-50-C-7107, Design of Class I Seismic Pipe and Tubing Supports, Rev. 7, dated 4/6/94
General Design Standard DS-C1.2.6, General Pipe Support Design Manual, Rev. 0
Assessment Report BFN-REN-04-012, CRDH Insert and Withdrawal Piping Program BFN Unit 1 Restart
Work Order numbers 02-008661-05, 02-008661-011, 03-008317-006, and 03-008317-015 and quality control inspection records for CRDH frame numbers 106, 112, 120, and 128
Work Order 06-712186-000, Repair CRDH supports 1-47E468-113, -114, -115, and -116 to resolve discrepancies identified during final as-built walkdown, reference PER 93329, 95942, 96114, & 95538
DCN 51419, Modifications to CRDH Frames in Unit 1 Reactor Building
NRC letter dated December 11, 1989, Browns Ferry Nuclear Plant - Unit 2 - Revised Program Plan - Seismic Qualification of the Control Rod Drive Hydraulic (CRDH) Piping System

Section E1.10: Special Program Activities - Small Bore Piping and Instrument Tubing

Specifications & Procedures

TVA General Engineering Specification G-43, Installation, Modification, and Maintenance of Pipe Supports and Pipe Rupture Mitigative Devices, Rev. 13
TVA General Engineering Specification G-32, Bolt Anchors set in Hardened Concrete, Rev. 21
TVA General Engineering Specification G-29A, PS 0.C.1.2, Specification for Welding of Structures Fabricated in Accordance with AISC Requirements for Buildings and Inspected to the Criteria of NCIG-01
TVA General Engineering Specification G-29-S01, PS 4.M.4.4, ASME Section III and Non-ASME (Including AISC, ANSI B31.1 and ANSI B31.5)
Procedure No. N-VT-6, Visual Examination of Structural Welds Using the Criteria of NCIG-01, Rev 6
MAI-4.2A, TVA-BFNP Piping/Tubing Supports, Rev. 33
MMDP-10, Controlling Welding, Brazing, and Soldering Processes, Rev. 4, dated 1/15/03
WI-BFN-1-GEN-01, General Requirements for Walkdowns, Rev. 4
NEDP-11, Rev. 5, Design Input Walkdown Controls

Drawings

Drawing number 0-47B435-1 through -21, Mechanical General Notes, Pipe Supports
Drawing numbers 1-47B455-3-1, -3-2, -4, -5, -6-1, and -6-2, Mechanical HP Coolant Injection System Pipe Support
Drawing numbers 1-47B600-2658 , -2659, -2662, and -2664, Reactor Feedwater System I & C Pipe Support (RVLIS)
Drawing numbers 1-47B451-3057, -3058, -3059, & -3061 EECW System Pipe Support
Drawing numbers 1-47B600-250-1, -250-2, -251-1, -251-2, -252-1, -252-2 - 255-1, -255-2, -256-1, -256-2, and -257, Main Steam System Pipe Support
Drawing numbers 1-47B600-2673, -2674, -2675, -2676, -2677, -2678, -2679, and -2681,Reactor Water Recirculation System Pipe Support

Calculations

Calculation number CDQ1-067-2003-2440, Rev. 0, Qualification of Pipe Support Nos. 1-47B451-3058 and 1-47B451-3059
Calculation number CDQ1-067-2003-2442, Rev. 0, Qualification of Pipe Support No. 1-47B451-3061
Calculation number CDQ1-067-2003-2453, Rev. 0, Small Bore Piping Program Qualification for the Unit 1 Seismic Class 1 EECW System 67 Piping
Calculation number CDQ1-073-2002-1062, Rev. 3, Small Bore Piping Program Qualification for the Unit 1 Seismic Class 1 HPCI System 73 Piping

Problem Evaluation Reports (PER)

94867, Slope of RVLIS Piping Did Not Conform to Criteria
95277, Unistrut on Support Damaged by Weld
96392, Drawing Discrepancies Pertaining to Incorrect Valve Numbers on Reactor Water Recirculation Small Bore Piping
96949, Missing Tube Track at Support 1-47B448-3071
97445, Violation of Expansion Anchor Installation Requirements for Support 1-47B600-5230
100686, Minimum Clearance Between RVLIS Tubing and Support 1-47B600-2658 Not Met

Miscellaneous Documents

TVA Nuclear Engineering Civil Design Standard DS-C1.7.1, General Anchorage to Concrete, Rev 9, dated 8/25/99
General Design Criteria Document BFN-50-C-7103, Structural Analysis and Qualification of Mechanical and Electrical Systems (Piping and Instrument Tubing), Rev. 5, dated 9/9/91
General Design Criteria Document BFN-50-C-7107, Design of Class I Seismic Pipe and Tubing Supports, Rev. 7, dated 4/6/94
General Design Standard DS-C1.2.6, General Pipe Support Design Manual, Rev. 0
Assessment Report BFN-REN-04-007, Small Bore Piping Program BFN Unit 1 Restart
Assessment Report BFN-RMM-06-002, Unit 1 79-14 Modification Implementation and Review of As-Built Verification Process
DCN 51163, Modifications to Small Bore Pipe Supports for Unit1 Emergemcy Equipment Cooling Water (EECW) System 67

DCN 51411A, Modifications to Small Bore Pipe Supports for Unit1 Emergemcy Equipment Cooling Water (EECW) System 67
DCN 51333, Additions and Modifications to Small Bore Pipe Supports for Unit1 Reactor Vessel Level Instrumentation (RVLIS) System
Work Order numbers 02-012029-007 and 02-012029-008 and quality control inspection records for small bore support numbers 1-47B600-2658 and -2659
Engineering Change Control Documents, Post Issue Change (PIC) numbers 60091, 62406, 62457, 62458, 62754, 62974, and 65958

Section E1.11: Special Program Activities - Containment Coatings

Specifications & Procedures

TVA General Engineering Specification G-55, Technical and Programmatic Requirements for the Protective Coating Program for TVA Nuclear Plants
0-SI-4.7.A.2.K, Primary Containment Drywell Surface Visual Inspection, Rev. 12
0-TI-417, Inspection of Service Level I, II, III Protective Coatings, Rev. 4
WI-BFN-1-GEN-01, General Requirements for Walkdowns, Rev. 4
WI-BFN-1-MEB-03, BFN Unit 1 Primary Containment Coatings Inspection Plan, Rev. 1

Problem Evaluation Reports (PER)

96634, Equipment Malfunctions Resulted in Incorrect Drywell Vessel Wall Thickness Readings
96670, Recordable Indications in Drywell Vessel Base Metal

Miscellaneous Documents

2006 Drywell Coating Inspection Records - Summary of Components with Unqualified Coating - area and average dry film thickness of existing coating
Notification of Indication Report Forms, numbers U1C6R-001 through -004, U1C6R-007 through -011, U1C6R-014, U1C6R-019, U1C6R-020, U1C6R-027, U1C6R-028, and U1C6R-032
Work Order 02-005150-000, Drywell Vessel Adhesion Test Results
Results of Long Term Corrosion Study Performed on Unit 1 Drywell Vessel at 6 Areas Located Adjacent to Moisture Barrier/Concrete Floor since May, 1987
TVA letter to NRC, dated October 4, 1989, Subject: Browns Ferry Nuclear Plant - Containment Coatings
NRC Safety Evaluation Report, NUREG-1232, Vol. 3, Supp. 2, Browns Ferry Restart, Section 3.7, Containment Coatings

Section E1.7: Special Program Activities - Large Bore Piping and Supports

Procedures

Procedure No., WI-BFN-0-CEB-01, Walkdown Instruction for Piping and Pipe Supports

Drawings

ISO. N1-174-5R, Residual Heat Removal System, Sheet 1to 4
ISO. N1-123-2R, Residual Heat Removal Service Water Line System, Sheet 6 & 7
ISO. N1-185-2R, CRD System, Sheet 9 to 28
Pipe Support Drawing No. 1- 47B452-1465, Rev. R003
Pipe Support Drawing No. 1- 47B452-1466, Rev. R002
Pipe Support Drawing No. 1- 47B452-1467, Rev. R003
Pipe Support Drawing No. 1-47B452-1468, Rev. R003
Pipe Support Drawing No. 1-47B452-1469, Rev. R001
Pipe Support Drawing No. 1-47B452-1470, Rev. R003
Pipe Support Drawing No. 1-47B452-1472, Rev. R001
Pipe Support Drawing No. 1-47B452-1473, Rev. R003
Pipe Support Drawing No. 1-47B452-1474, Rev. R003
Pipe Support Drawing No. 1-47B452-1475, Rev. 004
Pipe Support Drawing No. 1-47B452-1476, Rev. R003
Pipe Support Drawing No. 1-47B452-1479, Rev. R002
Pipe Support Drawing No. 1-47B452-1480, Rev. R002
Pipe Support Drawing No. 1-47B450-260, Rev. R003
Pipe Support Drawing No. 1-47B450-261, Rev. R003
Pipe Support Drawing No. 1-47B450-262, Rev. R002
Pipe Support Drawing No. 1-47B450-263, Rev. R003
Pipe Support Drawing No. 1-47B450-266, Rev. R004
Pipe Support Drawing No. 1-47B450-267, Rev. R005
Pipe Support Drawing No. 1-47B450-268, Rev. R003
Pipe Support Drawing No. 1-47B450-341, Rev. R003
Pipe Support Drawing No. 1-47B450-446, Rev. R006
Pipe Support Drawing No. 1-47B468-256, Rev. R003
Pipe Support Drawing No. 1-47B468-257, Rev. R003
Pipe Support Drawing No. 1-47B468-258, Rev. R005
Pipe Support Drawing No. 1-47B468-261, Rev. R003
Pipe Support Drawing No. 1-47B468-265, Rev. R003
Pipe Support Drawing No. 1-47B468-277, Rev. R003
Pipe Support Drawing No. 1-47B468-282, Rev. R004
Pipe Support Drawing No. 1-47B468-284, Rev. R003
Pipe Support Drawing No. 1-47B468-286, Rev. R002
Pipe Support Drawing No. 1-47B468-294, Rev. R003
Pipe Support Drawing No. 1-47B468-295, Rev. R004

Problem Evaluation Reports

91239
95156

Section E1.12 Special Program Activities - Long Term Torus Integrity

Procedures and Design Criteria

Procedure No., WI-BFN-0-CEB-01, Walkdown Instruction for Piping and Pipe Supports
Design Criteria BFN-50-C-7100, Design of Civil Structures, Attachment A - General Design Criteria for the Torus Integrity Long Term Program, Rev. 16
Design Criteria BFN-50-C-7107, Design of Class I Seismic Pipe and Tubing Supports, Rev. 7

Other Documents

ISO. N1-175-1R, Torus Analysis of Core Spray Piping System
Support Calculation CDQ1-075-2003-1427 for Support 1-47B458-815
Support Calculation CDQ1-075-2003-1431 for Support 1-47B458-819
Support Calculation CDQ1-075-2003-1432 for Support 1-47B458-820
Pipe Support Drawing No. 1- 47B458-815, Rev. R001
Pipe Support Drawing No. 1- 47B458-819, Rev. R000
Pipe Support Drawing No. 1- 47B458-820, Rev. R000

Section E1.13: Boiling Water Reactor Vessel Internals Program (BWRVIP) Activities

Specifications & Procedures

TVA Procedure, Technical Instruction, 0-TI-365, Reactor Pressure Vessel Internals Inspections, Units 1, 2, and 3, Rev 17
1-SI-4.6.G, Inservice Inspection Program - Unit 1, Rev 5
0-TI-365, Reactor Pressure Vessel Internals Inspections (RPVII) Units 1, 2, and 3, Rev 17
GE Procedure, GE-VT-204, Procedure for In-vessel Visual Inspection (IVVI) of BWR4 RPV Internals, Rev. 8
GE-ADM-1025, Procedure for Training and Qualification for QE Specilized NDE Applications, Rev 8
GE-ADM-1046, Process for Analysis of Ultrasonic Data for BWR Core Shroud Assembly Welds, Rev 8
GE-UTM-300, Procedure for Manual Examination of Reactor Vessel Assembly Welds In Accordance with PDI, Rev 9
BWRVIP-03, Reactor Pressure Vessel and Internals Examination Guidelines
BWRVIP-138, BWR Jet Pump Beam Examination Guidelines

Miscellaneous Documents

Final Unit 1 Phase I IVVI Report, December 2005
Final Unit 1 ISI Report, BFN-1C06R-KCZKG, September 2005
Final Unit 1 Phase II IVVI Report, BFN-1C06R-MJJYJ, February 2006

Section E7.1: Licensee Quality Assurance Oversight of Recovery Activities (Identification and Resolution of Problems)

Miscellaneous Documents

WO 03-013930-006, Perform BME on 14 inch HPCI line near support 1-47B455-2260
Base metal Evaluation 1-03013930-006-S22-BME-001, HPCI pipe damage evaluation
Liquid penetrant Exam Report of 14 inch HPCI pipe dated February 27, 2006
Ultrasonic Thickness Measurement Report of 14 inch HPCI pipe dated February 28, 2006

Section E8.6: Generic Letter 87-02, Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors and USI A-46, Seismic Qualification of Equipment in Operating Plants

Problem Evaluation Reports (PERs)

64133, Hoist and chains from the overhead jib crane in the drywell are free to swing and pose a seismic interaction.
64136, Two bolts missing on Junction Box 1-JB-1079 containing hand switches 1-HS-74-52B and -53B.
64139, Grout under the base channels for instrument racks 1-LPNL-925-006A is deteriorated
64141, Breaker lifting device on the top of the 1-BDBB-231-0001A switchgear cabinet is not properly parked in place

DCNs

51085, BFNP Unit 1 Recovery - Electrical Lead DCN - System 252
51202, U1 Recovery Reactor Building System 077 (Radwaste System)
51521, U1 Recovery Reactor Building Structural Modification Required by A-46 evaluation.

Calculations

CDQ1-999-2003-1199, Relay Safe Shutdown Equipment List (SSEL) for A-46 and Seismic IPEEE Programs
CDQ1-999-2003-0654, Composite Safe Shutdown Equipment List (SSEL) for USI A-46 A-46.and Seismic IPEEE Programs - BFN Unit1.

Work Orders

04-715781-000, Outlier OSVS 19195-01, local panel 1-LPNL-925-0005A, bolt connecting one of the floor mounted brace is missing
04-715782-000, Outlier OSVS 10007-01: valve 1-FCV-85-83 actuator diaphragm housing is in contact with the sharp end of a cut off beam section of a pipe support.

Drawings

1-47E200-3
1-48N1114 (DCA 51521-001)
0-47E605-1A (DCA 51085-202)
0-45N230 (DCA 51085-201)
1-45N812-1 (DCA 51085-101)
1-45W804-1 (DCA 51085-100)
1/48B900-3085
1-48B500-3644
0-45N800-19 (DCA 51190-159)

Other Documents

“USI A-46 / Seismic IPEEE Relay Evaluation at BF1", January 2004
“BF1 USI A-46 Seismic Evaluation Report”, September 2004
“Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment“, March 1993 - Seismic Qualification Utility Group (SQUG)

Section M1: Conduct of Maintenance

Procedures and Standards

SPP-10.2, Clearance Program, Revision 6
TI-106, General Leak Rate Test Procedure, Revision 10

Work Orders

04-723636-00, leak rate test of the 1A outboard MSIV 1-FCV-01-15
04-723637-00, leak rate test of the 1B outboard MSIV 1-FCV-01-27
04-723638-00, leak rate test of the 1C outboard MSIV 1-FCV-01-38
04-723639-00, leak rate test of the 1D outboard MSIV 1-FCV-01-52
05-719158-00, removal of the expandable plugs in the main steam piping
04-724879-12, removal and replacement of the 1C RHR Heat Exchanger floating head